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NUREG/CR-2800
PNL-4297
Supplement 2

Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development

Prepared by W. B. Andrews, R. H. V. Gallucci, G. J. Konzek, S. W. Heaberlin, B. A. Fecht, C. H. Allen, R. D. Allen, W. E. Bickford, E. H. Carbaugh, J. R. Lewis, M. A. McKinnon, M. A. McLean, T. B. Powers, G. E. Russcher, R. E. Schreiber, S. A. Weakley, R. D. Widrig

Pacific Northwest Laboratory
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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

This is the third in a series of reports to document the use of a methodology developed by the Pacific Northwest Laboratory to calculate, for prioritization purposes, the risk, dose and cost impacts of implementing resolutions to reactor safety issues (NUREG/CR-2800, Andrews et al. 1983). This report contains results of issue-specific analyses for 31 issues. Each issue was considered within the constraints of available information as of summer 1983, and two staff-weeks of labor. The results are referenced, as one consideration in setting priorities for reactor safety issues, in NUREG-0933, A Prioritization of Generic Safety Issues.

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 to quantify risk, dose and cost impacts of resolutions to reactor safety issues and apply it to issues of interest to the NRC. Results of this project will be used by the NRC to support, in part, decisions on resource allocation to resolve specific issues. Prioritization decisions by the NRC are documented in NUREG-0933, A Prioritization of Generic Safety Issues.

This is the third in a series of reports from the PSI project. The first report contains a description of the methodology and three example issue analyses. The second report contains results for 15 additional issues. This report contains results of analyses for 31 more issues. Future supplements are planned to document additional issues.

The following is a list of issues published in previous volumes:

NUREG/CR-2800 (PNL-4297)	
18	Steam Line Break with Consequential Small LOCA
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
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

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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

IV.E.5 Safety Decision Making--Assess Currently Operating Reactors

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CONTENTS

1.0 INTRODUCTION

This report documents the use of a methodology developed by the Pacific Northwest Laboratory[a] to provide the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Reactor Regulation (NRR) with information to use in prioritizing 31 safety issues related to nuclear power plants. Estimates in this report, along with other subjective factors, were used by the NRC to rank safety issues for further investigation or possible implementation. The safety issue ranking decisions made by NRC are documented in NUREG-0933 (NRC 1983).

This document is not intended to stand alone. A summary of risk, dose and cost factors considered in the issue analyses is provided in this section to delineate the scope of work for each issue. Details of the methodology, data and format are contained in NUREG/CR-2800 (Andrews et al. 1983).

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

(a) Operated by Battelle Memorial Institute.

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

1.1 PUBLIC RISK REDUCTION

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

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

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

where N = number of reactors affected by the safety issue resolution (SIR)

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

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

1.2 OCCUPATIONAL DOSE

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

G = occupational dose increase due to implementation and O/M of the SIR

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

where N = number of reactors affected by the SIR

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

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

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

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

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

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

where N = number of reactors affected by the SIR

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

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

1.3 COSTS

Costs incurred for implementing the SIR include

1) the cost to the NRC for developing each requirement and reviewing the utility's design to assure that the requirement is properly implemented, operated, and maintained

2) the utility's cost of design, procurement, installation, and testing to implement the requirement and its cost for O/M.

Accident avoidance results in cost savings to the utility. Information on both NRC and industry costs is considered since both represent costs that are paid by the public, either as taxpayers or ratepayers. Only future costs are relevant to current decisions, so sunk costs are ignored. All costs are considered to be in 1982 dollars.

1.3.1 NRC Costs

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

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

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

where N = number of plants affected by the SIR

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

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

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

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

1.3.2 Industry Costs

Industry costs are defined as follows:

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

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

where N = number of reactors affected

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

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

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

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

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

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

where N = number of reactors affected

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

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

REFERENCES FOR SECTION 1.0

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

US NRC. 1983. A Prioritization of Generic Safety Issues. NUREG-0933, U.S. Nuclear Regulatory Commission, Washington, D.C.

2.0 ISSUE ANALYSES

Thirty-one issue analyses are described in this section. All are similar in format and contain the following components:

Safety Issue Summary Work Sheet	- Results are summarized for the issue.
Section 1.0, Issue Description	- The safety issue resolution (SIR) and affected plants are described.
Section 2.0, Safety Issue Risk and Dose	- Analysis of public risk reduction and the occupational dose resulting from the SIR is presented. Results are summarized in the Public Risk Reduction Work Sheet and the Occupational Dose Work Sheet, respectively.
Section 3.0, Safety Issue Costs	- Analysis of industry and NRC costs attributable to the SIR is presented. Results are summarized in the Safety Issue Cost Work Sheet.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 15, Radiation Effects on Reactor Vessel Support Structures

SUMMARY OF PROBLEM ANO PROPOSED RESOLUTION

Low-energy neutrons are very plentiful in the vicinity of power reactor vessel support structures (RVSSs). A large neutron fluence can induce changes in the nil ductility transition temperature and cause some loss of fracture toughness in structural steel. One potential solution at operating plants is the application of local heaters to maintain the RVSS well above the range of brittle fracture temperatures. At planned plants, use of nonsusceptible structural steel is the preferred solution.

AFFECTED PLANTS			
	BWR:	Operating = 5	Planned = 4
	PWR:	Operating = 9	Planned = 9

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REOUCTION =	3100
OCCUPATIONAL DOSES:	
SIR Implementation =	3000
SIR Operation/Maintenance =	6500
Total of Above =	9500
Accident Avoidance =	39

COST RESULTS ($$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	7.3
SIR Operation/Maintenance =	18
Total of Above =	25
Accident Avoidance =	3.2
NRC COSTS:	
SIR Development =	0.54
SIR Implementation Support =	0.95
SIR Operation/Maintenance Review =	0.28
Total of Above =	1.8

RADIATION EFFECTS ON REACTOR VESSEL SUPPORT STRUCTURES

ISSUE 15

1.0 SAFETY ISSUE DESCRIPTION

The potential problem addressed by this issue is radiation embrittlement of structural materials. In the past, most neutron damage has been associated only with those whose energy is $\geq$1 MeV. However, it has also been recognized that neutrons whose energy lies between 0.1 and 1 MeV also contribute to damage (McElroy 1982). An upward shift in the nil ductility transition temperature (NDTT) has been related to high fluence exposure from low-energy neutrons. They are very plentiful in the vicinity of a reactor vessel support structure (RVSS) because most of the fast neutrons (E>1 MeV) have been moderated or shielded in leaving the reactor vessel. The transition temperature for brittle failure of many structural steels begins in the neighborhood of -50°F, but after high exposure to a neutron fluence, the transition temperature can become as high as 200°F. This means that loss of fracture toughness may become evident in a rapidly propagating fracture of the RVSS and consequent movement of the RV, given an accident condition which provides a transient stress or shock (e.g., an earthquake).

Several solutions to the problem have been considered; prime candidates are as follows. At operating plants, the choices are 1) to provide local heating and insulation for the RVSS to keep it well above the NDTT, and 2) to reinforce the RVSS in those areas where fracture toughness loss may no longer enable the RVSS to meet seismic requirements. For the purposes of estimating the risk, dose, and cost associated with this issue, the first of these two resolutions is presumed to apply generically to operating plants. It is recognized that plant-specific fixes will be employed, but the scope of this analysis precludes plant-specific assessment. At planned plants, the problem can be precluded by using structural steel which is nonsusceptible to this shift in NDTT. This safety issue resolution (SIR) is assumed for those plants.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with issue resolution are estimated in this section. The analysis results are summarized in Tables 1 and 2.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Radiation Effects on Reactor Vessel Support Structures (15)

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

   The research needed to understand the problem has been underway since about 1979, but little data are available (McElroy 1982; Marston 1980). The severity of the problem is not well documented, nor are the susceptible RVSS materials identified. Consequently, to represent the technical community opinion, the probability that a problem exists is assumed to be 0.5. In addition, to represent the uncertainty as to which RVSS materials may be susceptible, the probability that a plant is vulnerable is assumed to be 0.4. Thus, 20% of all BWRs and PWRs are assumed to be affected. However, plants should be susceptible to this problem only during the last third of their lifetimes. Thus, the average remaining lifetimes of the affected plants are taken to be one-third of the values given in Appendix C of Andrews et al., 1983.

	N	$\bar{T}$ (yr)
PWR	18	9.6
BWR	9	9.1

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf 1 - representative BWR

4. Parameters Affected by SIR:

   Oconee: SS_1, SS_2, and SS_3

   Grand Gulf: SS

   These are seismically-induced LOCA initiators analogous to S_1, S_2, S_3, and S.[a]

5. Base-Case Values for Affected Parameters:

   Oconee: SS_1 = 1.2E-4/py[a]
   SS_2 = 1.4E-4/py[a]
   SS_3 = 1.8E-4/py[a]

   Grand Gulf: SS = 1.8E-4/py[a]

(a) See Attachment 1.

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies:

Oconee

SS_3H -	γ (PWR-3)	=	6.9E-7/py
	β (PWR-5)	=	1.0E-8/py
	ϵ (PWR-7)	=	6.9E-7/py
SS_1D -	α (PWR-1)	=	8.0E-8/py
	γ (PWR-3)	=	1.6E-6/py
	β (PWR-5)	=	5.9E-8/py
	ϵ (PWR-7)	=	6.4E-6/py
SS_3FH -	γ (PWR-2)	=	2.9E-7/py
	β (PWR-4)	=	4.2E-9/py
	ϵ (PWR-6)	=	2.9E-7/py
SS_2FH -	α (PWR-1)	=	4.6E-9/py
	β (PWR-4)	=	3.3E-9/py
	ϵ (PWR-6)	=	3.6E-7/py
SS_2D -	α (PWR-1)	=	7.0E-9/py
	γ (PWR-3)	=	1.4E-7/py
	β (PWR-5)	=	5.1E-9/py
	ϵ (PWR-7)	=	5.6E-7/py
SS_3D -	γ (PWR-3)	=	9.7E-8/py
	β (PWR-5)	=	1.4E-9/py
	ϵ (PWR-7)	=	9.7E-8/py

Grand Gulf

SI -	α (BWR-1)	=	5.9E-9/py
	δ (BWR-2)	=	5.9E-7/py

The seismically-induced LOCA initiators SS_1, SS_2, SS_3, and SS are assumed to generate the same accident sequences as their corresponding LOCA initiators, S_1, S_2, S_3, and S, respectively.

TABLE 1. (contd)

7. Affected Release Categories and Base-Case Frequencies:

Oconee	Grand Gulf
PWR-1 = 9.2E-8/py	BWR-1 = 5.9E-9/py
PWR-2 = 2.9E-7/py	BWR-2 = 5.9E-7/py
PWR-3 = 2.5E-6/py	
PWR-4 = 7.6E-9/py	
PWR-5 = 7.5E-8/py	
PWR-6 = 6.5E-7/py	
PWR-7 = 7.8E-6/py	

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

$\bar{F}$(PWR) = 1.1E-5/py $\bar{F}$(BWR) = 5.9E-7/py

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

W(PWR) = 16 man-rem/py W(BWR) = 4.2 man-rem/py

10-14. Steps Related to Adjusted-Case Values of Affected Parameters, Accident Sequences, Release Categories, Core-melt Frequency, and Public Risk:

SIR is assumed to virtually eliminate the potential for radiation embrittlement of the RVSS. Thus, the adjusted-case, affected core-melt frequency and public risk are essentially zero.

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

$\Delta\bar{F}$(PWR) = 1.1E-5/py $\Delta\bar{F}$(BWR) = 5.9E-7/py

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

ΔW(PWR) = 16 man-rem/py ΔW(BWR) = 4.2 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
3100	9.3E+4	0

ATTACHMENT 1

The assumed accident scenario is occurrence of a seismic event of sufficient magnitude (presumably a 0.2-g peak ground acceleration) to cause fracture of an embrittled RVSS, subsequent movement of the RV, and a corresponding LOCA as attached piping ruptures or crimps. The accident sequences will be analogous to those for the S_1, S_2, and S_3 initiators for Oconee and the S initiator for Grand Gulf since these are the corresponding LOCA initiators due to pipe rupture. However, only seismically-induced pipe rupture is of concern here. Since the original Oconee and Grand Gulf studies did not specifically address seismically-induced LOCAs, it is necessary to define such LOCA initiators for the base case.

Corresponding to the four LOCA initiators mentioned above (S_1, S_2, S_3, and S) are four seismically-induced LOCA initiators, SS_1, SS_2, SS_3, and SS, assumed to be the affected parameters for Oconee (SS_1, SS_2, SS_3) and Grand Gulf (SS). Their base-case frequencies are estimated as follows.

$\underline{SS_1}$

SS_1 = Rupture of reactor coolant systems (RCS) or loss of flow in piping with $10" < d < 13.5"$ due to a seismic event where the peak ground acceleration (PGA) is $\geq$ 0.2 g.

$f(SS_1) = f(PGA \geq 0.2\ g) \cdot p(NDTT) \cdot p(FB)$
$(7E\text{-}4/\overline{py})(0.33)(0.5) = 1.2E\text{-}4/py$

where $f(PGA \geq 0.2\ g)$ = frequency of $PGA \geq 0.2$ g (from WASH-1400, NRC 1975)

p(NDTT) = probability of NDTT-induced susceptibility of RVSS to failure with subsequent RV movement (assumed to be one in three, or 0.33)

p(FB) = probability of RCS flow blockage or pipe rupture due to RV movement (assumed to be 0.5 for piping with $10" < d < 13.5"$)

$\underline{SS_2}$

SS_2 = Rupture of RCS or loss of flow in piping with $4" < d < 10"$ due to a seismic event where the PGA is $\geq$ 0.2 g

$f(SS_2) = f(PGA \geq 0.2\ g) \cdot p(NDTT) \cdot p(FB)$
$(7E\text{-}4/\overline{py})(0.33)(0.6) = 1.4E\text{-}4/py$

where $f(PGA \geq 0.2\ g)$ and p(NDTT) are as before p(FB) = probability of RCS flow $\overline{b}$lockage or pipe rupture due to RV movement (assumed to be 0.6 for piping with $4" < d < 10"$)

$\underline{SS_3}$

SS_3 = Rupture of RCS or loss of flow in piping with d<4" due to a seismic event where the PGA is $\geq$ 0.2 g.

$f(SS_2) = f(PGA > 0.2\ g) \cdot p(NDTT) \cdot p(FB)$
$(7E\text{-}4/\overline{py})(0.33)(0.8) = 1.8E\text{-}4/py$

where $f(PGA \geq 0.2\ g)$ and $p(NDTT)$ are as before,

p(FB) = probability of RCS flow blockage or pipe rupture due to RV movement (assumed to be 0.8 for piping with d<4")

<u>SS</u>

SS = Small LOCA (rupture area <1 ft^2) or loss of flow due to a seismic event where the PGA is $\geq$ 0.2 g.

$f(SS) = f(SS_3) = 1.8E\text{-}4/py$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Radiation Effects on Reactor Vessel Support Structures (15)

2. Affected Plants (N):

   Affected operating plants will presumably implement the SIR (provide local heating and insulation of their RVSSs) after two-thirds of their operating lives have expired. Affected planned plants will implement the SIR (use nonsusceptible structural steel for their RVSSs) during construction. Thus, no occupational dose will be incurred during SIR implementation, operation, and maintenance at planned plants, and they may be viewed as unaffected for these calculations. However, occupational dose reduction due to accident avoidance will be realized at these planned plants over the last one-third of their operating lifetimes.

		N	
PWRs:	Operating	9	
	Planned	9	(accident-avoidance dose only)
	Total	18	
BWRs:	Operating	5	
	Planned	4	(accident-avoidance dose only)
	Total	9	

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

		$\bar{T}$ (yr)	
PWRs:	Operating	9.2	
	Planned	10	(accident-avoidance dose only)
	Total	9.6	
BWRs:	Operating	8.4	
	Planned	10	(accident-avoidance dose only)
	Total	9.1	

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

$$\Delta(\bar{FD}_R)_{PWR} = (19{,}900 \text{ man-rem})(1.1E\text{-}5/py) = 0.22 \text{ man-rem/py}$$

$$\Delta(\bar{FD}_R)_{BWR} = (19{,}900 \text{ man-rem})(5.9E\text{-}7/py) = 0.012 \text{ man-rem/py}$$

TABLE 2. (contd)

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
39	230	0

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

Labor in radiation zones to fabricate, install, and test the RVSS heaters and insulation at operating plants will presumably require 54 man-wk/plant (including a contingency factor of 50%). This estimate applies to both PWRs and BWRs.

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

It is assumed that radiation fields of 0.1 R/hr exist in the vicinity of the reactor vessel.

D = (54 man-wk/plant)(40 man-hr/man-wk)(0.10 R/hr) = 216 man-rem/plant

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

ND = 14(216 man-rem/plant) = 3020 man-rem

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

Based on the information in Step 9 of Table 3, labor in radiation zones for SIR operation and maintenance will presumably require the following:

Routine Operation & Maintenance	=	4 man-wk/py
Periodic Heater Replacement (including fabrication, installation, testing, but excluding staff retraining--taken to be 75% of estimate in Step 9 of Table 3)	=	9 man-wk/py
Total	=	13 man-wk/py

This applies only to operating plants (both PWRs and BWRs). Planned plants will employ nonsusceptible materials in their RVSSs, and no additional labor is foreseen beyond that for routine inspection and maintenance.

TABLE 2. (contd)

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

Again, assuming a 100 mR/hr radiation field,

D_o = (13 man-wk/py)(40 man-hr/man-wk)(0.10 R/hr) = 52 man-rem/py

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

$N\bar{T}D_o$ = [9(9.2 yr) + 5(8.4 yr)](52 man-rem/py) = 6490 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
9500	2.9E+4	3200

3.0 SAFETY ISSUE COSTS

In Section 1.0, two prime candidates for solutions to the problem at operating plants were presented: the first, to provide local heating and insulation; the second, to reinforce areas where fracture toughness loss would occur. Costs for the first candidate are assumed to be representative of the total costs for issue resolution at operating plants. Thus, only this candidate's costs are estimated. This is not to imply that the first candidate is the better.

At planned plants, the problem is precluded by the assumed SIR of using nonsusceptible structural steel to the shift in NDTT. Since this can be incorporated during initial plant design, no additional costs are foreseen beyond those normally incurred during design.

The SIR would probably provide protection only during the latter third of a reactor's life. Implementation of the SIR is thus assumed to occur after two-thirds of the plants' operating lives have expired. The industry and NRC cost analyses results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Radiation Effects on Reactor Vessel Support Structures (15)

TABLE 3. (contd)

2. Affected Plants (N):

Affected operating plants will presumably implement the SIR (provide local heating and insulation of their RVSSs) after two-thirds of their operating lives have expired. Affected planned plants will implement the SIR (use nonsusceptible structural steel for their RVSSs) during construction. Thus, no industry cost will be incurred during SIR implementation, operation, and maintenance at planned plants, and they may be viewed as unaffected for these calculations. However, industry cost savings due to accident avoidance will be realized at these planned plants over the last one-third of their operating lifetimes.

		N	
PWRs:	Operating	9	
	Planned	9	(accident-avoidance cost only)
	Total	18	
BWRs:	Operating	5	
	Planned	4	(accident-avoidance cost only)
	Total	9	

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

		$\bar{T}$ (yr)	
PWRs:	Operating	9.2	
	Planned	10	(accident-avoidance cost only)
	Total	9.6	
BWRs:	Operating	8.4	
	Planned	10	(accident-avoidance cost only)
	Total	9.1	

Industry Costs (Steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, $\Delta(\bar{FA})$:

$$\Delta(\bar{FA})_{PWR} = (\$1.65E+9)(1.1E-5/py) = \$1.8E+4/py$$

$$\Delta(\bar{FA})_{BWR} = (\$1.65E+9)(5.9E-7/py) = \$970/py$$

TABLE 3. (contd)

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

Best Estimate	Upper Bound	Lower Bound
\$3.2E+6	\$1.9E+7	0

6. Per-Plant Industry Resources for SIR Implementation:

The resources required to implement the SIR at each of the affected operating plants are labor and equipment. It is assumed that heaters will be attached to four reactor vessel support columns and that mounting hardware, wiring, metal-sheathed heating cables, switchgear, transformers, and a power controller will be installed. It is also assumed that the equipment would be installed during scheduled reactor outages, thus requiring no additional replaced power. It is further assumed that access to the reactor cavity would be possible for the heater installation.

Equipment per plant:

- 4 strip heaters clamped to support columns
- Mounting hardware, materials, wiring
- Power controller
- Switchgear, transformers
- Metal-sheathed heating cables

Labor per plant:

195 man-wk (including 50% contingency factor)

These estimates apply to both PWRs and BWRs and are developed in Attachment 2.

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

Equipment Cost (including 50% contingency)	= \$5.2E+4/plant[(a)]
Labor Cost = (195 man-wk/plant)(\$2270/man-wk)	= \$4.43E+5/plant
Class V License Amendment Fee (PWRs and BWRs)	= \$2.6E+4/plant
I	= \$5.2E+5/plant

(a) See Attachment 2.

TABLE 3. (contd)

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

   NI = 14($5.2E+5/plant) = $7.3E+6

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

   - Routine operation and maintenance (assumed):

     Operation = 2 man-wk/py

     Maintenance = 2 man-wk/py

     Subtotal = 4 man-wk/py

   - Periodic repair or replacement of heaters:

     Heater life assumed at 5 yr. It will take about 60 man-wk/plant to retrain staff, conduct tests, fabricate and install replacement heater elements (equipment costs estimated directly in next step).

   - Annual labor per plant:

     Total = 4 + 12 = 16 man-wk/py

   This estimate applies to both PWRs and BWRs.

10. Per-Plant Industry Costs for SIR Operation and Maintenance (I_o):

   - Labor cost = (16 man-wk/py)($2270/man-wk) = $3.63E+4/py

   - Material cost = (Heater cost including 50% contingency factor)/(5 yr) = ($1.4E+4/plant)/(5 yr) = $2800/py

   - Power cost for operating 300 kW heaters with 80% availability at a unit cost of 5¢/kWh = (365 days/py)(24 hr/day)(0.80) (300 kW)($0.05/kWh) = $1.1E+5/py

   I_o = $36,300 + $2,800 + $110,000 = $1.44E+5/py

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

   $N\bar{T}I_o$ = [9(9.2 yr) + 5(8.4 yr)]($1.44E+5/py) = $1.80E+7

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$2.5E+7	$3.5E+7	$1.6E+7

TABLE 3. (contd)

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC staff labor = 16 man-wk

Contractor support (cost estimated directly in next step).

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

Labor = (16 man-wk)($2270/man-wk)	=	$3.6E+4
Contractor Support	=	$5.0E+5
C_D	=	$5.36E+5

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

NRC labor is assumed to be about 15% of industry labor (195 man-wk/plant) or 30 man-wk/plant. This estimate applies to both PWRs and BWRs.

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

C = (30 man-wk/plant)($2270/man-wk) = $6.81E+4/plant

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

NC = 14($6.81E+4/plant) = $9.53E+5

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

Assumed to be 1 man-wk/py (both PWRs and BWRs).

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

C_O = (1 man-wk/py)($2270/man-wk) = $2270/py

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

$N\bar{T}C_O$ = [9(9.2 yr) + 5(8.4 yr)]($2270/py) = $2.83E+5

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$1.8E+6	$2.3E+6	$1.2E+6

ATTACHMENT 2

Industry Resources for SIR Implementation

For reactor vessel supports that may require improvement in reliability, it is asssumed that electric heaters could be installed at operating plants and operated at 500°F to avoid susceptibility to the NDTT shift.

The equipment required and costs are presented below.

Equipment	Cost (per plant)
Plate-mounted strip heaters clamped to 4 support columns	\$9,000
Mounting hardware, materials, wiring	4,000
Power controller 310 kW, 480V	8,000
Switchgear/transformers	2,000
Metal-sheathed heating cables	12,000
Equipment cost, rounded	\$35,000
Contingency at 50%	17,000
Total	\$52,000

The labor required is as follows:

Task	Man-wk/plant
Develop design changes, mechanical and electrical engineering and drafting.	16
Analyze, document, and process through NRC approval.	66
Procure equipment specifications, purchase, and inspect.	3
Plan implementation effort.	4
Train staff.	2
Change plant procedures.	3
Fabricate heater assemblies.	18
Complete Installation:	
Mockup	4
Installation.	8
Conduct final tests.	6
Labor, subtotal	130
Contingency at 50%	65
Total	195

REFERENCES

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: A-18, Pipe Rupture Design Criteria

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This program is intended to fulfill short-term goals related to the development of consistent criteria for immediate application in licensing processes. The task remaining to be resolved addresses guidelines for limiting break exclusion regions, developing criteria for using guard pipes and design adequacy of break exclusion areas. At this point, the resolution is to develop criteria to limit the extent of these regions.

AFFECTED PLANTS	BWR:	Operating = 0	Planned = 12
	PWR:	Operating = 0	Planned = 26

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 0

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	-250
Total of Above =	-250
Accident Avoidance =	0

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:

SIR Implementation =	2
SIR Operation/Maintenance =	-0.14
Total of Above =	1.9
Accident Avoidance =	0

NRC COSTS:

SIR Development =	0
SIR Implementation Support =	0.26
SIR Operation/Maintenance Review =	0.22
Total of Above =	0.48

PIPE RUPTURE DESIGN CRITERIA

ISSUE A-18

1.0 ISSUE DESCRIPTION

A major objective of Issue A-18 is to develop consistent criteria for application in licensing processes. Additional research programs to implement licensing positions are to be conducted under separate issues. The problems specific to Issue A-18 are as follows (NUREG-0471, NRC 1978):

A. "Current design criteria for the postulation of pipe breaks and protection therefrom have been developed over a period of time and lack consistency when applied inside and outside containment. Regulatory Guide 1.46, issued in 1973, which addresses pipe breaks inside containment, is based on the concept of a limited number of design basis breaks. Section 3.6 of the Standard Review Plan, issued in 1975, which addresses pipe breaks outside containment, combines limited design basis breaks for mechanistic protection and unlimited breaks for nonmechanistic protection. Current staff efforts toward documentation of the rationale and engineering justification for the existing pipe break criteria should continue. These efforts will assist in focusing on areas requiring first attention and will provide a valuable document for both public and staff use as bases for testimony before the ACRS and hearing boards." Work in this area is complete.

B. "An evaluation of the pipe break exclusion concept in the containment penetration area of both PWR and BWR plants is required. The need to specify the extent of break exclusion regions, criteria for the use of guard pipes, and adequacy of design requirements for piping systems in break exclusion regions are topics for which improved guidance will be developed.

C. "The development of postulated pipe rupture criteria and the trend towards more conservative seismic criteria have placed increased emphasis on piping system design to withstand these dynamic events. However, these have also resulted in systems which are significantly more rigid. These more rigidly designed systems in the newer plants, which are not yet operating, have resulted in calculated stresses for normal operation which, although still within code limits, are significantly higher than in earlier plants. In addition, dynamic event devices, such as snubbers and pipe-whip restraints, which have been added in increased numbers, have the potential for deleterious interaction with the piping system during its normal operation. A balance in piping system design for both normal and abnormal situations should be achieved to assure that consideration is given to the effects that abnormal situation design criteria have on normal operation." The effects that abnormal loading scenario design criteria have on normal operation have been examined (Landers et al. 1981). Determining licensing position and consequences of implementing results of this task were not a portion of the issue. Issue B-6 more directly addresses the consequences of combining unusual dynamic events and normal plant operating conditions on plant safety and addresses the option of limiting numbers of dynamic event devices.

PROPOSED SAFETY ISSUE RESOLUTION

The criteria used for designing and constructing containment penetrations were to be evaluated in this issue. Guidelines for limiting the extent of break exclusion areas, criteria for the use of guard pipes and adequacy of design requirements for piping systems in break exclusion areas were of concern. The consequences of implementing the resultant guidelines may differ for various plant types and piping systems. It is assumed that the safety issue resolution (SIR) will, in general, limit the number of break exclusion areas. It is further assumed that this limitation will affect only 60 percent of all planned PWRs and BWRs.

2.0 SAFETY RISK AND DOSE

PUBLIC RISK REDUCTION

The reduction in public risk was determined to be negligible. Limiting the extent of break exclusion areas does not increase or decrease the probability of a pipe rupture. Thus, the Public Risk Reduction Work Sheet has been omitted.

OCCUPATIONAL DOSE

When a line is excluded from a break exclusion area, associated welds would no longer require 100 percent volumetric inspection every 10 years. Instead, inservice inspections of these welds would be scheduled once during the lifetime of the plant; i.e., 25 percent of welds are inspected every 10 years. Radiation exposure is somewhat reduced because of this resolution. Table 1 includes results of this analysis.

TABLE 1. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Pipe Rupture Design Criteria (A-18)

2. Affected Plants (N):

   It is estimated that only 60% of all planned plants would require redesign to meet limitation requirements on break exclusion areas:

		N
Planned PWRs:	(0.60)(43) = 25.8 ~	26
Planned BWRs:	(0.60)(20) =	12
	Total =	38

TABLE 1. (contd)

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

	$\bar{T}$(yr)
26 Planned PWRs:	30.0
12 Planned BWRs:	30.0

4-5. Steps Related to Occupational Dose Reduction Due to Accident Avoidance:

There is no change in core-melt frequency; thus, the occupational dose reduction due to accident avoidance is zero.

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

Implementation occurs during stages of plant design. Any alterations made in break exclusion areas would occur before plant operation and start-up. Thus, no radiation exposure would be accrued. (D = 0).

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

In recent years, the tendency toward overextending the break exclusion area has been reversed. It is estimated that, of the 40 high-energy penetrations/plant (an average maximum assumed from information provided in section 3.6 of several BWR and PWR FSARs), very few excessive break exclusion areas exist. With this in mind, it is assumed that 12 weld design locations per plant could be transferred from the break exclusion area, thus implying that they no longer require the 100% volumetric inspection every ten years. Actual labor time for weld inspection is estimated at 0.5 man-hr/weld. However, due to restrictions imposed by guard pipe assemblies, inspection parts, etc. in break exclusion areas, it is anticipated that the actual inspection time is four times greater, i.e., 2.0 man-hr/weld.

The labor saved by changing the inservice inspection from that required in a break exclusion area to that required outside the break exclusion area is as follows (the requirement is that 25% of welds be inspected every 10 years, or basically one inspection during the lifetime of the plant--in this case, a 30-year plant life is assumed):

[(2.0 man-hr/weld) (1 inspection period/10 py) - (0.5 man-hr/weld) (1 inspection period/30 py)] (12 welds) = 2.2 man-hr/py

This difference of -2.2 man-hr/py, due to implementation and maintenance of the SIR, assumes that the time required for equipment setup, providing access to the general area, etc. is roughly equivalent for both inspection procedures. This estimate applies to both PWRs and BWRs.

TABLE 1. (contd)

10. Per-Plant Occupational Dose Increase for SIR Operation and Maintenance (D_0):

A 0.10 rem/hr average dose rate is assumed for inservice inspections.(Duke Power Co. 1982)

D_0 = (-2.2 man-hr/py) (0.10 rem/hr) = -0.22 man-rem/py (Negative sign indicates reduction.)

11. Total Occupational Dose Increase for SIR Operation and Maintenance ($N\bar{T}D_0$):

$N\bar{T}D_0$ = (38) (30 yr) (-0.22 man-rem/py) = -251 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
-250	-84	-750

(Negative signs indicate reductions.)

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost calculations are included in this section. Best estimates are used for labor time required in the analysis of pipe rupture and time required for follow-up studies. Table 2 includes the results of this analysis.

TABLE 2. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Pipe Rupture Design Criteria (A-18)

2. Affected Plants (N):

	N
Planned PWRs:	26
Planned BWRs:	12
Total	38

TABLE 2. (contd)

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

	$\bar{T}$(yr)
26 Planned PWRs:	30.0
12 Planned BWRs:	30.0

INDUSTRY COSTS (Steps 4 through 12)

4-5. Steps Related to Industry Cost Savings Due to Accident Avoidance:

There is no change in core-melt frequency; thus, no accident-avoidance cost savings result.

6. Per-Plant Industry Resources for SIR Implementation:

Labor includes implementation of criteria for defining pipe break and crack locations and configurations; criteria dealing with special features, such as augmented inservice inspections or use of postulated event devices; acceptability of analysis results, including jet-thrust and impingement forcing functions, pipe-whip dynamic effects and design adequacy of systems to assure that function is not impaired as a result of pipe-whip or jet impingement loadings.(NUREG-0800, NRC 1981) It is assumed that labor includes the time required to analyze lines now located outside the break exclusion regions and that analysis procedures, computer codes, applicable transient data, etc. are readily available. It also assumes that only 50% of the 12 welds under investigation will need analysis (i.e., those excluded either already fall into an analyzed line or do not fall into a high-energy/high-stress area which requires analysis).

Labor: (4 man-wk/line segment)(6 line segments/plant) = 24 man-wk/plant

(Note: 1 affected weld/line segment is assumed.)

(Same for PWRs and BWRs.)

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

I = (24 man-wk/plant)($2270/man-wk) = $5.45E+4/plant

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

NI = (38)($5.45E+4/plant) = $2.07E+6

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

Due to fewer inservice inspection periods when weld design locations are shifted from a break exclusion area, labor is assumed to decrease by 2.2 man-hr/py (see Occupational Risk Reduction Work Sheet, Step 9). This estimate applies to both PWRs and BWRs.

TABLE 2. (contd)

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

I_0 = (-2.2 man-hr/py)(1 man-wk/40 man-hr)($2270/man-wk) = -$1.25E+2/py
(Negative sign indicates cost savings.)

11. Total Industry Cost for SIR Operation and Maintenance (NTI_0):

$\bar{NTI_0}$ = (38)(30.0 yr)(-$1.25E+2/py) = -$1.43E+5

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$1.9E+6	$3.0E+6	$8.9E+5

NRC Costs (Steps 13 through 21)

13-14. Steps Related to NRC Cost for SIR Development:

The NRC will provide criteria to limit the extent of break exclusion regions for plant types and piping systems. Independent plant reviews with respect to new SRP regulations will be conducted. The generic issue resolution has been completed. Thus, C_D = 0.

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

It is assumed that implementation support will require 3 man-wk/plant due to individual plant equipment and design. This estimate applies to both PWRs and BWRs.

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

C = (3 man-wk/plant)($2270/man-wk) = $5.81E+3/plant

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

NC = (38)($6810/plant) = $2.59E+5

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

No change in review time of piping systems is anticipated. However, a review of the consequences of imposing limitations on break exclusion areas would be in order.

Actual review = 2 man-wk/plant

Review analysis and report = 20 man-wk/38 plants = 0.53 man-wk/plant

TABLE 2. (contd)

18. Per-Plant NRC Labor for Review of SIR Operation and Maintenance: (contd)

(2 man-wk/plant + 0.53 man-wk/plant)(1 plant/30 py) = 0.084 man-wk/py
(Same for both PWRs and BWRs)

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

C_O = (0.084 man-wk/py)($2270/man-wk) = $1.91E+2/py

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

$N\bar{T}C_O$ = (38)(30 yr)($191/py) = $2.2E+5

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$4.8E+5	$6.5E+5	$3.1E+5

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: A-29, Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This issue is concerned with considering plant design alternatives to reduce the vulnerability of reactors to sabotage. The proposed design alternative is to add an independent, hardened decay heat removal system as a redundant train of the emergency feedwater system to all new PWRs and BWRs. This system would only be activated during a sabotage attack or other extreme emergency.

AFFECTED PLANTS	BWR:	Operating = 0	Planned = 20
	PWR:	Operating = 0	Planned = 43

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	1.5E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	90

COST RESULTS ($$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	630
SIR Operation/Maintenance =	11
Total of Above =	640
Accident Avoidance =	7.5
NRC COSTS:	
SIR Development =	1.0
SIR Implementation Support =	0.57
SIR Operation/Maintenance Review =	4.3
Total of Above =	5.9

NUCLEAR POWER PLANT DESIGN FOR THE REDUCTION OF VULNERABILITY TO INDUSTRIAL SABOTAGE

ISSUE A-29

1.0 SAFETY ISSUE DESCRIPTION

Safety issue A-29 deals with the consideration of alternatives to the basic design of nuclear power plants with the emphasis primarily on reduction of the vulnerability of reactors to industrial sabotage. Extensive efforts and resources are expended in designing nuclear power plants to minimize the risk to the public health and safety from equipment or system malfunction or failure. However, reduction of the vulnerability of reactors to industrial sabotage is currently treated as a plant physical security function and not as a plant design requirement. Although present reactor designs do provide a great deal of inherent protection against industrial sabotage, extensive physical security measures are still required to provide an acceptable level of protection. An alternate approach would be to more fully consider reactor vulnerabilities to sabotage along with economy, operability, reliability, maintainability, and safety during the preliminary design phase. Since emphasis is being placed on standardizing plants, it is especially important to consider measures which could reduce the vulnerability of reactors to sabotage. Design features to enhance physical protection must be consistent with present and future system safety requirements (NRC 1978).

The proposed resolution for this safety issue is the addition of an independent, hardened decay heat removal system as a redundant train of the emergency feedwater system which is used only in a sabotage incident or other extreme emergency as determined by plant operators. This proposed design change is based on considerations and recommendations in a Sandia report for the NRC titled <u>Nuclear Power Plant Design Concepts for Sabotage Protection</u> (Ericson and Varnado 1981). Several other design changes were considered in the report. The independent, hardened decay heat removal system was chosen as the basis for estimating the risk reduction dose and cost associated with resolution of Issue A-29.

The independent, hardened decay heat removal system would have the following general features:

- location in hardened buildings or structures complete with power, water, and controls
- manual activation from local control panel
- independence from the remainder of the plant when operating
- design for removal of decay heat from an LWR in hot shutdown for a specified period of time without operator intervention

- design to continue decay heat removal under manual control beyond automatic operation period
- dedication for use only in extreme emergency
- provision for isolation of fluid lines as required
- noninterference with operation of other engineered safety features.

The design chosen for development and for estimating cost uses electric power for its operation. Power is supplied by a diesel generator located, with the remainder of the equipment required for the system, in a hardened building. Heat loads associated with the diesel generator and other mechanical equipment are transferred to the atmosphere by an air-cooled heat exchanger. A pipe tunnel connects the hardened decay heat removal building with the containment. The system is a single, 100-percent system without redundancy or single-failure capability. The design period of unattended operation is 10 hours (Ericson and Varnado 1981). The independent, hardened decay heat removal system is assumed to be added only to new PWRs and BWRs based on information in the Sandia report (Ericson and Varnado 1981).

This issue affects all planned PWRs and BWRs. The Oconee 3 (B&W) PWR is chosen to represent all planned PWRs. The results from the PWR analysis are scaled for the Grand Gulf 1 (GE) BWR, chosen to represent all planned BWRs.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose are estimated in this section and summarized in Tables 1 and 2, respectively.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage (A-29)

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

   All planned PWRs and BWRs

	N	$\bar{T}$ (yr)
Planned	43	30
Planned	20	30
	63	30

TABLE 1. (contd)

3. Plants Selected for Analysis:

Oconee 3 - representative PWR

Grand Gulf 1 - representative BWR

(The analysis is conducted for Oconee 3, and the results are scaled for Grand Gulf 1)[a]

4. Parameters Affected by SIR:

Oconee:

Symbol	Description
CONST1	Failure of the emergency feedwater system due primarily to hardware failure of the turbine pump train and both of the electric pump trains, or blockage of flow to both steam generators
CONST2	Failure of the emergency feedwater system due to failure of both electric pump trains or blockage of flow to both steam generators.

5. Base-Case Values for Affected Parameters:

CONST1 = 2.1E-4 (original Oconee value)

CONST2 = 6.3E-4 (original Oconee value)

6. Affected Accident Sequences and Base-Case Frequencies:

Sequence	Base-Case Frequency (1/py)
$T_2MLU - \gamma$ (PWR-3)	4.7E-7
$T_2MLU - \beta$ (PWR-5)	6.9E-9
$T_2MLU - \epsilon$ (PWR-7)	4.7E-7
$T_1MLU - \gamma$ (PWR-3)	9.5E-7
$T_1MLU - \beta$ (PWR-5)	1.4E-8
$T_1MLU - \epsilon$ (PWR-7)	9.5E-7

(a) See Attachment 1.

TABLE 1. (contd)

7. Affected Release Categories and Base-Case Frequencies:

PWR-3 = 1.4E-6/py

PWR-5 = 2.1E-8/py

PWR-7 = 1.4E-6/py

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

$\bar{F}_{PWR}$ = 2.9E-6/py $\bar{F}_{BWR}$ = 1.3E-6/py[a]

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

W_{PWR} = 7.7 man-rem/py W_{BWR} = 9.2 man-rem/py[a]

10. Adjusted-Case Values for Affected Parameters:

The values of CONST1 and CONST2 are redefined due to the addition of the independent, hardened decay heat removal system as a redundant train of the emergency feedwater system. Thus,

CONST1 = 3.4E-6

CONST2 = 1.0E-5

The derivation of the adjusted-case values for redefined CONST1 and CONST2 is given in the first part of Attachment 1.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Sequence		Adjusted-Case Frequency (1/py)
T_2MLU -	γ (PWR-3)	7.7E-9
	β (PWR-5)	1.1E-10
	ϵ (PWR-7)	7.7E-9
T_1MLU -	γ (PWR-3)	1.5E-8
	β (PWR-5)	2.2E-10
	ϵ (PWR-7)	1.5E-8

(a) See Attachment 1.

TABLE 1. (contd)

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-3 = 2.3E-8/py

PWR-5 = 3.3E-10/py

PWR-7 = 2.3E-8/py

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

$F^*_{PWR} = 4.6E-8/py$

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

$W^*_{PWR} = 1.2E-1$ man-rem/py

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

$(\Delta\bar{F})_{PWR} = 2.9E-6/py$ $(\Delta\bar{F})_{BWR} = 1.3E-6/py$[a]

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

$(\Delta W)_{PWR} = 7.6$ man-rem/py

$(\Delta W)_{BWR} = 9.1$ man-rem/py[a]

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.5E+4	4.6E+5	0

(a) See Attachment 1.

ATTACHMENT 1

The variables CONST1 and CONST2 involve the emergency feedwater system in the Oconee reactor. The addition of the independent, hardened decay heat removal system redefines the values CONST1 and CONST2 for the adjusted case. A flow diagram showing the assumed interaction of independent, hardened decay heat removal system equipment as a redundant train of the emergency feedwater system at Oconee is shown in Figure 1. Thus, the failure of the independent decay heat removal system to operate properly is the sum of the failure probabilities of its components.

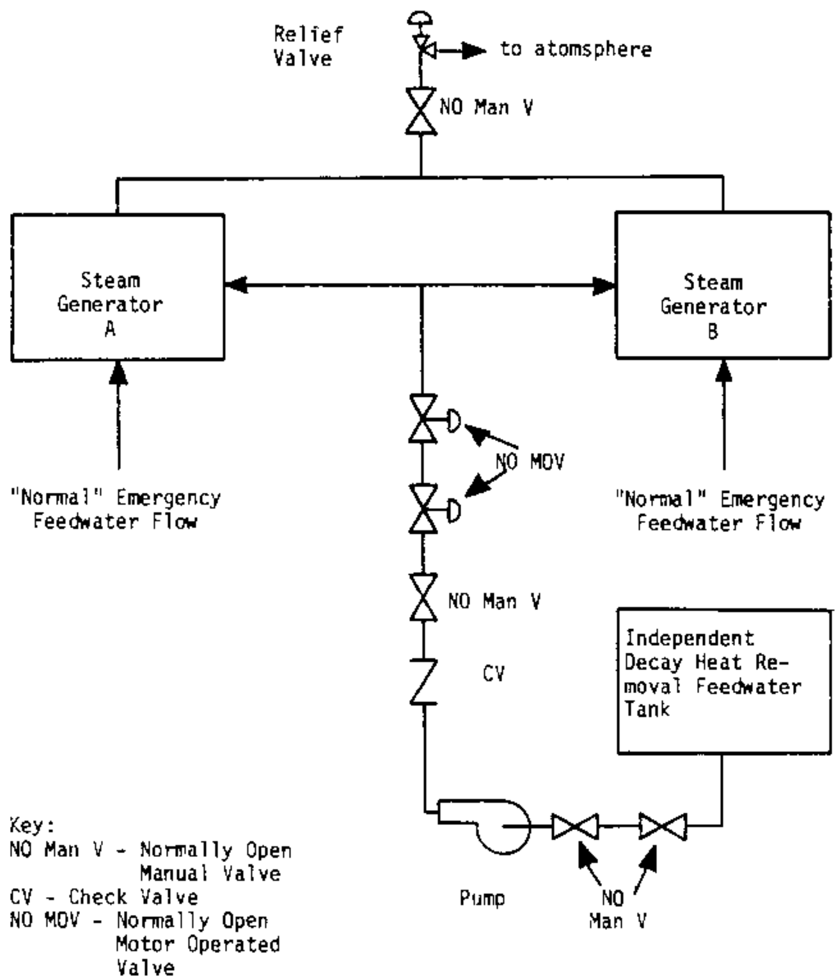

FIGURE 1. Proposed Flow Diagram of Independent Decay Heat Removal Feedwater Flow

ATTACHMENT 1 (contd)

		Failure Probability
3 NO Man Vs	= 3(0.0002)	= 0.0006
2 NO MOVs	= 2(0.0032)	= 0.0064
1 CV	= 0.0001	= 0.0001
1 Pump	= 0.0093	= 0.0093
1 Tank (rupture)	= ~0	= ~0
		0.0164

The failure probabilities listed above are taken from Table A.4 in PNL-4297 (Andrews et al. 1982). The failure probability of the pump mentioned above is assumed to have the same failure probability as an electric pump in the original emergency feedwater system.

The adjusted-case values of the redefined CONST1 and CONST2 are:

$$\text{CONST1*} \begin{cases} = (\text{CONST1})\ (0.0164) \\ = (2.1\text{E-}4)\ (0.0164) \\ = 3.4\text{E-}6 \end{cases}$$

$$\text{CONST2*} \begin{cases} = (\text{CONST2})\ (0.0164) \\ = (6.3\text{E-}4)\ (0.0164) \\ = 1.0\text{E-}5 \end{cases}$$

The RSSMAP studies for Oconee 3 and Grand Gulf 1 give total core-melt frequencies ($\bar{F}_0$) of 8.2E-5/py and 3.7E-5/py, respectively, for these plants. Using the original release category frequencies and the public dose factors (Appendix D of PNL-4297), one obtains total public risks (W_0) of 207 man-rem/py and 250 man-rem/py, respectively, for Oconee and Grand Gulf. For the purposes of scaling the base-case, affected core-melt frequency ($\bar{F}$) and public risk (W), and the reductions in the core-melt frequency ($\Delta\bar{F}$) and public risk (ΔW) from Oconee to Grand Gulf, the following are assumed:

$$\left.\begin{array}{l} \bar{F}_{BWR}/\bar{F}_{PWR} \\ (\Delta\bar{F})_{BWR}/(\Delta\bar{F})_{PWR} \end{array}\right\} = (\bar{F}_0)_{BWR}/(\bar{F}_0)_{PWR}$$

$$\left.\begin{array}{l} W_{BWR}/W_{PWR} \\ (\Delta W)_{BWR}/(\Delta W)_{PWR} \end{array}\right\} = (W_0)_{BWR}/(W_0)_{PWR}$$

Using the original values of $\bar{F}_0$ and W_0 for Oconee and Grand Gulf, the scaling equations become:

$$\bar{F}_{BWR} = 0.45\ \bar{F}_{PWR}$$
$$(\Delta\bar{F})_{BWR} = 0.45\ (\Delta\bar{F})_{PWR}$$
$$W_{BWR} = 1.2\ W_{PWR}$$
$$(\Delta W)_{BWR} = 1.2\ (\Delta W)_{PWR}$$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage (A-29)

2. Affected Plants (N):

All new PWRs and BWRs

Planned	43
Planned	20
	63

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

Since this issue resolution only applies to new (planned) reactors, the average remaining life is 30 years.

4. Per-Plant Occupational Dose Reduction Due to Accident Avoidance, $\Delta(FD_R)$:

$$PWR = (19{,}900 \text{ man-rem})(2.9E\text{-}6/py) = 5.8E\text{-}2 \ \frac{\text{man-rem}}{py}$$

$$BWR = (19{,}900 \text{ man-rem})(1.3E\text{-}6/py) = 2.6E\text{-}2 \ \frac{\text{man-rem}}{py}$$

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
9.0E+1	5.4E+2	0

6-11. Steps Related to Occupational Dose Increase for SIR Implementation and Operation/Maintenance:

These steps are omitted since the issue resolution assumes implementation during construction; thus, no radiation zone work is involved. Also operation/maintenance involves no radiation zone work because the hardened decay heat removal system will be in an independent building and will only be used after a sabotage attack. Thus, during normal operation the system is not considered to be located in a radiation zone, and $D = D_0 = 0$.

TABLE 2. (contd)

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
0	0	0

3.0 SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. Results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage (A-29)

2. Affected Plants (N):

   All new PWRs and BWRs

Planned	43
Planned	20
	63

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

   Since this issue resolution only applies to new (planned) reactors, the average remaining life is 30 years.

Industry Costs (Steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, $\Delta(\bar{FA})$:

   PWR = ($1.65E+9)(2.9E-6/py) = 4.8E+3/py

   BWR = ($1.65E+9)(1.3E-6/py) = $2.1E+3/py

TABLE 3. (contd)

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

Best Estimate	Upper Bound	Lower Bound
\$7.5E+6	\$4.5E+7	0

6. Per-Plant Industry Resources for SIR Implementation:

In Appendix G of the Sandia report (Ericson and Varnado 1981), details on the resources needed to add an independent, hardened decay heat removal system to a PWR are listed. The resources include the labor, materials and/or equipment for the substructure, superstructure, process equipment, and building services to construct an independent building for the independent, hardened decay heat removal system. The resources are assumed to be very similar for addition to PWRs and BWRs.

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

The industry cost of \$1.0E+7/plant is assumed based on cost estimation in the Sandia report (Ericson and Varnado 1981) adjusted to include engineering costs. This industry cost is assumed to be the same for a PWR or a BWR.

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

(\$1.0E+7/plant)(63 plants) = \$6.3E+8

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

It is assumed that 2.5 man-wk/py are necessary to check the diesel power source each month and the pumps every three months as part of routine maintenance.

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_o):

(2.5 man-wks/py)(\$2270/man-wk) = \$5.7E+3/py

This cost is the same for PWRs and BWRs.

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

(\$5.7E+3/py)(63 plants)(30 yr) = \$1.1E+7

TABLE 3. (contd)

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$6.4E+8	\$9.6E+8	\$3.2E+8

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Estimates included directly in next step.

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

It is assumed that NRC will spend \$1.0E+6 to develop the issue resolution.

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

NRC is assumed to expend 4 man-wk/plant to check the design of each plant's independent, hardened decay heat removal system.

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

(4 man-wk/plant)(\$2270/man-wk) = 9.1E+3/plant

This cost is the same for PWRs and BWRs.

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

(\$9.1E+3/plant)(63 plants) = \$5.7E+5

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

NRC is assumed to expend 1 man-wk/py to review industry surveillance results for the independent, hardened decay heat removal systems.

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

(1 man-wk/py)(\$2270/man-wk) = \$2.3+3/py

This cost is the same for PWRs and BWRs.

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

(\$2.3E+3/py)(63 plants)(30 yr) = \$4.3E+6

TABLE 3. (contd)

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$5.9E+6	\$8.1E+6	\$3.7E+6

REFERENCES

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

Ericson, D. M. Jr., and G. B. Varnado. 1981. Nuclear Power Plant Design Concepts for Sabotage Protection. NUREG/CR-1345, SAND 80-0477/2, Sandia National Laboratories, Albuquerque, New Mexico.

Flood, M. 1976. "Nuclear Sabotage." Bulletin of the Atomic Scientists. 32(8):29-36.

U.S. NRC. 1978. Task Action Plans for Generic Activities (Category A). NUREG-0371, U.S. Nuclear Regulatory Commission, Washington D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: C-11, Assessment of Failure and Reliability of Pumps and Valves

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

"Valve malfunctions can cause forced outages of operating plants. It is noted that about 10% of all outage time can be attributed to the malfunction of the critical pumps and valves within the plant" (NUREG-0471, NRC 1978). This issue will address active pump and valve operability and reliability, with the assumed intent being to replace those valves and pumps which have a history of failure due to design and fabrication error.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	6.6E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	9800
SIR Operation/Maintenance =	-2400
Total of Above =	7400
Accident-Avoidance =	510

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	730
SIR Operation/Maintenance =	-750
Total of Above =	-22
Accident-Avoidance =	44
NRC COSTS:	
SIR Development =	2.3
SIR Implementation Support =	1.1
SIR Operation/Maintenance Review =	0.14
Total of Above =	3.6

ASSESSMENT OF FAILURE AND RELIABILITY OF PUMPS AND VALVES

ISSUE C-11

1.0 SAFETY ISSUE DESCRIPTION

The operating experience of nuclear power plants indicates that a number of valves, valve operators and pumps fail to operate as specified in the technical specifications either under testing conditions or when they are called upon to perform. The operating experience is documented by the Office of Management Information and Program Control (MIPC) publications in a monthly report of Licensee Event Reports (LERs), sorted by components which include pumps, valves, and valve operators. Most of these occurrences relate to valve leakage, valve actuation, and safety/relief valve operation outside their operational bounds. The main steam isolation, safety and solenoid valves caused the most frequent abnormal occurrences in safety-related systems. Valve malfunctions can cause forced outages of operating plants. It is noted that about 10% of all outage time can be attributed to the malfunction of the critical pumps and valves within the plant. Of primary interest are outages caused by the main steam isolation and safety/relief valves.

The principal activity under this Safety Issue Resolution (SIR) task will be the evaluation of active pumps and valves with respect to their operability and reliability under accident loading, i.e., loss of coolant accident and safe shutdown earthquake (NUREG-0471, NRC 1978), and implementation of corrective action programs specifically directed toward improved design and fabrication of active pumps and valves.

2.0 SAFETY ISSUE RISK AND DOSE

PUBLIC RISK REDUCTION

It is assumed that resolution of this issue will serve to identify active pumps and valves that need redesign and replacement. Results from other issues (e.g., II.E.6, "In-Situ Testing of Valves," and II.D.2, "Research on Relief and Safety Valve Test Requirements") will supplement study done in the equipment identification and qualification process of Issue C-11. The reduction in public risk will result from a decreased probability of valve and pump failure. All issues related to valves should be considered together for prioritization to avoid double-counting of risk reductions. Table 1 includes the results for this analysis.

OCCUPATIONAL DOSE

Issue resolution requires additional radiation exposure to personnel during replacement of designated valves and pumps in operating plants. It is possible that those designated for replacement exceed the number that would actually fail and require replacement over the lifetime of the plant. A decrease in radiation exposure would result, however, from elimination of clean-up associated with valve and pump failures.

Replacement of failed parts and associated clean-up would be decreased in planned plants due to initial installation of replacement parts. Table 2 includes the results for this analysis.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Assessment of Failure and Reliability of Pumps and Valves (C-11)

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

All operating and planned PWRs and BWRs will be affected by resolution of this issue. It is assumed that five years will be required to completely resolve the issue and redesign, fabricate and install design-improved valves and pumps. The projected number of operating and planned plants was based on available start dates (Andrews et al. 1983). The remaining number of operating years was determined accordingly. As licensing schedules are altered over the next five years, these projections should be modified.

		N	$\bar{T}$(yr)
PWR:	planned (plants commencing operation after 1987)	7	30.0
	operating (plants operational by 1988)	83	24.8
	All PWRs	90	25.2
BWR:	planned (as above)	4	30.0
	operating (as above)	40	23.0
	All BWRs	44	23.6

3. Plants Selected for Analysis:

Oconee 3: Representative PWR

Grand Gulf 1: Representative BWR

4. Parameters Affected by SIR:

"The pumps and valves identified as active, whose operation is relied upon to assure safe plant shutdown or mitigate the consequences of an accident, are listed in Tables 3.9(B)-22 and 3.9(B)-23, respectively (Seabrook FSAR, 1981). These pumps and valves, classified as seismic Category I, are designed to perform their intended functions during

4. Parameters Affected by SIR (contd)

postulated plant conditions. Their operability is assured by adherence to the design limits and supplemental stress requirements specified in NRC Regulatory Guide 1.48" (Seabrook FSAR 1981).

Where possible, the decision regarding valve or pump status (i.e., active or passive) was based on FSAR piping and instrument diagrams and available listings in FSAR Section 3.9.2. Consider Oconee cut set elements B, C, D, and E. Failure of a pump suction valve in train A or B of the low pressure/containment spray injection system (LP/CSIS) occurs if 1) a normally-open (NO) motor-operated valve (MOV) fails, or 2) a check valve (CV) fails. The NO MOV is assumed to be a passive valve, not to have a hardware failure contributory mode. The CV is assumed to be an active valve and to have a hardware failure probability of .0001. Thus, both elements B and C are assumed to be affected parameters.

In a similar analysis, elements D and E of Oconee are affected if failure of a pump discharge valve in train A or B of the LP/CSIS occurs. This assumes failure of either of 2 CVs, either of 2 NO MOVs, or a normally-closed (NC) MOV. The CVs and NC MOV are assumed to be active and the NO MOVs passive. In addition, hardware failure of an active pump is included.

The following list includes all affected elements of the dominant minimal cut sets for Oconee and Grand Gulf dominant accident sequences:

Oconee 3: B, C, D, E, CONST1, CONST2, A1, B1, C1, Q, F1, G1, D•E, W•X, B•W, C•X, D•X, E•W, B•D, E•C.

Grand Gulf 1: H, P, R, L, LA2, LB1, LB2, LC, PA27, PB27, VGA1, VGA2, VGB1, VGB2, SA, SB, SSA, SSB, SSC, V1, V2, SCVA, SCVB.

5. Base-Case Values for Affected Parameters:

Base-case values remain unchanged from original values; refer to the Guidelines, Tables A.4 and B.4 (Andrews et al. 1983).

6. Affected Accident Sequences and Base-Case Frequencies:

Oconee:

$$T_2MLU - \begin{cases} \gamma(PWR\text{-}3) = 5.5E\text{-}7/py \\ \beta(PWR\text{-}5) = 8.1E\text{-}9/py \\ \varepsilon(PWR\text{-}7) = 5.5E\text{-}7/py \end{cases}$$

$$T_1MLU - \begin{cases} \gamma(PWR\text{-}3) = 1.0E\text{-}6/py \\ \beta(PWR\text{-}5) = 1.5E\text{-}8/py \\ \varepsilon(PWR\text{-}7) = 1.0E\text{-}6/py \end{cases}$$

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

$$T_2MQH - \left\{\begin{array}{lcl} \gamma(\text{PWR-3}) & = & 5.0\text{E-6/py} \\ \beta(\text{PWR-5}) & = & 7.3\text{E-8/py} \\ \varepsilon(\text{PWR-7}) & = & 5.0\text{E-6/py} \end{array}\right.$$

$$S_3H - \left\{\begin{array}{lcl} \gamma(\text{PWR-3}) & = & 6.0\text{E-7/py} \\ \beta(\text{PWR-5}) & = & 8.8\text{E-9/py} \\ \varepsilon(\text{PWR-7}) & = & 6.0\text{E-7/py} \end{array}\right.$$

$$S_1D - \left\{\begin{array}{lcl} \alpha(\text{PWR-1}) & = & 5.3\text{E-8/py} \\ \gamma(\text{PWR-3}) & = & 1.1\text{E-6/py} \\ \beta(\text{PWR-5}) & = & 3.9\text{E-8/py} \\ \varepsilon(\text{PWR-7}) & = & 4.2\text{E-6/py} \end{array}\right.$$

$$T_2MQFH - \left\{\begin{array}{lcl} \gamma(\text{PWR-2}) & = & 2.4\text{E-6/py} \\ \beta(\text{PWR-4}) & = & 3.4\text{E-8/py} \\ \varepsilon(\text{PWR-6}) & = & 2.4\text{E-6/py} \end{array}\right.$$

$$S_3FH - \left\{\begin{array}{lcl} \gamma(\text{PWR-2}) & = & 9.0\text{E-8/py} \\ \beta(\text{PWR-4}) & = & 1.3\text{E-9/py} \\ \varepsilon(\text{PWR-6}) & = & 9.0\text{E-8/py} \end{array}\right.$$

$$S_2FH - \left\{\begin{array}{lcl} \alpha(\text{PWR-1}) & = & 5.7\text{E-10/py} \\ \beta(\text{PWR-4}) & = & 4.2\text{E-10/py} \\ \varepsilon(\text{PWR-6}) & = & 4.6\text{E-8/py} \end{array}\right.$$

$$T_2MLUO - \left\{\begin{array}{lcl} \gamma(\text{PWR-3}) & = & 4.0\text{E-6/py} \\ \beta(\text{PWR-5}) & = & 5.8\text{E-8/py} \\ \varepsilon(\text{PWR-7}) & = & 4.0\text{E-6/py} \end{array}\right.$$

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

$$S_2D - \begin{cases} \alpha(\text{PWR-1}) = 6.9\text{E-9/py} \\ \gamma(\text{PWR-3}) = 1.4\text{E-7/py} \\ \beta(\text{PWR-5}) = 5.1\text{E-9/py} \\ \varepsilon(\text{PWR-7}) = 5.5\text{E-7/py} \end{cases}$$

$$S_3D - \begin{cases} \gamma(\text{PWR-3}) = 6.5\text{E-7/py} \\ \beta(\text{PWR-5}) = 9.5\text{E-9/py} \\ \varepsilon(\text{PWR-7}) = 6.5\text{E-7/py} \end{cases}$$

$$T_1MLUO - \begin{cases} \gamma(\text{PWR-3}) = 2.7\text{E-6/py} \\ \beta(\text{PWR-5}) = 3.9\text{E-8/py} \\ \varepsilon(\text{PWR-7}) = 2.7\text{E-6/py} \end{cases}$$

$$T_3MLUO - \begin{cases} \gamma(\text{PWR-3}) = 5.5\text{E-7/py} \\ \beta(\text{PWR-5}) = 8.0\text{E-9/py} \\ \varepsilon(\text{PWR-7}) = 5.5\text{E-7/py} \end{cases}$$

$$T_2MQD - \begin{cases} \gamma(\text{PWR-3}) = 7.5\text{E-7/py} \\ \beta(\text{PWR-5}) = 1.1\text{E-8/py} \\ \varepsilon(\text{PWR-7}) = 7.5\text{E-7/py} \end{cases}$$

Grand Gulf:

$$T_1PQI - \begin{cases} \alpha(\text{BWR-1}) = 1.5\text{E-8/py} \\ \delta(\text{BWR-2}) = 1.5\text{E-6/py} \end{cases}$$

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

$T_{23}PQI$ - $\begin{cases} \alpha(\text{BWR-1}) = 3.7\text{E-8/py} \\ \delta(\text{BWR-2}) = 3.7\text{E-6/py} \end{cases}$

$T_{1}PQE$ - $\begin{cases} \gamma(\text{BWR-3}) = 9.5\text{E-8/py} \\ \delta(\text{BWR-4}) = 9.5\text{E-8/py} \end{cases}$

$T_{23}PQE$ - $\begin{cases} \gamma(\text{BWR-3}) = 2.6\text{E-7/py} \\ \delta(\text{BWR-4}) = 2.6\text{E-7/py} \end{cases}$

SI - $\begin{cases} \alpha(\text{BWR-1}) = 4.6\text{E-8/py} \\ \delta(\text{BWR-2}) = 4.6\text{E-6/py} \end{cases}$

$T_{1}QW$ - $\delta(\text{BWR-2}) = 4.5\text{E-6/py}$

$T_{23}QW$ - $\delta(\text{BWR-2}) = 1.1\text{E-5/py}$

$T_{1}QUV$ - $\begin{cases} \gamma(\text{BWR-3}) = 9.2\text{E-7/py} \\ \delta(\text{BWR-4}) = 9.2\text{E-7/py} \end{cases}$

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 6.1E-8/py
PWR-2 = 2.5E-6/py
PWR-3 = 1.7E-5/py
PWR-4 = 3.6E-8/py
PWR-5 = 2.7E-7/py
PWR-6 = 2.5E-6/py
PWR-7 = 2.1E-5/py

BWR-1 = 9.8E-8/py
BWR-2 = 2.5E-5/py
BWR-3 = 1.3E-6/py
BWR-4 = 1.3E-6/py

TABLE 1. (contd)

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

$\bar{F}_{PWR}$ = 4.3E-5/py $\bar{F}_{BWR}$ = 2.8E-5/py

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

W_{PWR} = 1.05E+2 man-rem/py W_{BWR} = 1.85E+2 man-rem/py

10. Adjusted-Case Values for Affected Parameters:

NUREG/CR-0848 summarizes LERs filed during the period 1965-1978 relating to valve failures (Scott and Gallager 1979). The tabular data provided was utilized to estimate the reduction in number of reports due to resolution of this issue. It was assumed that administrative installation, maintenance and operator error were not affected (i.e., not directly applicable to failure due to hardware malfunction) and that, due to issue resolution, design and fabrication problems resulting in valve failures were reduced. By decreasing the number of valve failures due to design and fabrication errors by 25%, fatigue failure (assumed to be a direct result of design error) by 25% and all inherent causes by 10%, the total number of projected reports is reduced by 9% in BWRs and PWRs. Therefore, it was assumed that the probability of hardware failure of valves for both PWRs and BWRs due to issue resolution was reduced by 9%. This assumption was also applied to pumps.

The following is a list of the adjusted-case values for the affected parameters.

Oconee:

B = C =	3.29E-3
D = E =	2.29E-3
CONST1 =	1.87E-4
CONST2 =	5.76E-4
A1 = C1 =	9.71E-3
B1 =	3.47E-2

TABLE 1. (contd)

10. Adjusted-Case Values for Affected Parameters (contd):

Q' =	4.55E-2
F1 =	1.31E-3
G1 =	1.34E-2
D•E =	4.85E-4
W•X =	8.60E-5
B•W = C•X =	2.69E-5
D•X = E•W =	2.05E-4
B•D = E•C =	6.21E-5

Grand Gulf:

H =	2.10E-2
P =	9.10E-2
R =	5.05E-2
L =	2.10E-2
LA2 = LB2 =	1.39E-2
LB1 =	1.33E-2
LC =	2.13E-2
PA27 = PB27 =	7.37E-4
VGA1 = VGB1 =	1.46E-2
VGA2 = VGB2 =	2.33E-2
SA = SB =	1.42E-2
SSA = SSB =	2.04E-2
SSC =	1.39E-2
V1 = V2 =	7.94E-3
SCVA = SCVB =	3.10E-2

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Oconee:

$$T_2MLU - \begin{cases} \gamma(PWR\text{-}3) = 5.5E\text{-}7/py \\ \beta(PWR\text{-}5) = 8.1E\text{-}9/py \\ \varepsilon(PWR\text{-}7) = 5.5E\text{-}7/py \end{cases}$$

$$T_1MLU - \begin{cases} \gamma(PWR\text{-}3) = 9.3E\text{-}7/py \\ \beta(PWR\text{-}5) = 1.4E\text{-}8/py \\ \varepsilon(PWR\text{-}7) = 9.3E\text{-}7/py \end{cases}$$

$$T_2MQH - \begin{cases} \gamma(PWR\text{-}3) = 4.7E\text{-}6/py \\ \beta(PWR\text{-}5) = 6.9E\text{-}8/py \\ \varepsilon(PWR\text{-}7) = 4.7E\text{-}6/py \end{cases}$$

$$S_3H - \begin{cases} \gamma(PWR\text{-}3) = 6.0E\text{-}7/py \\ \beta(PWR\text{-}5) = 8.8E\text{-}9/py \\ \varepsilon(PWR\text{-}7) = 6.0E\text{-}7/py \end{cases}$$

$$S_1D - \begin{cases} \alpha(PWR\text{-}1) = 1.1E\text{-}8/py \\ \gamma(PWR\text{-}3) = 2.2E\text{-}7/py \\ \beta(PWR\text{-}5) = 8.0E\text{-}9/py \\ \varepsilon(PWR\text{-}7) = 8.8E\text{-}7/py \end{cases}$$

$$T_2MQFH - \begin{cases} \gamma(PWR\text{-}2) = 2.1E\text{-}6/py \\ \beta(PWR\text{-}4) = 3.1E\text{-}8/py \\ \varepsilon(PWR\text{-}6) = 2.1E\text{-}6/py \end{cases}$$

$$S_3FH - \begin{cases} \gamma(PWR\text{-}2) = 9.0E\text{-}8/py \\ \beta(PWR\text{-}4) = 1.3E\text{-}9/py \\ \varepsilon(PWR\text{-}6) = 9.0E\text{-}8/py \end{cases}$$

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

$$S_2FH - \begin{cases} \alpha(\text{PWR-1}) = 5.6E\text{-}10/py \\ \beta(\text{PWR-4}) = 4.1E\text{-}10/py \\ \varepsilon(\text{PWR-6}) = 4.5E\text{-}8/py \end{cases}$$

$$T_2MLUO - \begin{cases} \gamma(\text{PWR-3}) = 3.6E\text{-}6/py \\ \beta(\text{PWR-5}) = 5.3E\text{-}8/py \\ \varepsilon(\text{PWR-7}) = 3.6E\text{-}6/py \end{cases}$$

$$S_2D - \begin{cases} \alpha(\text{PWR-1}) = 5.9E\text{-}9/py \\ \gamma(\text{PWR-3}) = 1.2E\text{-}7/py \\ \beta(\text{PWR-5}) = 4.3E\text{-}9/py \\ \varepsilon(\text{PWR-7}) = 4.7E\text{-}7/py \end{cases}$$

$$S_3D - \begin{cases} \gamma(\text{PWR-3}) = 6.0E\text{-}7/py \\ \beta(\text{PWR-5}) = 8.8E\text{-}9/py \\ \varepsilon(\text{PWR-7}) = 6.0E\text{-}7/py \end{cases}$$

$$T_1MLUO - \begin{cases} \gamma(\text{PWR-3}) = 2.4E\text{-}6/py \\ \beta(\text{PWR-5}) = 3.5E\text{-}8/py \\ \varepsilon(\text{PWR-7}) = 2.4E\text{-}6/py \end{cases}$$

$$T_3MLUO - \begin{cases} \gamma(\text{PWR-3}) = 4.8E\text{-}7/py \\ \beta(\text{PWR-5}) = 7.0E\text{-}9/py \\ \varepsilon(\text{PWR-7}) = 4.8E\text{-}7/py \end{cases}$$

$$T_2MQD - \begin{cases} \gamma(\text{PWR-3}) = 2.9E\text{-}8/py \\ \beta(\text{PWR-5}) = 4.2E\text{-}10/py \\ \varepsilon(\text{PWR-7}) = 2.9E\text{-}8/py \end{cases}$$

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

Grand Gulf:

$$T_1PQI - \begin{cases} \alpha(BWR\text{-}1) = 1.3E\text{-}8/py \\ \delta(BWR\text{-}2) = 1.3E\text{-}6/py \end{cases}$$

$$T_{23}PQI - \begin{cases} \alpha(BWR\text{-}1) = 2.9E\text{-}8/py \\ \delta(BWR\text{-}2) = 2.9E\text{-}6/py \end{cases}$$

$$T_1PQE - \begin{cases} \gamma(BWR\text{-}3) = 7.5E\text{-}8/py \\ \delta(BWR\text{-}4) = 7.5E\text{-}8/py \end{cases}$$

$$T_{23}PQE - \begin{cases} \gamma(BWR\text{-}3) = 2.2E\text{-}7/py \\ \delta(BWR\text{-}4) = 2.2E\text{-}7/py \end{cases}$$

$$SI - \begin{cases} \alpha(BWR\text{-}1) = 3.6E\text{-}8/py \\ \delta(BWR\text{-}2) = 3.6E\text{-}6/py \end{cases}$$

$$T_1QW - \delta(BWR\text{-}2) = 4.3E\text{-}6/py$$

$$T_{23}QW - \delta(BWR\text{-}2) = 9.8E\text{-}6/py$$

$$T_1QUV - \begin{cases} \gamma(BWR\text{-}3) = 8.9E\text{-}7/py \\ \delta(BWR\text{-}4) = 8.9E\text{-}7/py \end{cases}$$

TABLE 1. (contd)

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 1.8E-8/py

PWR-2 = 2.0E-6/py

PWR-3 = 1.4E-5/py

PWR-4 = 3.0E-8/py

PWR-5 = 2.2E-7/py

PWR-6 = 2.0E-6/py

PWR-7 = 1.5E-5/py

BWR-1 = 7.8E-8/py

BWR-2 = 2.2E-5/py

BWR-3 = 1.2E-6/py

BWR-4 = 1.2E-6/py

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

$\bar{F}^*_{PWR}$ = 3.3E-5/py $\bar{F}^*_{BWR}$ = 2.4E-5/py

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

W^*_{PWR} = 86 man-rem/py W^*_{BWR} = 163 man-rem/py

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

$\Delta\bar{F}_{PWR}$ = 1.0E-5/py $\Delta\bar{F}_{BWR}$ = 4.0E-6/py

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

ΔW_{PWR} = 19 man-rem/py ΔW_{BWR} = 22 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
6.6E+4	1.3E+7	0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Assessment of Failure and Reliability of Pumps and Valves (C-11)

2. Affected Plants (N):

			N
PWR:	planned		7
	operating		83
BWR:	planned		4
	operating		40
		Total	134

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

			$\bar{T}$(yr)
PWR:	planned		30.0
	operating		24.8
		All PWRs	25.2
BWR:	planned		30.0
	operating		23.0
		All BWRs	23.6

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

$\Delta(\bar{FD}_R)_{PWR}$ = (1.0E-5/py(19,900 man-rem) = 1.9E-1 man-rem/py

$\Delta(\bar{FD}_R)_{BWR}$ = (4.0E-6/py(19,900 man-rem) = 8.0E-2 man-rem/py

TABLE 2. (contd)

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
5.1E+2	1.5E+4	0

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

For operating plants, implementation could mean replacement of valves and pumps designated as inadequate through resolution of the study portion of this issue. For planned plants, implementation is essentially eliminated because replacement valves are introduced during design and construction phases.

Using data from NUREG/CR-0848 (Scott and Gallaher 1979) and assuming all plants operating between 1965-1978 reported, the average number of plants operating during the reporting period was as follows:

PWR: 17 plants
BWR: 14 plants

Of the total number of valve failures reported, 678 PWR and 639 BWR valve failures were attributed to design, fabrication, fatigue and inherent causes. It is recognized that the structural integrity of many values exposed to adverse conditions is adequate, that these figures do not reveal which valves were repeatedly replaced, and that potentially more valves would be included during exposure of plants to accident conditions or simply longer lifetimes. Assuming these failures to occur over the given 14-year period, with all plants reporting and the above number of PWRs and BWRs in operation, the following failure rates are estimated:

$$\text{PWR} \quad \frac{678 \text{ failures}}{17 \text{ plants} \quad 14 \text{ yr}} \qquad 3 \text{ valve failures/plant-yr}$$

$$\text{BWR} \quad \frac{639 \text{ failures}}{14 \text{ plants} \quad 14 \text{ yr}} \qquad 3 \text{ valve failures/plant-yr}$$

The same failure rates are assumed for pumps.

Assuming that the failure rates cited above represent 15% of the potential valve and pump failures (15% of the inadequate designs or

TABLE 2. (contd)

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

fabricated equipment) avoidable by the SIR over a plant's full lifetime, then an average of 20 valves and 20 pumps would require replacing per plant. If one assumes that some pumps require only replacement parts (assume 50%), then 10 pumps might need replacement. Assuming an average of 40 man-hours for replacement of valves and 80 man-hours for replacement of pumps:

(20 valves/plant)(40 man-hr/valve) = 800 man-hr/plant

(10 pumps/plant)(80 man-hr/pump) = 800 man-hr/plant

Total = 1600 man-hr/plant (only operating plants)

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

It is assumed that the dose per task for maintenance on pumps and valves averages 0.05R/hr (Palo Verde FSAR 1981).

D = (1600 man-hr/plant)(0.05 R/hr) = 80 man-rem/plant

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

ND = (123)(80 man-rem/plant) = 9.8E+3 man-rem

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

Operation and maintenance of replaced equipment is assumed equal to original equipment. Labor savings are obtained because of a reduced failure rate and repairs.

Base Case: It is assumed that 1 man-wk is required to replace a failed valve and that 2 man-wk are required to operate/maintain a replace a failed pump. Also assuming that the number of replacements equals the number of potential failures addressed by the SIR, one obtains the following labor for both PWRs and BWRs:

(20 valves/plant)(1 man-wk/valve) + (10 pumps/plant)(2 man-wk/pump) = 40 man-wk/plant

Resolved Case: Assuming that the number of failures of valves and pumps addressed by the SIR (i.e., failures due to design and fabrication error) is reduced by 25% due to the SIR (see Table 1, Step 10), the replacement of 15 failed valves and 8 pumps is still necessary over the remaining life of the plant. The labor for both PWRs and BWRs is as follows:

TABLE 2. (contd)

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

(15 valves/plant)(1 man-wk/valve) + (8 pumps/plant)(2 man-wk/pump) = 31 man-wk/plant

The difference in labor between the resolved case and the base case is

31 man-wk/plant - 40 man-wk plant = -9 man-wk/plant (applies to all plants)

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

Again, a 0.5 R/hr dose rate is assumed for maintenance on pumps and valves.

$\bar{T}D_o$[(a)] = (-9 man-wk/plant)(40 man-hr/man-wk)(0.05R/hr)

= -18 man-rem/plant

(Negative sign indicates reduction.)

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

$N\bar{T}D_o$ = (134)(-18 man-rem/plant) = -2.4E+3 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
7.4E+3	2.2E+4	2.5E+3

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Best estimates are used for labor time and outage time in the generic issue portion, as well as in the equipment replacement portion. Table 3 includes the results of this analysis.

(a) In this issue analysis, this value is calculated over the entire plant lifetime.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Assessment of Failure and Reliability of Pumps and Valves (C-11)

2. Affected Plants (N):

			N
PWR:	planned		7
	operating		83
BWR:	planned		4
	operating		40
		Total	134

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

			$\bar{T}$(yr)
PWR:	planned		30.0
	operating		24.8
		All PWRs	25.2
BWR:	planned		30.0
	operating		23.0
		All BWRs	23.6

Industry Costs (Steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, $\Delta(\bar{FA})$:

$$\Delta(\bar{FD}_R)_{PWR} = (\$1.65E+9)(1.0E-5/py) = \$1.65E+4/py$$

$$\Delta(\bar{FD}_R)_{BWR} = (\$1.65E+9)(4.0E-6/py) = \$6.6E+3/py$$

TABLE 3. (contd)

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

Best Estimate	Upper Bound	Lower Bound
\$4.4E+7	\$1.3E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

Costs are estimated for SIR implementation on this issue assuming that 30 parts were replaced per operating plant as a result of a plant walk-down and analysis of new criteria. This is expected to reduce part failures to 25% of their current rate. Cost estimates are divided into labor and parts.

Labor: a) 2 man-yr/plant (88 man-wk/plant) for pump/valve survey and analysis

b) 1 man-wk/valve replacement; 2 man-wk pump replacement

Parts: 10 pump parts/plant @ \$5.0E+5 each; 20 valve parts/plant @ \$3.0E+4 each.

(These estimates apply only to operating plants.)

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

I = (88 man-wks/plant + (2 man-wk/pump)(10 pumps/plant) + (1 man-wk/valve)(20 valves/plant)(2270 man-wk) + (10 pumps/plant) (\$5.0E+5/pump) + (20 valves/plant)(\$3.0E+4/valve)

= \$5.89E+6/plant

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

NI = (123)(\$5.89E+6/plant) = \$7.25E+8

TABLE 3. (contd)

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

Base- and resolved-case labor estimates assume 1 man-wk to operate/maintain a replaced valve and 2 man-wk to operate/maintain a replaced pump, with a 25% reduction in the number of resolved-case failures (see Table 2, Step 9). Following the same calculational procedure used in Table 2, Step 9, the difference in labor between the resolved case and the base case is -9 man-wk/plant (applicable to all plants). Note that this estimate is over the entire plant lifetime.

If it is estimated that outage time averages two months/yr and that 5% of all outages are attributed to pump and valve malfunctions, then an estimate of base-case down-time is approximately 3 days/ry. Assuming a 25% reduction in pump and valve failures for the adjusted case, a resulting 25% decrease in down-time is expected. This reduction in downtime amounts to 0.75 days/ry.

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

$\bar{T}I_0$[a] (for labor) = (-9 man-wk/plant)($2270 man-wk)
= -$2.04 +4/plant

I_0(for saved down time) = ($3.0E+5/day)(-0.75 days/ry)
= -$2.25E+5/ry

(Negative signs indicate reductions.)

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

NTI_0 = ($-2.04E+4/plant)(134 plants) + ($-2.25E +5/ry)[(90)(25.2 yr)(44)(23.6)]
= -$7.47E+8

(Negative signs indicate reduction.)

(a) In this issue analysis, this value is calculated over the entire plant lifetime.

TABLE 3. (contd)

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
-\$2.2E+7	\$5.0E+8	-5.4E+8

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

The NRC will review the generic issue, assess the failures reported to date on active pumps and valves, make recommendations regarding possible equipment specifications and applications, and monitor implementation activities at the operating plants. It is estimated that technical assistance funding will be required for review and testing of new and possibly old designs. Testing procedures for qualifying equipment, however, will be accommodated under separate issues.

Generic issue development: 3 man-yr
Technical assistance funds: \$2.0E+6

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

C_D = (3 man-yr)(\$1.0E+5/man-yr) + \$2.0E+6 = \$2.30E+6

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

Monitoring implementation activities at operating plants and reviewing results:

4 man-wk/operating plant

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

C = (4 man-wk/plant)(\$2270/man-wk) = \$9.08 E+3/plant

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

NC = (\$9.08E+3/plant)(123) = \$1.12E+6

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

A program review is anticipated to determine performance of new equipment and acceptability of any licensing changes related to resolution of this issue. An estimate of time over all the plants would be 12 man-wk/yr for 5 consecutive years following issue resolution. The cost is estimated directly in Step 20. Routine inspection labor is assumed to be about the same both prior and subsequent to SIR.

TABLE 3. (contd)

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

Cost estimated directly in next step.

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

$N\bar{T}C_o$ = (12 man-wk/yr)(5 yr)($2270/man-wk) = $1.36E+5

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$3.6E+6	$4.8E+6	$2.3E+6

REFERENCES

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: D-1, Advisability of a Seismic Scram--High Trip Level

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

A seismic scram system set to trip at a high level ($\geq$0.60 SSE) may prove beneficial in alleviating stresses imposed on the primary system during earthquakes from scrams subsequently induced by the earthquakes. While current evidence may not substantiate this claim (further analysis may be pending), it is assumed that all plants install high-level seismic scram systems (except those plants currently with such systems or planning to install them).

AFFECTED PLANTS	BWR:	Operating = 24	Planned = 20
	PWR:	Operating = 46	Planned = 39

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	$\leq$790
OCCUPATIONAL DOSES	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	$\leq$9.2

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	23
SIR Operation/Maintenance =	33
Total of Above =	56
Accident Avoidance =	$\leq$0.77
NRC COSTS:	
SIR Development =	0.050
SIR Implementation Support =	0.59
SIR Operation/Maintenance Review =	0.83
Total of Above =	1.5

ADVISABILITY OF A SEISMIC SCRAM--HIGH TRIP LEVEL

ISSUE D-1

1.0 SAFETY ISSUE DESCRIPTION

The advisability of requiring commercial nuclear power plants to install seismic scram systems set at a high trip level, e.g., 0.60 of the Safe Shutdown Earthquake (SSE), has been examined by Lawrence Livermore National Laboratory (LLNL) (O'Connell and Wells 1983). The high trip level is intended to reduce the frequency of reactor shutdowns due to low acceleration earthquakes, aftershocks, or spurious causes. In a previous investigation by LLNL (Cummings et al. 1976), the proposed scram system was to be activated by the compressional waves (P waves) when this first arrival caused displacement or acceleration greater than the calculated maximum allowable P wave for an Operating Basis Earthquake (OBE). This automatic shutdown would trip three safety systems (control rods, main steam isolation valves, and turbine stop valves) before arrival of the strong displacement shear waves (S waves), several seconds after the P waves had arrived. In contrast, the high trip level system is triggered by detection of an acceleration greater than a specified threshold level, apparently without regard for the nature of the elastic earthquake waves. Thus, the high-level trip does not necessarily occur before the reactor is subject to the strong motion of the shear waves. Nevertheless, this system would usually give a lead time of 5 to 20 seconds before initiation of other reactor trips, e.g., trips caused by turbine vibration or loss of offsite AC power. This lead time is sufficient to achieve significant changes in reactor state--3 seconds to scram and 5 to 10 seconds for 50 percent reduction in the heat generation rate.

Earthquakes are a concern throughout the United States. Although earthquake occurrence is less frequent in the eastern areal two-thirds of the 48 conterminous states, the area affected may be much larger because of elastic wave transmission through rock formations. Current U.S. regulations require seismic instrumentation for timely information and evaluation. If the OBE is exceeded, the plant must be shut down for inspection. A normal shutdown procedure may be initiated in the control room.

To identify the possible advantages of a seismic scram system, it is necessary to consider possible transients and accident sequences that could lead to core melt and offsite exposure (O'Connell and Wells 1983). An early seismic trip that precludes waiting for a later trip will reduce transient pressure and loads and the heat generation rate in the core. This will decrease the burden on the reactor's safety systems, e.g., safety/relief valves and turbine-driven pumps. In the event of a loss-of-coolant accident (LOCA), an earlier trip will reduce the fuel rod temperature transient and the containment vessel pressure. Less fluid will be lost during the blowdown phase before the safety injection system operating pressure is reached.

Three potential disadvantages are associated with a seismic scram system: 1) a seismic trip would be more likely to disable offsite AC power needed for the reactor's safety systems; 2) a reactor trip and transient could occur when none would have started without the seismic scram system (i.e., resulting from spurious sources); 3) for a multi-unit site or a wide-area earthquake, many plants could be tripped almost simultaneously.

NUREG/CR-2513 utilizes a decision tree method to compare the risks of employing and not employing a high-level seismic scram system (O'Connell and Wells 1983). However, due to recent NRC questions on the validity of some of the assumptions and analysis techniques, it is felt that direct use of this study's results is currently questionable.[a] Pending possible reanalysis by LLNL, an alternative approach is taken in estimating the public risk reduction for this issue. This is discussed in Attachment 1 to the Public Risk Reduction Work Sheet.

PROPOSED ISSUE RESOLUTION

If provision for high trip level seismic detectors and scram systems is deemed necessary, existing designs and equipment may be elected. For example, the Diablo Canyon PWR has a seismic scram system which uses three triaxial seismic acceleration detectors at diverse locations near and in the reactor building. If any two of the three detectors signal an acceleration above the action level (0.35g in the free field, which is 47 percent of the Diablo Canyon SSE level), then the reactor scram system is activated. The scram also trips the turbine generator, and the turbine bypass valves open. The reactor decay heat is removed through the steam generators, with the steam in the secondary circuit bypassing the turbine and going into the condenser.

In contrast to Diablo Canyon, the triggering acceleration may be the g-force equivalent to 60 percent of the SSE for the particular reactor. (The 47 percent level at Diablo canyon was probably set because of its proximity to a fault line.) Installation of high trip level ($\geq$0.60 SSE) seismic scram systems at all plants (operating and planned), except the five currently having or planning to have such systems, is taken to be the safety issue resolution (SIR) for D-1.

AFFECTED PLANTS

This issue affects all BWRS and PWRs outside California. San Onofre 1, 2 and 3 and Diablo Canyon 1 and 2 are excluded because seismic detection and scram systems are already planned or in place.

(a) Burdick, G., "Review of Seismic Scram Report, UCRL-53037." March 3, 1983, Memorandum to G. Arndt, U.S. Nuclear Regulatory Commission, Washington, D.C.

2.0 SAFETY ISSUE RISK AND DOSE

The results of the analyses of the public risk reduction and occupational dose associated with issue resolution are summarized in Tables 1 and 2, respectively. Attachment 1 to Table 1 is provided to develop the alternative approach to estimating the public risk reduction.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Advisability of a Seismic Scram--High Trip Level (D-1)

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

   All plants except the San Onofre 1, 2, and 3 and Diablo Canyon 1 and 2 PWRs are assumed to be affected.

	N	$\bar{T}$ (yr)
PWRs	85	28.8
BWRs	44	27.4

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR[a]

   Grand Gulf 1 - representative BWR[a]

4. Parameters Affected by SIR:

   Oconee

   S_{1e}, S_{2e}, T_{1e}, S_{3e}, T_{1e}, T_{2e} — Earthquake-induced LOCA and transient initiators corresponding to S_1, S_2, S_3, T_1, and T_2 terms defined in Table A.4 of NUREG/CR-2800 (Andrews et al. 1983).

   Grand Gulf

   S_e, T_{1e}, T_{2e} — Earthquake-induced LOCA and transient initiators corresponding to S, T_1, and T_{23} terms (T_{2e} is assumed to induce the same sequences as T_{23}) defined in Table B.4 of NUREG/CR-2800.

(a) See Attachment 1.

TABLE 1. (contd)

5. Base-Case Values for Affected Parameters:

Oconee[a]	Grand Gulf[a]
$\left.\begin{array}{l} S_{1e} \\ S_{2e} \\ S_{3e} \end{array}\right\} = 1.9E\text{-}6/py$	$S_e = 1.9E\text{-}6/py$
$T_{1e} = 2.7E\text{-}4/py$	$T_{1e} = 2.7E\text{-}4/py$
$T_{2e} = 2.6E\text{-}4/py$	$T_{2e} = 2.6E\text{-}4/py$

6. Affected Accident Sequences and Base-Case Frequencies:

These are listed in Attachment 1 and not repeated here. Note that sequences T_{2e}KMU (Oconee) and T_{2e}C (Grand Gulf) are not affected with respect to the issue resolution of installation of a high-level seismic scram system. Their inclusion in Attachment 1 reflects their potential for being seismically induced. This is discussed further in Attachment 1. All values in subsequent steps of this work sheet exclude these two unaffected sequences.

7. Affected Release Categories and Base-Case Frequencies:

Oconee[a]	Grand Gulf[a]
PWR-1 = 1.4E-9/py	BWR-1 = 8.5E-11/py
PWR-2 = 3.3E-9/py	BWR-2 = 1.7E-8/py
PWR-3 = 4.3E-8/py	BWR-3 = 1.4E-9/py
PWR-4 = 9.3E-11/py	BWR-4 = 1.4E-9/py
PWR-5 = 1.2E-9/py	
PWR-6 = 8.2E-9/py	
PWR-7 = 1.3E-7/py	

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

PWR: $\bar{F}$ = 1.8E-7/py BWR: $\bar{F}$ = 2.0E-8/py

(a) See Attachment 1.

TABLE 1. (contd)

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

PWR: $W = 0.26$ man-rem/py BWR: $W = 0.13$ man-rem/py

10-14. Steps Related to Adjusted-Case Values of Affected Parameters, Accident Sequences, Release Categories, Core-Melt Frequency, and Public Risk:

The reductions in core-melt frequency and public risk are estimated directly in Steps 15 and 16, respectively. Thus, these steps are omitted.

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

PWR: $\Delta\bar{F} \leq 1.8E-7/py$[a] BWR: $\Delta\bar{F} \leq 2.0E-8/py$[a]

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

PWR: $\Delta W \leq 0.26$ man-rem/py[a] BWR: $\Delta W \leq 0.13$ man-rem/py[a]

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
$\leq$ 790	2.4E+4	not estimated

(a) See Attachment 1. The estimates given here represent maximum possible reductions based on conservatisms discussed in Attachment 1, hence the use of the "$\leq$" symbols.

ATTACHMENT 1

NUREG/CR-2513 (O'Connell and Wells 1983) presents a reasonably detailed scoping analysis of the reduction in core-melt frequency attributable to installation of a high-level seismic scram system. However, the NRC has questioned the validity of some of the assumptions and analysis techniques employed in this study.[a] Pending possible reanalysis by LLNL, it is felt that direct use of this study's results for estimating the core-melt frequency reduction is currently questionable. An alternative approach is taken which utilizes some of the LLNL assumptions (which have not been questioned) in the framework of the standardized analysis technique of NUREG/CR-2800 (Andrews et al. 1983).

NUREG/CR-2513 discusses several advantages of a high-level seismic scram system regarding core-melt frequency, primarily in connection with earlier reactor scram than would occur in the absence of such a system, provided that the earthquake would have eventually scrammed the reactor anyway. In such instances, the earlier scram (compared to a later scram) reduces the phenomenological stresses (e.g., pressure and temperature) which would be imposed on the primary system during a transient or LOCA. This is believed to reduce the frequency of core-melt from seismic initiation, and arguments are presented to substantiate this claim.

Difficulty arises in attempting to specify which aspects of transient or LOCA scenarios are potentially affected by the earlier seismic scram. If an earthquake should cause structural collapse or falling equipment which could damage engineered safety features, then the benefits of an earlier scram might be minimal at best. If an earthquake should cause equipment vibration, then the reduction in phenomenological stresses due to the earlier scram might have some small benefit in reducing the likelihood of failure in this vibrating equipment. Such effects are difficult to quantify; they are felt to be minimal at most. It is assumed for this analysis that the prime benefit of an earlier seismic scram lies in reducing the frequency of a seismically-induced transient or LOCA initiator as a result of the reduction in phenomenological stresses. Thus, the high-level seismic scram system is given no credit for affecting the likelihood of failures conditional upon transient or LOCA initiators.

The quantitative portion of the analysis begins with an estimation of the core-melt frequency due to earthquakes where the earthquake's only effect is to induce a transient or LOCA initiator. No attempt is made to estimate the total core-melt frequency due to earthquakes (which would include the earthquake's effects on conditional failures of engineered safety features) since the seismic scram system has been assumed to affect only the frequency of seismically-induced initiators.

(a) Burdick, G. "Review of Seismic Scram Report, UCRL-53037." March 3, 1983, Memorandum to G. Arndt, U.S. Nuclear Regulatory Commission, Washington, D.C.

Tables 7.1 and 7.2 of NUREG/CR-2513 (reproduced below) present the annual frequency of earthquakes at the Zion site and the conditional probabilities of transient or LOCA initiators given an earthquake.

Earthquake Frequency at Zion Site (During Plant Operation)[a]

Earthquake Interval (SSE)	Frequency (1/yr)
0.4-0.6	8.4E-4
0.6-0.9	4.5E-4
0.9-1.8	2.5E-4
1.8-2.5	1.3E-5
>2.5	2.2E-6

Conditional Probability of Transient or LOCA Initiator (Given an Earthquake)[a]

Earthquake Interval (SSE)	Conditional Probability		
	LOCA	T_1 Transient	T_2 Transient
0.6-0.9	6.0E-5	0.36	0.24
0.9-1.8	0.0025	0.40	0.59
1.8-2.5	0.052	0.019	0.93
>2.5	0.26	0	0.74

Based on these tables, the following frequencies of seismically-induced transient or LOCA initiators are calculated:[b]

LOCA = (4.5E-4/py)(6.0E-5) + (2.5E-4/py)(0.0025) + (1.3E-5/py)(0.052) + (2.2E-6/py)(0.26) = 1.9E-6/py

T_1 = (4.5E-4/py)(0.36) + (2.5E-4/py)(0.40) + (1.3E-5/py)(0.019) = 2.7E-4/py

T_2 = (4.5E-4/py)(0.24) + (2.5E-4/py)(0.59) + (1.3E-5/py)(0.93) + (2.2E-6/py)(0.74) = 2.6E-4/py

(a) O'Connell and Wells 1983.
(b) Since no data are provided on the conditional probability of transient or LOCA initiators for 0.4-0.6 SSEs, no contribution is estimated for this SSE interval.

The initiators are designated below with "e" subscripts to indicate that they are induced by earthquakes. The following accident sequences are presumed to result, based on the Oconee 3 and Grand Gulf 1 risk equations presented in NUREG/CR-2800 (the "S" terms represent the LOCA initiators):[a]

Sequence	Release Category		Frequency (1/py)
Oconee:			
$T_{2e}MLU$ -	γ PWR-3	=	5.4E-11
	β PWR-5	=	7.9E-13
	ε PWR-7	=	5.4E-11
$T_{1e}MLU$ -	γ PWR-3	=	1.3E-9
	β PWR-5	=	1.9E-11
	ε PWR-7	=	1.3E-9
$T_{1e}(B_3)MLU$ -	γ PWR-3	=	1.4E-9
	β PWR-5	=	2.1E-11
	ε PWR-7	=	1.4E-9
$T_{2e}MQH$ -	γ PWR-3	=	5.0E-10
	β PWR-5	=	7.2E-12
	ε PWR-7	=	5.0E-10
$S_{3e}H$ -	γ PWR-3	=	7.3E-9
	β PWR-5	=	1.1E-10
	ε PWR-7	=	7.3E-9

(a) Since NUREG/CR-2513 does not specify which LOCA initiators (large, small, etc.) can be seismically induced, all LOCA initiators are conservatively assumed to be potentially affected, each with a frequency of 1.9E-6/py.

ATTACHMENT 1. (contd)

Sequence	Release Category		Frequency (1/py)
$S_{1e}D$ -	α PWR-1	=	1.3E-9
	γ PWR-3	=	2.5E-8
	β PWR-5	=	9.3E-10
	ε PWR-7	=	1.0E-7
$T_{2e}MQFH$ -	γ PWR-2	=	2.3E-10
	β PWR-4	=	3.3E-12
	ε PWR-6	=	2.3E-10
$S_{3e}FH$ -	γ PWR-2	=	3.1E-9
	β PWR-4	=	4.5E-11
	ε PWR-6	=	3.1E-9
$S_{2e}FH$ -	α PWR-1	=	6.2E-11
	β PWR-4	=	4.5E-11
	ε PWR-6	=	4.9E-9
$T_{2e}MLUO$ -	γ PWR-3	=	3.6E-10
	β PWR-5	=	5.3E-12
	ε PWR-7	=	3.6E-10
$T_{2e}KMU$ -	γ PWR-3	=	3.5E-10
	β PWR-5	=	5.1E-12
	ε PWR-7	=	3.5E-10
$S_{2e}D$ -	α PWR-1	=	9.5E-11
	γ PWR-3	=	1.9E-9
	β PWR-5	=	6.9E-11
	ε PWR-7	=	7.6E-9

ATTACHMENT 1. (contd)

Sequence	Release Category		Frequency (1/py)
$S_{3e}D$ -	γ PWR-3	=	1.0E-9
	β PWR-5	=	1.5E-11
	ε PWR-7	=	1.0E-9
$T_{1e}MLUO$ -	γ PWR-3	=	3.5E-9
	β PWR-5	=	5.1E-11
	ε PWR-7	=	3.5E-9
$T_{2e}MQD$ -	γ PWR-3	=	6.8E-11
	β PWR-5	=	9.9E-13
	ε PWR-7	=	6.8E-11
Grand Gulf:(a)			
$T_{1e}PQI$ -	α BWR-1	=	2.1E-11
	δ BWR-2	=	2.1E-9
$T_{2e}PQI$ -	α BWR-1	=	1.4E-12
	δ BWR-2	=	1.4E-10
$T_{1e}PQE$ -	γ BWR-3	=	1.5E-10
	δ BWR-4	=	1.5E-10
$T_{2e}PQE$ -	γ BWR-3	=	1.0E-11
	δ BWR-4	=	1.0E-11
S_eI -	α BWR-1	=	6.2E-11
	δ BWR-2	=	6.2E-11

(a) T_{2e} is assumed to initiate the same accident sequences as T_{23} in Grand Gulf.

Sequence	Release Category		Frequency (1/py)
$T_{1e}QW$ -	δ BWR-2	=	8.1E-9
$T_{2e}QW$ -	δ BWR-2	=	4.6E-10
$T_{2e}C$ -	δ BWR-2	=	2.1E-10
$T_{1e}QUV$ -	γ BWR-3	=	1.2E-9
	δ BWR-4	=	1.2E-9

The affected release category and core-melt frequencies, and the affected public risk resulting from these seismically-induced accident sequences (where only the initiator frequencies are affected) are as follows:

Oconee

$(PWR\text{-}1)_e$ = 1.4E-9/py

$(PWR\text{-}2)_e$ = 3.3E-9/py

$(PWR\text{-}3)_e$ = 4.3E-8/py

$(PWR\text{-}4)_e$ = 9.3E-11/py

$(PWR\text{-}5)_e$ = 1.2E-9/py

$(PWR\text{-}6)_e$ = 8.2E-9/py

$(PWR\text{-}7)_e$ = 1.3E-7/py

$\bar{F}_e$ = 1.8E-7/py

W_e = 0.26 man-rem/py (using dose factors from Appendix D of NUREG/CR-2800, Andrews et al. 1983)

Grand Gulf

$(BWR\text{-}1)_e$ = 8.5E-11/py

$(BWR\text{-}2)_e$ = 1.7E-8/py

$(BWR\text{-}3)_e$ = 1.4E-9/py

$(BWR\text{-}4)_e$ = 1.4E-9/py

$\bar{F}_e$ = 2.0E-8/py

W_e = 0.13 man-rem/py (using dose factors from Appendix D of NUREG/CR-2800)

Installation of a high-level seismic scram system can presumably reduce the frequencies of all these sequences, with the exception of those where failure to scram is an inherent part of the sequence (i.e., $T_{2e}KMU$ in Oconee and $T_{2e}C$ in Grand Gulf), by reducing the frequencies of their seismically-induced initiators. Removing the contributions from the two failure-to-scram sequences from the totals for all these sequences, one calculates the following maximum reductions from installation of high-level seismic scram systems:

Oconee	Grand Gulf
$(\Delta \bar{F})_e$ = 1.8E-7/py	$(\Delta \bar{F})_e$ = 2.0E-8/py
$(\Delta W)_e$ = 0.26 man-rem/py	$(\Delta W)_e$ = 0.13 man-rem/py

These presume that the seismic scram systems eliminate the potential for seismically inducing transient and LOCA initiators, a conservative assumption from a risk reduction viewpoint (i.e., yielding the maximum possible risk reduction).

To this point, only the advantages of a high-level seismic scram system have been considered. NUREG/CR-2513 also discusses several disadvantages of a high-level seismic scram system regarding core-melt frequency, the prime one being in connection with additional spurious scrams resulting from the system itself. Since each scram places some stress on the primary system, a spurious scram bears some potential for inducing a transient sequence which could eventually result in a core melt. LLNL estimates a conditional probability of core melt due solely to a scram at 1E-6 or less. Coupled with their estimate that a seismic scram system could induce additional spurious scram at a frequency of 0.1/yr, LLNL obtains a potential 1E-7/py increase in core-melt frequency from the seismic scram system due to spurious scrams.

While these estimates may only be approximate, they serve to alert the analyst to the possibility that a high-level seismic scram system could lead to an overall increase in core-melt frequency (and public risk) should the spurious scram contribution outweigh that from earlier reactor scrams. For this analysis, the potential for increasing core-melt frequency, while recognized, is not quantified or included in the public risk reduction estimation. This provides a conservative estimate of the public risk reduction (i.e., yielding the maximum possible value).

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Advisability of a Seismic Scram--High Trip Level (D-1)

2. Affected Plants (N):

All plants except the San Onofre 1, 2, and 3 and Diablo Canyon 1 and 2 PWRs are assumed to be affected, i.e., 85 PWRs and 44 BWRs.

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

	$\bar{T}$ (yr)
85 PWRs	28.8
44 BWRs	27.4

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

$$\Delta(\bar{F}D_R)_{PWR} \leq (19{,}900 \text{ man-rem})(1.8E\text{-}7/\text{py}) = 0.0036 \text{ man-rem/py}$$

$$\Delta(\bar{F}D_R)_{BWR} \leq (19{,}000 \text{ man-rem})(2.0E\text{-}8/\text{py}) = 4.0E\text{-}4 \text{ man-rem/py}$$

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
$\leq$ 9.2	55	not estimated

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

Seismic detection devices for triggering shutdown would be installed on the containment foundation, essentially a nonradiation zone. The seismic trip system is a triaxial detector with a threshold tailored to an earthquake ground motion signature. Structural response detectors, if employed, may be installed within high-radiation zones, e.g., above the reactor pressure vessel; but these sensors do not accurately measure the magnitude of an earthquake because mounting and leverage features cause damping and magnification. It is assumed that these detectors will not be used due to their ineffectiveness. Thus, no radiation zone labor is assumed for either implementation or operation/maintenance, and $D = D_o = G = 0$.

3.0 SAFETY ISSUE COSTS

The results of the analysis of the industry and NRC costs associated with issue resolution are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Advisability of a Seismic Scram--High Trip Level (D-1)

2. Affected Plants (N):

All plants except the San Onofre 1, 2, and 3 and Diablo Canyon 1 and 2 PWRs are assumed to be affected.

		N
PWRs:	Operating	46
	Planned	39
		85
BWRs:	Operating	24
	Planned	20
		44

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

	$\bar{T}$ (yr)
85 PWRs	28.8
44 BWRs	27.4
129 LWRs	28.3

Industry Costs (Steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, $\Delta(\bar{FA})$:

$$\Delta(\bar{FA})_{PWR} \leq (\$1.65E+9)(1.8E-7/py) = \$300/py$$

$$\Delta(\bar{FA})_{BWR} \leq (\$1.65E+9)(2.0E-8/py) = \$33/py$$

TABLE 3. (contd)

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

Best Estimate	Upper Bound	Lower Bound
$\leq$ \$7.7E+5	\$4.6E+6	not estimated

6. Per-Plant Industry Resources for SIR Implementation:

Labor will be required to install and test the seismic scram system, including hookup to the control room scram circuits. This work can presumably be performed during scheduled outages. In addition, plants holding operating licenses will presumably require a Class III amendment.

Labor = 10 man-wk/plant

Equipment = seismic sensors, cables, recorders, etc.--cost estimated directly in next step

Class III License Amendment (operating plants only)--cost estimated directly in next step

These estimates should not vary between PWRs and BWRs.

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

	Cost (\$/plant)	
	Operating Plants	Planned Plants
Labor [(10)(\$2270)]	2.3E+4	2.3E+4
Equipment	1.5E+5	1.5E+5
Class III License Amendment	4000	-
I =	1.77E+5	1.73E+5

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

NI = (70 operating plants)(\$1.77E+5/plant) + (59 planned plants) (\$1.73E+5/plant) = \$2.26E+7

TABLE 3. (contd)

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

In addition to periodic testing and maintenance of the seismic scram system, it is assumed that operators will be retrained periodically on the system's use. Labor estimates are as follows:

Operator retraining	= 2 man-wk/py
System maintenance	= 2 man-wk/py
	4 man-wk/py

This labor is presumed to be the same for PWRs and BWRs.

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_O):

$$I_O = (4 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$9080/\text{py}$$

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

$$N\bar{T}I_O = (129 \text{ plants})(28.3 \text{ yr})(\$9080/\text{py}) = \$3.31\text{E+7}$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$5.6E+7	\$7.6E+7	\$3.6E+7

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Issue resolution development, which may include reanalysis by LLNL, is assumed to require the equivalent of 0.5 man-yr of NRC staff labor.

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

$$C_D = (0.5 \text{ man-yr})(\$1.0\text{E+5/man-yr}) = \$5.0\text{E+4}$$

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

Support of SIR implementation will most likely involve an initial inspection of the installed seismic scram system plus routine review of documentation. Two man-weeks/plant of NRC staff labor are assumed necessary (for both PWRs and BWRs).

TABLE 3. (contd)

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

C = (2 man-wk/plant)($2270/man-wk) = $4540/plant

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

NC = (129 plants)($4540/plant) = $5.86E+5

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

The seismic scram system will be included as part of a plant's routine inspection by the NRC. Only a small increase in NRC staff labor of 0.5 man-day/py is presumed necessary (for both PWRs and BWRs).

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

C_O = (0.5 man-day/py)(1 man-wk/5 man-days)($2270/man-wk) = $227/py

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

$N\bar{T}C_O$ = (129 plants)(28.3 yr)($227/py) = $8.29E+5

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$1.5E+6	$2.0E+6	$9.6E+5

REFERENCES

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

Cummings, G., et al. 1976. Advisability of Seismic Scram. UCRL-52156, Lawrence Livermore National Laboratory, Livermore, California.

O'Connell, W., and J. Wells. 1983. On the Advisability of an Automatic Seismic Scram. NUREG/CR-2513, UCRL-53037, Lawrence Livermore National Laboratory, Livermore, California.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: I.A.2.6(1-3,5), Long-Term Upgrading of Training and Qualifications (Simulators)

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This TMI action item calls for the upgrading of training and qualification for operational staff. The specific subjects (1,2,3,5) focus on reactor operators and emphasize the use of reactor simulators in training, requalification, and testing. The resolution of this issue is assumed to be a major enhancement of training and requalifications for reactor operators.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 1.2E+5

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	750

COST RESULTS ($$10^6$)

INDUSTRY COSTS:

SIR Implementation =	160
SIR Operation/Maintenance =	1900
Total of Above =	2100
Accident Avoidance =	62

NRC COSTS:

SIR Development =	0.28
SIR Implementation Support =	0.28
SIR Operation/Maintenance Review =	40
Total of Above =	40

LONG-TERM UPGRADING OF TRAINING AND QUALIFICATIONS (SIMULATORS)

ISSUE I.A.2.6(1-3,5)

1.0 SAFETY ISSUE DESCRIPTION

The TMI action item I.A.2.6, described in NUREG-0660 (NRC 1980), calls for the long-term upgrading of training and qualification for operations personnel. Subparts 1, 2, 3, and 5 delineate a program for upgrades for reactor operators, senior reactor operators, and shift supervisors and emphasize the use of simulators in training and requalification.

The assessment of this safety issue was conducted by Pacific Northwest Laboratory (PNL) staff with experience in reactor operator licensing, reactor operation, and general reactor safety, in consultation with General Physics Corporation. General Physics Corporation provides utility training services and has considerable experience with reactor simulators, providing procurement and startup assistance, operation and maintenance services, and simulator modifications.

It is assumed that this safety issue resolution (SIR) will take the form of upgrading utility training and qualification programs to meet the requirements of the Institute of Nuclear Power Operations (INPO) accreditation standards. Although these standards are not yet finalized, it is assumed that this will represent a major enhancement of the training and qualification programs.

Since many of the TMI action items associated with operator training are interrelated, it is difficult to assess them independently. This issue is strongly tied to I.A.4.1, Initial Simulator Improvement, which is also being assessed in this program. For the purposes of the analysis, these two issues are separated as follows: I.A.2.6(1-3,5) deals with training improvements, including the enhanced use of existing simulators, whereas I.A.4.1 deals with the improvement of simulators, which provide more realistic modeling of the actual plant. Either item, by itself, would improve operator performance. However, there is significant overlap. Therefore, if both items were implemented, the total improvement would be less than the sum of the individual contributions as assessed in the program.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with this SIR are estimated in this section. Analysis results are summarized in Tables 1 and 2, respectively.

The public risk reduction arises out of a reduction in core-melt frequency which comes from a reduction in operator error probabilities. Reduction in operator errors is assumed to result from the upgraded training and qualification which form the assumed SIR.

The upgrades presumably include an increase in time spent in simulator operation, both in training and in requalification. The simulator time is assumed to improve in quality as well as quantity. Emphasis on improvements in the operator's diagnostic capability is felt to be especially important. Furthermore, the enforcement activities in terms of NRC-administered examinations and inspection of training programs by the Office of Inspection and Enforcement (IE) are assumed to be strong and comprehensive.

Even with these assumptions describing the SIR, it is difficult to estimate the effect on operator error probabilities. Studies relating quantity and quality of training to error likelihoods for these areas do not exist. Clearly, as training improves, human errors decrease. However, the effect is obviously not linear. Based on engineering judgment, it was estimated that the resolution of this safety issue would result in a 30 percent reduction in operator error probabilities.

Regarding occupational dose associated with the SIR, none will presumably result from its implementation or operation/maintenance. The only occupational dose associated with the SIR is that saved by accident avoidance.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Long-Term Upgrading of Training and Qualifications (Simulators) [I.A.2.6(1-3,5)]

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

All plants are assumed to be affected.

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

3. Plants Selected for Analysis:

Oconee 3 - representative PWR

Grand Gulf 1 - representative BWR

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

TABLE 1. (contd)

4. Parameters Affected by SIR:

Oconee: B, C, D, E, CONST1, CONST2, A1, B1, C1, HHMAN, HPMAN, HPMAN1, HPRSCM, WXCM, D•E, B•W, C•X, D•X, E•W, B•D, E•C

5. Base-Case Values for Affected Parameters:

Original values from Appendix A (Andrews et. al 1983) are assumed.

6. Affected Accident Sequences and Base-Case Frequencies:

Sequence	Release category	Frequency
T_2MLU	γ (PWR-3) =	5.8E-7/py
	β (PWR-5) =	8.5E-9/py
	ε (PWR-7) =	5.8E-7/py
T_1MLU	γ (PWR-3) =	9.8E-7/py
	β (PWR-5) =	1.4E-8/py
	ε (PWR-7) =	9.8E-7/py
$T_1(B_3)MLU$	γ (PWR-3) =	1.1E-6/py
	β (PWR-5) =	1.6E-8/py
	ε (PWR-7) =	1.1E-6/py
T_2MQH	γ (PWR-3) =	3.2E-6/py
	β (PWR-5) =	4.7E-8/py
	ε (PWR-7) =	3.2E-6/py
S_3H	γ (PWR-3) =	2.8E-6/py
	β (PWR-5) =	4.1E-8/py
	ε (PWR-7) =	2.8E-6/py
S_1D	α (PWR-1) =	5.3E-8/py
	γ (PWR-3) =	1.1E-6/py
	β (PWR-5) =	3.9E-8/py
	ε (PWR-7) =	4.3E-6/py
T_2MQFH	γ (PWR-2) =	2.4E-6/py
	β (PWR-4) =	3.6E-8/py
	ε (PWR-6) =	2.4E-6/py

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

S_3FH -	γ (PWR-2) =	2.0E-6/py
	β (PWR-4) =	3.0E-8/py
	ϵ (PWR-6) =	2.0E-6/py
S_2FH -	α (PWR-1) =	1.2E-8/py
	β (PWR-4) =	8.9E-9/py
	ϵ (PWR-6) =	9.8E-7/py
T_2KMU -	γ (PWR-3) =	3.9E-6/py
	β (PWR-5) =	5.7E-8/py
	ϵ (PWR-7) =	3.9E-6/py
S_2D -	α (PWR-1) =	7.2E-9/py
	γ (PWR-3) =	1.4E-7/py
	β (PWR-5) =	5.2E-9/py
	ϵ (PWR-7) =	5.7E-7/py
S_3D -	γ (PWR-3) =	6.7E-7/py
	β (PWR-5) =	9.8E-9/py
	ϵ (PWR-7) =	6.7E-7/py
T_2MQD -	γ (PWR-3) =	7.2E-7/py
	β (PWR-5) =	1.1E-8/py
	ϵ (PWR-7) =	7.2E-7/py

(Note: In each affected accident sequence, the contribution from the non-dominant minimal cut sets is scaled by the ratio of the sum of the affected dominant minimal cut set frequencies to the sum of all the dominant minimal cut set frequencies.)

TABLE 1. (contd)

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 8.0E-8/py

PWR-2 = 5.8E-6/py

PWR-3 = 1.6E-5/py

PWR-4 = 9.3E-8/py

PWR-5 = 2.6E-7/py

PWR-6 = 7.1E-6/py

PWR-7 = 2.0E-5/py

(Note: In each affected release category, with Sequence V excluded from PWR-2, the contribution from the non-dominant accident sequences is scaled by the ratio of the sum of the affected dominant accident sequence frequencies to the sum of all the dominant accident sequence frequencies.)

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

$\bar{F}_{PWR}$ = 4.9E-5/py $\bar{F}_{BWR}$ = 2.2E-5/py[a]

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

W_{PWR} = 116 man-rem/py W_{BWR} = 140 man-rem/py[a]

10. Adjusted-Case Values for Affected Parameters:

B = C	=			0.0030
D = E	=			0.022
CONST1	=			2.0E-4
CONST2	=			5.8E-4
A1 = C1	=			0.0098
B1	=			0.035
HHMAN	=	HPMAN1	=	0.07
HPMAN	=			0.0105
HPRSCM	=	WXCM	=	0.0021
D•E	=			4.4E-4

(a) See Attachment 1.

TABLE 1. (contd)

10. Adjusted-Case Values for Affected Parameters (contd):

$B \cdot W = C \cdot X = 2.4E\text{-}5$

$D \cdot X = E \cdot W = 2.0E\text{-}4$

$B \cdot D = E \cdot C = 5.3E\text{-}5$

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Sequence	Release category	Frequency
T_2MLU -	γ (PWR-3) =	3.9E-7/py
	β (PWR-5) =	5.7E-9/py
	ε (PWR-7) =	3.9E-7/py
T_1MLU -	γ (PWR-3) =	6.4E-7/py
	β (PWR-5) =	9.3E-9/py
	ε (PWR-7) =	6.4E-7/py
$T_1(B_3)MLU$ -	γ (PWR-3) =	7.5E-7/py
	β (PWR-5) =	1.1E-8/py
	ε (PWR-7) =	7.5E-7/py
T_2MQH -	γ (PWR-3) =	2.4E-6/py
	β (PWR-5) =	3.5E-8/py
	ε (PWR-7) =	2.4E-6/py
S_3H -	γ (PWR-3) =	2.1E-6/py
	β (PWR-5) =	3.1E-8/py
	ε (PWR-7) =	2.1E-6/py
S_1D -	α (PWR-1) =	5.1E-8/py
	γ (PWR-3) =	1.0E-6/py
	β (PWR-5) =	3.7E-8/py
	ε (PWR-7) =	4.1E-6/py
T_2MQFH -	γ (PWR-2) =	1.7E-6/py
	β (PWR-4) =	2.5E-8/py
	ε (PWR-6) =	1.7E-6/py

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

$$S_3FH - \begin{cases} \gamma \text{ (PWR-2)} = & 1.4E\text{-}6/py \\ \beta \text{ (PWR-4)} = & 2.1E\text{-}8/py \\ \varepsilon \text{ (PWR-6)} = & 1.4E\text{-}6/py \end{cases}$$

$$S_2FH - \begin{cases} \alpha \text{ (PWR-1)} = & 8.6E\text{-}9/py \\ \beta \text{ (PWR-4)} = & 6.3E\text{-}9/py \\ \varepsilon \text{ (PWR-6)} = & 6.9E\text{-}7/py \end{cases}$$

$$T_2KMU - \begin{cases} \gamma \text{ (PWR-3)} = & 2.7E\text{-}6/py \\ \beta \text{ (PWR-5)} = & 4.0E\text{-}8/py \\ \varepsilon \text{ (PWR-7)} = & 2.7E\text{-}6/py \end{cases}$$

$$S_2D - \begin{cases} \alpha \text{ (PWR-1)} = & 6.8E\text{-}9/py \\ \gamma \text{ (PWR-3)} = & 1.4E\text{-}7/py \\ \beta \text{ (PWR-5)} = & 5.0E\text{-}9/py \\ \varepsilon \text{ (PWR-7)} = & 5.5E\text{-}7/py \end{cases}$$

$$S_3D - \begin{cases} \gamma \text{ (PWR-3)} = & 6.7E\text{-}7/py \\ \beta \text{ (PWR-5)} = & 9.8E\text{-}9/py \\ \varepsilon \text{ (PWR-7)} = & 6.7E\text{-}7/py \end{cases}$$

$$T_2MQD - \begin{cases} \gamma \text{ (PWR-3)} = & 7.2E\text{-}7/py \\ \beta \text{ (PWR-5)} = & 1.1E\text{-}8/py \\ \varepsilon \text{ (PWR-7)} = & 7.2E\text{-}7/py \end{cases}$$

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

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 7.3E-8/py

PWR-2 = 4.1E-6/py

PWR-3 = 1.2E-5/py

PWR-4 = 6.5E-8/py

PWR-5 = 2.0E-7/py

12. Affected Release Categories and Adjusted-Case Frequencies (contd):

PWR-6 = 5.0E-6/py

PWR-7 = 1.6E-5/py

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

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

$\bar{F}^* = 3.7E-5/py$

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

$W^*_{PWR} = 86$ man-rem/py

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

$(\Delta\bar{F})_{PWR} = 1.2E-5/py$ $(\Delta\bar{F})_{BWR} = 5.4E-6/py$[a]

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

$(\Delta W)_{PWR} = 30$ man-rem/py $(\Delta W)_{BWR} = 36$ man-rem/py[a]

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.2E+5	1.4E+7	0

(a) See Attachment 1.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Long-Term Upgrading of Training and Qualifications (Simulators) [I.A.2.6(1-3,5)]

2. Affected Plants (N):

All plants are assumed to be affected.

	N
PWRs	90
BWRs	44

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

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

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

PWR: (19,900 man-rem)(1.2E-5/py) = 0.24 man-rem/py

BWR: (19,900 man-rem)(5.4E-6/py) = 0.11 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
750	1.8E+4	0

6-12. Steps Related to Occupational Dose Increase for SIR Implementation and Operation/Maintenance:

These steps are omitted since the occupational doses for implementation and operation/maintenance are estimated to be zero. Thus, $D = D_0 = G = 0$.

ATTACHMENT 1

The RSSMAP studies for Oconee 3 and Grand Gulf 1 give total core-melt frequencies ($\bar{F}_o$) of 8.2E-5/py and 3.7E-5/py, respectively, for these plants. Using the original release category frequencies and the public dose factors (Appendix D of Andrews et. al 1983), one obtains total public risks (W_o) of 207 man-rem/py and 250 man-rem/py, respectively, for Oconee and Grand Gulf. For the purposes of scaling the base-case, affected core-melt frequency ($\bar{F}$) and public risk (W), and the reductions in the core-melt frequency ($\Delta\bar{F}$) and public risk (ΔW) from Oconee to Grand Gulf, the following are assumed:

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

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

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

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

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

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

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

3.0 SAFETY ISSUE COSTS

The resolution of this safety issue was assumed to be a major enhancement of the training and qualification programs. The programs would have to be upgraded in order to meet the requirements of INPO accreditation. These requirements are assumed to be far reaching and require significant effort on the part of utility training staffs. The amount of effort will vary among utilities, depending on the present state of their programs. The effort required to implement the program is estimated to require 10 to 20 person-years of effort for each plant. The mean value is expected to be shifted toward the lower end since many utilities are currently improving their training programs. A 12 person-year effort is taken as the central estimate.

Operation under the upgraded programs would require enhanced training activities and more operator time in training. The training staff is estimated to require three additional full-time people. It is assumed that the major cost of additional operator time can be estimated from increased time at simulators. It is estimated that 40 hours of simulator time will be added to operator training and requalification. For 20 operators per year passing through these programs, this is equivalent to 800 additional hours per year. It is further assumed that operators can be trained three at a time on the simulator and that simulator time can be acquired for $600/hour. This gives an additional simulator cost of $160,000/year.

The NRC effort to implement the resolution of this issue would be significant. NUREG-0660 estimates that 5.4 person-years plus $259,000 would be required. Some of these development activities have been completed. However, much work remains to be done. The remaining effort is estimated to be 4.5 person-years and $100,000. These activities are assumed to be equally divided between development and implementation.

The operational activities of the NRC would include reviews of training programs, increased inspections and additional examinations. The annual labor for reviews and inspections is estimated to be equivalent to 3 person-years. The principal addition in examinations is assumed to be NRC conduct of a portion of requalification examinations. It is assumed that the NRC will conduct 25 percent of the requalification examinations and that 20 operators are requalified at each plant every year. It is estimated that one person-month is required for each plant. This assumes that five (25 percent of 20) operators selected for NRC examination at each plant are tested at the same time.

Table 3 summarizes the results for analysis of the industry and NRC costs due to resolution of Issue I.A.2.6 (1-3,5).

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Long-Term Upgrading of Training and Qualifications (Simulators) [I.A.2.6(1-3,5)]

2. Affected Plants (N):

   All plants are assumed to be affected.

	N
PWRs	90
BWRs	44
Total	134

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

	$\bar{T}$(yr)
PWRs	28.8
BWRs	27.4
Avg. for all 134 plants	28.3

Industry Costs (Steps 4 through 12)

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

   PWR: (\$1.65E+9)(1.2E-5/py) = \$2.0E+4/py

   BWR: (\$1.65E+9)(5.4E-6/py) = \$8.9E+3/py

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

Best Estimate	Upper Bound	Lower Bound
\$6.2E+7	\$1.5E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

   12 person-yr/plant

   This applies to all plants.

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

   I = (12 person-yr/plant)(\$1.0E+5/person-yr) = \$1.2E+6/plant

TABLE 3. (contd)

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

NI = 134($1.2E+6/plant) = $1.6E+8

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

Training staff =	(3 people/py)(1 yr)	= 3.0 person-yr/py
Operator training =	(800 person-hr/py)(1 person-yr/1760 person-hr)	= 0.46 person-yr/py
		3.46 person-yr/py

Additional simulator time is estimated at (800 person-hr/py)/(3 person-hr/simulator-hr) = 267 simulator-hr/py

This applies to all plants.

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_O):

Labor =	(3.46 person-yr/py)($1.0E+5/person-yr)	= $3.46E+5/py
Simulator time =	(267 simulator-hr/py)($600/simulator-hr)	= $1.60E+5/py
	I_O	= $5.06E+5/py

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

$N\bar{T}I_O$ = 134 (28.3 yr)($5.06E+5/py) = $1.92E+9

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$2.1E+9	$3.0E+9	$1.1E+9

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Labor = 2.25 person-yr

Other funding = $5.0E+4

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

C_D = (2.25 person-yr)($1.0E+5/person-yr) + $5.0E+4 = $2.75E+5

TABLE 3. (contd)

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

Cost is estimated directly in Step 17.

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

Cost is estimated directly in Step 17.

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

$$NC = (2.25 \text{ person-yr})(\$1.0E{+}5/\text{person-yr}) + \$5.0E{+}4 = \$2.75E{+}5$$

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

Annual NRC labor for reviews and inspections is assumed to require 3 person-yrs. Over the entire nuclear industry, this breaks down to

$$(3 \text{ person/yr})/(134 \text{ plants}) = 0.022 \text{ person-yr/py}$$

To conduct requalification examinations, NRC will presumably expend 1 person-mo/py, or 0.083 person-yr/py. Thus, the total NRC labor becomes 0.105 person-yr/py.

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

$$C_o = (0.105 \text{ person-yr/py})(\$1.0E{+}5/\text{person-yr}) = \$1.05E{+}4/\text{py}$$

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

$$N\bar{T}C_o = 134(28.3 \text{ yr})(\$10{,}500/\text{py}) = \$3.98E{+}7$$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$4.0E+7	$6.0E+7	$2.0E+7

REFERENCES

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

U.S. NRC. 1980. NRC Action Plan Developed as a Result of the TMI-2 Accident. NUREG-0660, U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: I.A.2.6(4), Long-Term Upgrading of Training and Qualification (Training Workshops)

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This TMI action item seeks to improve communication among licensees and between licensees and the NRC through the use of workshops. Mandatory attendance of a representative from each operating shift once a year would help share operating experiences and provide regulators with greater insights.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	2.4E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	61

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	1.1
SIR Operation/Maintenance =	32
Total of Above =	33
Accident Avoidance =	5.0
NRC COSTS:	
SIR Development =	0.10
SIR Implementation Support =	0.20
SIR Operation/Maintenance Review =	8.5
Total of Above =	8.8

LONG-TERM UPGRADING OF TRAINING AND QUALIFICATION
(TRAINING WORKSHOPS)
ISSUE I.A.2.6(4)

1.0 SAFETY ISSUE DESCRIPTION

In the description of this safety issue in NUREG-D660 (NRC 1980), NRR is required to develop a commission paper on training workshops for licensed personnel. Reference is made to NUREG-0585 (NRC 1979), NRR, which is the source of information for this safety issue. This document clarifies that the intent of the issue is to conduct seminar-type workshops to exchange information between the NRC and licensed staff and among licensees on operations experience. This would assist in the improvement of operator performance and in improvements to reactor regulation, both resulting in improved safety. The proposed requirements would have one representative for each shift at each unit attend such a workshop annually. The focus is clearly put on reactor operators.

The Pacific Northwest Laboratory (PNL) has conducted and is conducting a series of these workshops for NRR. In the assessment of this issue, PNL staff responsible for these workshops were consulted. Their judgments form the basis of our analysis. This analysis assumes that the major gains in reactor safety will come through the improvement in operator performance, that is, a reduction in their error rates. There is also a pathway, through improved regulations developed from operator input at the workshops, to improve safety by means other than human performance. These would be extremely difficult to quantify. Therefore, only the human error rate reduction pathway to improved safety will be treated.

The PNL staff felt that the workshops have a definite potential for improving safety. However, they also saw significant temptations to employ the workshops for other purposes. An optimal use of the workshops would be to share operational experiences among the facilities, perhaps walking through a recent transient. What appears to be a more likely course is use of the workshops to gather information and insights from the licensees for use by the NRC. While valuable, such efforts dilute the direct effect upon the reactor operators.

2.0 SAFETY ISSUE RISK AND DOSE

As stated previously, there are two potential pathways to improved safety for this issue. These are improved operator performance through the sharing of safety-relevant experiences and the effect of improved regulation arising from interaction between the operators and the NRC attending the workshops.

The second pathway would be a second-order effect and very difficult to quantify. Therefore, it was assumed that all benefit would be derived through the reduction in operator-error rates.

A panel of PNL experts was assembled, including staff conducting operator licensing examinations; staff with experience in reactor operations, reactor safety, and risk assessment; and the staff responsible for the current operator feedback workshops. This panel produced the estimates that form the basis of this analysis.

PUBLIC RISK REDUCTION

The PNL panel estimated that the most likely reduction in human error rates for operators due to the conduct of the proposed workshops would be 3 percent. This is assuming that the workshops are conducted in the manner now perceived, that is, to focus on data gathering for the NRC. This reduces the amount of time that could be devoted to inter-licensee sharing of operational experiences, which would have a more direct effect on safety-related operational performance in the plants. In bounding the potential error reduction, the panel estimated that the possible reduction from workshops ranged from 1 percent to 10 percent. If the focus could be shifted toward the inter-licensee exchange of operational experiences, the most likely reduction in error rate would shift upward. However, it is not expected to exceed 10 percent. The details of the public risk reduction estimate are worked out in Table 1.

OCCUPATIONAL DOSE

Since these issues deal with offsite conduct of workshops, there is no occupational dose associated with either implementation or operation of the change. Thus, only the occupational dose reduction due to accident avoidance is estimated in Table 2.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Long-Term Upgrading of Training and Qualification (Training Workshops) [I.A.2.6(4)]

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

All plants are assumed to be affected.

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

TABLE 1. (contd)

3. Plants Selected for Analysis:

Oconee 3 - representative PWR

Grand Gulf 1 - representative BWR

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

4. Parameters Affected by SIR:

Oconee: B, C, D, E, CONST1, CONST2, A1, B1, C1, HHMAN, HPMAN, HPMAN1, HPRSCM, WXCM, D•E, B•W, C•X, D•X, E•W, B•D, E•C.

5. Base-Case Values for Affected Parameters:

Original values from Appendix A are assumed (Andrews et al. 1983).

6. Affected Accident Sequences and Base-Case Frequencies:

Sequence		Frequency (1/py)
T_2MLU -	γ (PWR-3)	5.8E-7
	β (PWR-5)	8.5E-9
	ε (PWR-7)	5.8E-7
T_1MLU -	γ (PWR-3)	9.8E-7
	β (PWR-5)	1.4E-8
	ε (PWR-7)	9.8E-7
$T_1(B_3)MLU$ -	γ (PWR-3)	1.1E-6
	β (PWR-5)	1.6E-8
	ε (PWR-7)	1.1E-6
T_2MQH -	γ (PWR-3)	3.2E-6
	β (PWR-5)	4.7E-8
	ε (PWR-7)	3.2E-6
S_3H -	γ (PWR-3)	2.8E-6
	β (PWR-5)	4.1E-8
	ε (PWR-7)	2.8E-6

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

Sequence		Frequency (1/py)
S_1D -	α (PWR-1)	5.3E-8
	γ (PWR-3)	1.1E-6
	β (PWR-5)	3.9E-8
	ε (PWR-7)	4.3E-6
T_2MQFH -	γ (PWR-2)	2.4E-6
	β (PWR-4)	3.6E-8
	ε (PWR-6)	2.4E-6
S_3FH -	γ (PWR-2)	2.0E-6
	β (PWR-4)	3.0E-8
	ε (PWR-6)	2.0E-6
S_2FH -	α (PWR-1)	1.2E-8
	β (PWR-4)	8.9E-9
	ε (PWR-6)	9.8E-7
T_2KMU -	γ (PWR-3)	3.9E-6
	β (PWR-5)	5.7E-8
	ε (PWR-7)	3.9E-6
S_2D -	α (PWR-1)	7.2E-9
	γ (PWR-3)	1.4E-7
	β (PWR-5)	5.2E-9
	ε (PWR-7)	5.7E-7
S_3D -	γ (PWR-3)	6.7E-7
	β (PWR-5)	9.8E-9
	ε (PWR-7)	6.7E-7

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

Sequence	Frequency (1/py)
T_2MQD - γ (PWR-3)	7.2E-7
T_2MQD - β (PWR-5)	1.1E-8
T_2MQD - ε (PWR-7)	7.2E-7

(Note: In each affected accident sequence, the contribution from the non-dominant minimal cut sets is scaled by the ratio of the sum of the affected dominant minimal cut set frequencies to the sum of all the dominant minimal cut set frequencies.)

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 8.0E-8/py

PWR-2 = 5.8E-6/py

PWR-3 = 1.6E-5/py

PWR-4 = 9.3E-8/py

PWR-5 = 2.6E-7/py

PWR-6 = 7.1E-6/py

PWR-7 = 2.0E-5/py

(Note: In each affected release category, with Sequence V excluded from PWR-2, the contribution from the non-dominant accident sequences is scaled by the ratio of the sum of the affected dominant accident sequence frequencies to the sum of all the dominant accident sequence frequencies.)

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

$\bar{F}_{PWR}$ = 4.906E-5/py $\bar{F}_{BWR}$ = 2.2E-5/py[(a)]

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

W_{PWR} = 116.3 man-rem/py W_{BWR} = 140 man-rem/py[(a)]

(a) See Attachment 1.

TABLE 1. (contd)

10. Adjusted-Case, Affected Values for Affected Parameters:

All affected parameters are assumed to experience a 3% decrease in their failure probabilities as a result of the SIR. However, this decrease is evident to two significant figures only for the following parameters:

HHMAN = HPMAN1 = 0.09

HPRSCM = WXCM = 0.0029

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Sequence		Frequency (1/py)
T_2MQH -	γ (PWR-3)	3.1E-6
	β (PWR-5)	4.6E-8
	ε (PWR-7)	3.1E-6
S_3H -	γ (PWR-3)	2.8E-6
	β (PWR-5)	4.0E-8
	ε (PWR-7)	2.8E-6
T_2MQFH -	γ (PWR-2)	2.4E-6
	β (PWR-4)	3.4E-8
	ε (PWR-6)	2.4E-6
S_3FH -	γ (PWR-2)	2.0E-6
	β (PWR-4)	2.9E-8
	ε (PWR-6)	2.0E-6
S_2FH -	α (PWR-1)	1.2E-8
	β (PWR-4)	8.6E-9
	ε (PWR-6)	9.5E-7
T_2KMU -	γ (PWR-3)	3.8E-6
	β (PWR-5)	5.5E-8
	ε (PWR-7)	3.8E-6

(Note: Only affected accident sequences containing HHMAN, HPMAN1, HPRSCM or WXCM exhibit a change in frequency, from the base to the adjusted case, to two significant figures, and are shown here. The contributions from the non-dominant minimal cut sets are assumed to decrease in the same proportions as those from the dominant minimal cut sets in all affected accident sequences.)

TABLE 1. (contd)

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 8.0E-8/py

PWR-2 = 5.7E-6/py

PWR-3 = 1.5E-5/py

PWR-4 = 8.9E-8/py

PWR-5 = 2.6E-7/py

PWR-6 = 6.9E-6/py

PWR-7 = 2.0E-5/py

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

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

$\bar{F}^*_{PWR}$ = 4.810E-5/py

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

W^*_{PWR} = 110.4 man-rem/py

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

$(\Delta\bar{F})_{PWR}$ = 9.7E-7/py $(\Delta\bar{F})_{BWR}$ = 4.4E-7/py[(a)]

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

$(\Delta W)_{PWR}$ = 5.9 man-rem/py $(\Delta W)_{BWR}$ = 7.1 man-rem/py[(a)]

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.4E+4	1.4E+7	0

(a) See Attachment 1.

ATTACHMENT 1

The RSSMAP studies for Oconee 3 and Grand Gulf 1 give total core-melt frequencies ($\bar{F}_o$) of 8.2E-5/py and 3.7E-5/py, respectively, for these plants. Using the original release category frequencies and the public dose factors (Appendix D of Andrews et al. 1983), one obtains total public risks (W_o) of 207 man-rem/py and 250 man-rem/py, respectively, for Oconee and Grand Gulf. For the purposes of scaling the base-case, affected core-melt frequency ($\bar{F}$) and public risk (W), and the reductions in the core-melt frequency ($\Delta\bar{F}$) and public risk (ΔW) from Oconee to Grand Gulf, the following are assumed:

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

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

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

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

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

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

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

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Long-Term Upgrading of Training and Qualification (Training Workshops) [I.A.2.6(4)]

2. Affected Plants (N):

   All plants are assumed to be affected.

	N
PWRs	90
BWRs	44

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

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

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

   PWR: (19,900 man-rem)(9.7E-7/py) = 1.9E-2 man-rem/py

   BWR: (19,900 man-rem)(4.4E-7/py) = 8.8E-3 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
61	1.8E+4	0

6-12. Steps Related to Occupational Dose Increase for SIR Implementation and Operation/Maintenance:

   These steps are omitted since the occupational doses for implementation and operation/maintenance are estimated to be zero. Thus, $D = D_O = G = 0$.

3.0 SAFETY ISSUE COSTS

The PNL panel also estimated the costs associated with the training work shops. The development of those costs is summarized in Table 3 and detailed below.

The industry resources required for implementation are estimated to be one person-month per plant. This is the estimated personnel requirement associated with the trial workshops currently being conducted. It includes utility staff time for attendance of the workshop, preparation by staff and management, and staff time dedicated to the dissemination of insights gained at the workshop. At a cost of $1.0E+5/person-year (see Appendix E of Andrews et al. 1983), this yields a per-plant cost of $8300. Across the industry (i.e., 134 plants), this amounts to $1.1E+6.

The industry resources required annually to participate in the training workshops are estimated to be the same as those for implementation, that is, one person-month per plant. This includes workshop attendance, preparation before the workshop, and dissemination of information afterward. This would be equivalent to $8300/plant-year. For the total industry (134 plants), this works out to an estimated 134 person-months per year or $1.1E+6 per year. Given the average remaining lifetime for the plants, this gives a total operational cost of $3.2E+7.

The total cost to the NRC to develop and implement the resolution of this issue was estimated to be $3.0E+5. This includes NRC staff labor and services of a contractor. Since the development activities of the NRC staff and the contractor are to some degree interchangeable, no attempt was made to provide separate estimates. It is assumed that, of the $3.0E+5 for development and implementation, $1.0E+5 can be charged to development.

The annual cost to the NRC was also estimated to be $3.0E+5. Again, this was assumed to contain some mixture of staff and contractor expenses. Over the average remaining life, the operational cost comes to $8.5E+6.

While not specific, these estimates for implementation and operation are firmly based in the experience of conducting the present trial workshops.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Long-Term Upgrading of Training and Qualifications (Training Workshops) [I.A.2.6(4)]

TABLE 3. (contd)

2. Affected Plants (N):

All plants are assumed to be affected.

	N
PWRs	90
BWRs	44
	134

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

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

Industry Costs (Steps 4 through 12)

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

PWR: (\$1.65E+9)(9.7E-7/py) = \$1.6E+3/py

BWR: (\$1.65E+9)(4.4E-7/py) = \$7.3E+2/py

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

Best Estimate	Upper Bound	Lower Bound
\$5.0E+6	\$1.5E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

1 person-month/plant = 0.083 person-yr/plant

This applies to all plants.

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

I = \$8300/plant

TABLE 3. (contd)

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

$NI = \$1.1E+6$

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

1 person-month/py = 0.083 person-yr/py

This applies to all plants.

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_O):

$I_O = \$8300/py$

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

$N\bar{T}I_O = \$3.2E+7$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$3.5E+7	\$4.9E+7	\$1.7E+7

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Cost is estimated directly in next step.

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

$C_D = \$1.0E+5$

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

Cost is estimated directly in Step 17.

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

Cost is estimated directly in Step 17.

TABLE 3. (contd)

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

$NC = \$2.0E+5$

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

Cost is estimated directly in Step 20.

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

Cost is estimated directly in Step 20.

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

$N\bar{T}C_o = (\$3.0E+5/yr)(28.3\ yr) = \$8.5E+6$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$8.8E+6	\$1.3E+7	\$4.4E+6

REFERENCES

Andrews, W. B. et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

U.S. NRC. 1979. TMI-2 Lessons Learned Task Force: Final Report. NUREG-0585, U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. NRC. 1980. NRC Action Plan Developed as a Result of the TMI-2 Accident. NUREG-0660, U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: I.A.2.6(6), Long-Term Upgrading of Training and Qualifications (Nuclear Power Fundamentals for Operator Training)

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This TMI action item calls for NRR to establish definitive instructional requirements for the inclusion of nuclear power fundamentals within the reactor operator training courses. This Safety Issue Resolution (SIR) of the issue was felt to have no measurable public safety benefit. The training in nuclear fundamentals has already been improved over the pre-TMI training, and further improvements are not likely to produce measureable changes in operator performance.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	0
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	0

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	38
SIR Operation/Maintenance =	1100
Total of Above =	1100
Accident Avoidance =	0
NRC COSTS:	
SIR Development	0
SIR Implementation Support =	0.04
SIR Operation/Maintenance Review =	0
Total of Above =	0.04

LONG-TERM UPGRADING OF TRAINING AND QUALIFICATIONS (NUCLEAR POWER FUNDAMENTALS FOR OPERATOR TRAINING): ISSUE I.A.2.6(6)

1.0 SAFETY ISSUE DESCRIPTION

The TMI action item I.A.2.6(6) as described in NUREG-0660 (NRC 1980) calls for NRR to develop requirements for the inclusion of nuclear power fundamentals within the instruction given to reactor operators. This arose out of a concern expressed in NUREG-0585 (NRC 1979) that the twelve weeks of fundamentals training given to operators at that time was insufficient.

In order to assess this safety issue, a panel of experts was assembled from the Pacific Northwest Laboratory (PNL) staff. This panel was comprised of members experienced in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas.

The panel felt there had been significant progress across the industry in the area of instruction in nuclear power fundamentals since the time of NUREG-0585 (NRC 1979). Further increase in emphasis on fundamentals was felt to be unlikely to improve operator performance. The current trend in operator licensing examinations is to stress the view that further fundamentals training would not add to plant safety.

The Issue Summary Work Sheet presented on the cover page provides a summary of the analysis of the safety issue. The details of the analysis are described further in the following sections.

2.0 SAFETY ISSUE RISK AND DOSE

The benefit from the safety issue resolution (SIR) is normally described by two terms. These are the reduction in public risk and the reduction in occupational dose. For this safety issue, neither term is felt to have a measureable improvement. Additional explanation is given in subsequent subsections.

PUBLIC RISK REDUCTION

Safety issues which deal with operator training can affect the public risk by improvements in the operator safety-related performance. This can lead to a reduction in core-melt frequency and a reduced probabilistic risk. For this safety issue, the PNL panel felt that the current level of instruction in nuclear power fundamentals was adequate. Further emphasis of fundamentals was viewed as not likely to improve operator safety performance, and, therefore, there would be no measurable public risk reduction associated with the implementation of this issue. Thus, the Public Risk Reduction Work Sheet has been omitted.

OCCUPATIONAL DOSE

The PNL panel saw no reduction in occupational dose associated with the implementation of the safety issue. Therefore, the Occupational Dose Work Sheet has been omitted.

3.0 SAFETY ISSUE COSTS

The PNL panel also estimated the costs, to industry and the NRC, associated with the safety issue.

It was assumed that, if implemented, the additional nuclear power fundamentals training would add 4 weeks to the training period. Also, it was assumed that 20 operators complete the training course each year at every plant. In addition, one full-time instructor was assumed to be required. This yields 80 person-weeks for the operators, 44 person-weeks for the instructors, or 124 person-weeks overall, per plant, each year. To implement this practice, an effort equivalent to 124 person-weeks per calendar year was estimated to be required.

It is assumed there are no NRC development costs associated with this issue. The costs to NRC to implement the resolution are taken from the NUREG-0660 estimate of 0.4 person-years, or approximately 18 person-weeks. No added costs are estimated for operation for the NRC. The review of the additional instruction could be contained in the current routine function, thereby causing no added expense.

TABLE 1. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

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

2. Affected Plants (N):

    All plants are assumed to be affected.

	N
PWR	90
BWR	44
	134

TABLE 1. (contd)

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

	$\bar{T}$ yr
PWR	28.8
BWR	27.4
All	28.3

Industry Costs (Steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, $\Delta(\bar{FA})$:

   0

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

Best Estimate	Upper Bound	Lower Bound
0	0	0

6. Per-Plant Industry Resources for SIR Implementation:

   124 person-wk/plant

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

   I = \$2.8E+5/plant

   This applies to all plants.

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

   NI = \$3.8E+7

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

   124 person-wk/ry

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_o):

    I_o = \$2.8E+5/ry

    This applies to all plants.

TABLE 1. (contd)

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

$N\bar{T}I_O$ = \$1.1E+9

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$1.1E+9	\$1.6E+9	\$6.0E+8

NRC Costs (Steps 12 through 21)

13. NRC Resources for SIR Development:

0

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

$C_D = 0$

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

(18 person-wk)/(134 plants) = 0.13 person-wk/plant

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

C = \$300

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

NC = \$4.1E4

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

0

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

0

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

0

TABLE 1. (contd)

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$4.1E+4	$6.2E+4	$2.1E+4

REFERENCES

U.S. NRC. 1979. TMI-2 Lessons Learned Task Force: Final Report. NUREG-0585, U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. NRC. 1980. NRC Action Plan Developed as a Result of the Recent TMI-2 Accident. NUREG-0660, U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: I.A.3.3, Requirements for Operator Fitness

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This TMI action items seeks 1) to assure that applicants for operator and senior operator licenses are psychologically fit, and 2) to prohibit licensing of persons with histories of drug and alcohol abuse or criminal backgrounds.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	2.4E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	54

COST RESULTS ($$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	47
SIR Operation/Maintenance =	1100
Total of Above =	1200
Accident Avoidance =	4.5
NRC COSTS:	
SIR Development =	0.15
SIR Implementation Support =	0.15
SIR Operation/Maintenance Review =	8.6
Total of Above =	8.9

REQUIREMENTS FOR OPERATOR FITNESS

ISSUE I.A.3.3

1.0 SAFETY ISSUE DESCRIPTION

This safety issue as described in NUREG-0660 (NRC 1980) calls for the NRC to develop a regulatory approach, 1) to provide assurance that applicants for operator and senior operator licenses are psychologically fit, and 2) to prohibit licensing of persons with histories of drug and alcohol abuse or criminal backgrounds. The regulations will be applied to all current and future operating power plants.

This issue has two components, the first of which deals with alcohol and drug abuse problems among operators and senior operators. A proposed rule dealing with this problem was issued on August 5, 1982. Mr. Merschoff, who helped to write the rule at NRC, commented that the impact on most utilities would be minimal because they are already meeting the proposed guidelines. The rule would codify and standardize the treatment of drug and alcohol problems and hopefully prevent these problems from worsening at nuclear power plants.

The second component of this safety issue deals with limiting access of psychologically unstable individuals to vital plant areas. Mr. Prell of the NRC was contacted to obtain the details of a proposed program to address the safety issue. This program is comprehensive in that it is aimed at limiting the access to vital plant areas of disgruntled employees, unsuitable employees, and personnel under the influence of drugs or alcohol. Thus, it would include the proposed rule issued on August 5, 1982, as part of the program to be enforced at each plant.

The program has the following three parts: 1) background search, 2) psychological assessment, and 3) behavior observation. The first two parts would occur at the time of employment, and the last would be an ongoing activity. The background check would include examination of an individual's past for unstable activities, a criminal record, credit problems, and previous employment problems. According to Prell, that data on psychological screening show that 2 to 3 percent of white-collar people are identified as unstable; for blue-collar employees, the proportion is 7-10 percent.

To assess this safety issue, a number of engineers at the Pacific Northwest Laboratory (PNL) were consulted. These engineers have expertise in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas.

The problems addressed by the safety issue are society-wide; their importance at nuclear power plants is unclear. Obviously, significant damage could result if impaired personnel were performing critical safety operations. However, legal and institutional problems may limit a thorough implementation of the proposed program. If an adequate program were implemented at all power plants and integrated into overall plant operations, the new program would reduce operator error, which in turn would lower the risk associated with operation of the power plant. Thus, this safety issue resolution (SIR) assumes the implementation of the access authorization system at all 134 plants--63 under construction and 71 already in operation.

2.0 SAFETY ISSUE RISK AND DOSE

The analyses of public risk reduction and occupational dose are discussed below. Results are summarized in Tables 1 and 2, respectively.

For some utilities, the new system may result in a modest but significant reduction in operator error during an emergency, whereas in others the system may have no discernible effect. This SIR was assumed to reduce operator error probabilities by an average of about 2 percent at all currently operating and future plants.

Neither the implementation, operation, or maintenance of this SIR would involve any changes in occupational dose accrued by personnel.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Requirements for Operator Fitness (I.A.3.3)

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

   All 134 plants are assumed to be affected.

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

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf 1 - representative BWR

   (This analysis is conducted for Oconee 3, and the results are scaled for Grand Gulf 1, as discussed in Attachment 1.)

4. Parameters Affected by SIR:

   Oconee: B, C, D, E, CONST1, CONST2, A1, C1, B1, HHMAN, HPMAN1, HPMAN, HPRSCM, WXCM, D•E, B•W, C•X, D•X, E•W, B•D, E•C

TABLE 1. (contd)

5. Base-Case Values for Affected Parameters:

Original values from Appendix A are assumed (Andrews et al. 1983).

6. Affected Accident Sequences and Base-Case Frequencies:

Sequence	Release Category		Frequency
T_2MLU -	γ (PWR-3)	=	5.8E-7
	β (PWR-5)	=	8.5E-9
	ε (PWR-7)	=	5.8E-7
T_1MLU -	γ (PWR-3)	=	9.8E-7
	β (PWR-5)	=	1.4E-8
	ε (PWR-7)	=	9.8E-7
$T_1(B_3)MLU$ -	γ (PWR-3)	=	1.1E-6
	β (PWR-5)	=	1.6E-8
	ε (PWR-7)	=	1.1E-6
T_2MQH -	γ (PWR-3)	=	3.2E-6
	β (PWR-5)	=	4.7E-8
	ε (PWR-7)	=	3.2E-6
S_3H -	γ (PWR-3)	=	2.8E-6
	β (PWR-5)	=	4.1E-8
	ε (PWR-7)	=	2.8E-6
S_1D -	α (PWR-1)	=	5.3E-8
	γ (PWR-3)	=	1.1E-6
	β (PWR-5)	=	3.9E-8
	ε (PWR-7)	=	4.3E-6
T_2MQFH -	γ (PWR-2)	=	2.4E-6
	β (PWR-4)	=	3.6E-8
	ε (PWR-6)	=	2.4E-6

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

	γ (PWR-2)	=	2.0E-6
S_3FH -	β (PWR-4)	=	3.0E-8
	ε (PWR-6)	=	2.0E-6
	α (PWR-1)	=	1.2E-8
S_2FH -	β (PWR-4)	=	8.9E-9
	ε (PWR-6)	=	9.8E-7
	γ (PWR-3)	=	3.9E-6
T_2KMU -	β (PWR-5)	=	5.7E-8
	ε (PWR-7)	=	3.9E-6
	α (PWR-1)	=	7.2E-9
S_2D -	γ (PWR-3)	=	1.4E-7
	β (PWR-5)	=	5.2E-9
	ε (PWR-7)	=	5.7E-7
	γ (PWR-3)	=	6.7E-7
S_3D -	β (PWR-5)	=	9.8E-9
	ε (PWR-7)	=	6.7E-7
	γ (PWR-3)	=	7.2E-7
T_2MQD -	β (PWR-5)	=	1.1E-8
	ε (PWR-7)	=	7.2E-7

(Note: In each affected accident sequence, the contribution from the non-dominant minimal cut sets is scaled by the ratio of the sum of the affected dominant minimal cut set frequencies to the sum of all the dominant minimal cut set frequencies.)

TABLE 1. (contd)

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 8.0E-8/py

PWR-2 = 5.8E-6/py

PWR-3 = 1.6E-5/py

PWR-4 = 9.3E-8/py

PWR-5 = 2.6E-7/py

PWR-6 = 7.1E-6/py

PWR-7 = 2.0E-5/py

(Note: In each affected release category, with Sequence V excluded from PWR-2, the contribution from the non-dominant accident sequences is scaled by the ratio of the sum of the affected dominant accident sequence frequencies to the sum of all the dominant accident sequence frequencies.)

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

$\bar{F}_{PWR}$ = 4.906E-5/py $\qquad$ $\bar{F}_{BWR}$ = 2.2E-5/py[a]

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

W_{PWR} = 116.39 man-rem/py $\qquad$ W_{BWR} = 140 man-rem/py[a]

10. Adjusted-Case Values for Affected Parameters:

All affected parameters are assumed to experience a 2% decrease in their failure probabilities as a result of SIR. However, this decrease is evident to two significant figures only for the following parameters:

HHMAN = HPMAN1 = 0.098
HPRSCM = WXCM = 0.0029

11. Affected Accident Sequences and Adjusted-Case Frequencies:

$$T_2MQH - \begin{cases} \gamma(\text{PWR-3}) = 3.1\text{E-6} \\ \beta(\text{PWR-5}) = 4.6\text{E-8} \\ \varepsilon(\text{PWR-7}) = 3.1\text{E-6} \end{cases}$$

$$S_3H - \begin{cases} \gamma(\text{PWR-3}) = 2.8\text{E-6} \\ \beta(\text{PWR-5}) = 4.0\text{E-8} \\ \varepsilon(\text{PWR-7}) = 2.8\text{E-6} \end{cases}$$

(a) See Attachment 1.

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

$$T_2MQFH - \begin{cases} \gamma(PWR\text{-}2) = 2.4E\text{-}6 \\ \beta(PWR\text{-}4) = 3.4E\text{-}8 \\ \varepsilon(PWR\text{-}6) = 2.4E\text{-}6 \end{cases}$$

$$S_3FH - \begin{cases} \gamma(PWR\text{-}2) = 2.0E\text{-}6 \\ \beta(PWR\text{-}4) = 2.9E\text{-}8 \\ \varepsilon(PWR\text{-}6) = 2.0E\text{-}6 \end{cases}$$

$$S_2FH - \begin{cases} \alpha(PWR\text{-}1) = 1.2E\text{-}8 \\ \beta(PWR\text{-}4) = 8.6E\text{-}9 \\ \varepsilon(PWR\text{-}6) = 9.5E\text{-}7 \end{cases}$$

$$T_2KMU - \begin{cases} \gamma(PWR\text{-}3) = 3.8E\text{-}6 \\ \beta(PWR\text{-}5) = 5.6E\text{-}8 \\ \varepsilon(PWR\text{-}7) = 3.8E\text{-}6 \end{cases}$$

(Note: Only affected accident sequences containing HHMAN, HPMAN1, HPRSCM or WXCM exhibit a change in frequency, from the base to the adjusted case, to two significant figures, and are shown here. The contributions from the non-dominant minimal cut sets are assumed to decrease in the same proportions as those from the dominant minimal cut sets in all affected accident sequences.)

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 8.0E-8/py

PWR-2 = 5.7E-6/py

PWR-3 = 1.5E-5/py

PWR-4 = 8.9E-8/py

PWR-5 = 2.6E-7/py

PWR-6 = 6.9E-6/py

PWR-7 = 2.0E-5/py

TABLE 1. (contd)

12. Affected Release Categories and Adjusted-Case Frequencies (contd):

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

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

$\bar{F}^* = 4.819E\text{-}5/py$

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

$W^*_{PWR} = 110.4$ man-rem/py

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

$(\Delta\bar{F})_{PWR} = 8.7E\text{-}7/py$ $(\Delta\bar{F})_{BWR} = 3.9E\text{-}7/py$[(a)]

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

$(\Delta W)_{PWR} = 5.9$ man-rem/py $(\Delta W)_{BWR} = 7.1$ man-rem/py[(a)]

17. Total Public Risk Reduction, $(\Delta W)_{Total}$

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.4E+4	1.4E+7	0

(a) See Attachment 1.

ATTACHMENT 1

The RSSMAP studies for Oconee 3 and Grand Gulf 1 give total core-melt frequencies ($\bar{F}_o$) of 8.2E-5/py and 3.7E-5/py, respectively, for these plants (Andrews et al. 1983). Using the original release category frequencies and the public dose factors (Appendix D of Andrews et al. 1983), one obtains total public risks (W_o) of 207 man-rem/py and 250 man-rem/py, respectively, for Oconee and Grand Gulf. For the purpose of scaling the base-case, affected core-melt frequency ($\bar{F}$) and public risk (W), and the reductions in the core-melt frequency ($\Delta\bar{F}$) and public risk (ΔW) from Oconee to Grand Gulf, the following are assumed:

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

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

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

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

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

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

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

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Requirements for Operator Fitness (I.A.3.3)

2. Affected Plants (N):

   All currently operating or future plants (134)

	N
PWRs	90
BWRs	44

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

	$\bar{T}$ (yr)
PWR	28.8
BWR	27.4

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

   PWR = (19,900 man-rem)(8.7E-7/py) = 0.017 man-rem/py

   BWR = (19,900 man-rem)(3.9E-7/py) = 0.0078 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
54	1.8E+4	0

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

No change in occupational dose is anticipated for SIR implementation, operation, or maintenance. Thus, $D = D_0 = G = 0$.

3.0 SAFETY ISSUE COSTS

A value/impact analysis has been prepared by the NRC, and cost estimates for industry were developed. The Atomic Industrial Forum has seen these costs and concurs with them. For existing plants, implementation cost is $140,000 per plant. This cost includes the preparation of the plant and associated procedures ($33,000), licensee management and clerical staff ($63,000), training to implement the behavioral observation program ($34,000), and storage for files ($10,000). For future plants, implementation costs were estimated to be \$590,000 per plant. In addition to the costs noted above for existing plants, this includes the cost of background investigations (\$375,000), review process and appeals procedures (\$36,000), increased file storage requirements (\$30,000), and miscellaneous criminal checks with the FBI, etc. (\$9,000). The cost of operation of the access authorization system at each plant was estimated to be ~\$300,000 per year. This operating cost includes background investigations for new people as a result of employee turnover (\$94,000), professional management and clerical staff (\$63,000), review and appeal process (\$67,000), refresher training for old supervisors (\$19,000), training of new supervisors (\$9,000), plan maintenance and updates (\$8,000), file storage (\$39,000), and criminal history checks with the FBI for new people (\$2,000).

Included in the industry and NRC costs are the following components: NRC development and issuance of a proposed rule, utility review of the rule, utility preparation of a plan for each plant, NRC review of the plan and resolution of any problems before the operation of the plan.

The NRC labor for further development and issuance of the proposed plan is estimated to be 1.5 man-years. For implementation of the plan, which includes the review and modification of the utilities' plans, the NRC effort was estimated at 1.5 man-years. The NRC labor needed for the operation of this SIR was estimated to be 1 man-week/plant-year. This would involve a yearly inspection by the NRC of each plant's access authorization system.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Requirements for Operator Fitness (I.A.3.3)

2. Affected Plants (N):

   All currently operating or future plants (134).

		N
PWRs:	planned	43
	operating	47
BWRs:	planned	20
	operating	24

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

	$\bar{T}$ (yr)
PWR	28.8
BWR	27.4
All	28.3

TABLE 3. (contd)

Industry Costs (Steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, $\Delta(\bar{FA})$:

$PWR = (\$1.65E+9)(8.7E-7/py) = \$1400/py$

$BWR = (\$1.65E+9)(3.9E-7/py) = \$640/py$

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

Best Estimate	Upper Bound	Lower Bound
\$4.5E+6	\$1.5E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

Cost is estimated directly in next step.

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

For existing plants, I = \$140,00D/plant

For future plants, I = \$590,000/plant

(Same for both PWRs and BWRs.)

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

$NI = 71(\$1.4E+5/plant) + 63(\$5.9E+5/plant) = \$4.7E+7$

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

Cost is estimated directly in next step.

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

$I_0 = \$3.0E+5/py$

(Same for both PWRs and BWRs.)

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

$N\bar{T}I_0 = 134(28.3\ yr)(\$3.0E+5/py) = \$1.14E+9$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$1.2E+9	\$1.8E+9	\$6.1E+8

TABLE 3. (contd)

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

1.5 man-yr

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

(1.5 man-yr)($1.0E+5/man-yr) = $1.5E+5

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

Cost is estimated directly in Step 17.

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

Cost is estimated directly in Step 17.

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

NC = (1.5 man-yr)($1.0E+5/man-yr) = $1.5E+5

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

1 man-wk/plant-yr for both PWRs and BWRs.

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

C_o = (1 man-wk/py)($2270/man-wk) = $2270/py

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

$N\bar{T}C_o$ = 134(28.3 yr)($2270/py) = $8.61E+6

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$8.9E+6	$1.3E+7	$4.6E+6

REFERENCES

Andrews, W.B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

U.S. NRC. 1980. NRC Action Plan Developed as a Result of the TMI-2 Accident. NUREG-0660, U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: I.A.3.4, Licensing of Additional Operations Personnel

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This TMI action item seeks to upgrade the operations safety performance in nuclear power plants by considering licensing requirements for operations personnel in addition to reactor operators and senior operators. By undergoing licensing, such personnel as managers, engineers, and technicians would be better qualified and less likely to commit errors in the performance of their safety-related functions. The assumed safety issue resolution is the licensing of the majority of this personnel.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	2.6E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	131

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	35
SIR Operation/Maintenance =	190
Total of Above =	230
Accident Avoidance =	11
NRC COSTS:	
SIR Development =	30.
SIR Implementation Support =	5.0
SIR Operation/Maintenance Review =	170
Total of Above =	200

LICENSING OF ADDITIONAL OPERATIONS PERSONNEL

ISSUE I.A.3.4

1.0 SAFETY ISSUE DESCRIPTION

The description of Task I.A.3.4 given in NUREG-0660 (NRC 1980) calls for the Office of Nuclear Reactor Regulation to continue to study which operations personnel, in addition to the currently licensed reactor operators and senior reactor operators, should be licensed by the NRC. The operations positions specifically identified in NUREG-0660 are managers, engineers, auxiliary operators, maintenance personnel, technicians and shift technical advisors. The objective of such licensing would be to ensure that only properly qualified and trained individuals are employed in these operations positions.

To assess this safety issue, PNL assembled a panel of experts with considerable experience in the areas of reactor operator licensing, reactor operations, utility field work, and reactor vendor experience, as well as experience in general reactor safety areas. The assessment was based on the assumption that an effort to license the majority of the operations personnel would constitute the resolution of this issue. Furthermore, the licensing was assumed to be position specific, that is, a plant supervisor would undergo a plant supervisor examination in order to obtain a plant supervisor license.

The PNL panel felt that the effects of this safety issue resolution (SIR) would be minimal, since existing practices already bring qualified and trained individuals into responsible positions. Auxiliary operators, for example, are already qualified by training and perform their safety-related activities under the supervision of licensed operators. Maintenance personnel and technicians already have the benefit of apprenticeship programs.

The most significant effect of an expanded personnel licensing program is felt to be a screening effect, which would form an additional boundary to prevent poor performers from entering the operations staff.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reductioo and the occupational dose associated with this SIR are analyzed in this section. The results of the analyses are summarized in Tables 1 and 2, respectively.

The public risk reduction associated with this SIR results from a reduction in core-melt frequency caused by the improved safety-related performance of operations personnel. To relate the improved performance to core-melt frequency, those operations personnel whose errors are displayed in the reference RSSMAP studies were identified: reactor operators and maintenance personnel. The improvement in reactor operator performance is assumed to result from improved supervision through managers, engineers, and shift technical advisors who would become licensed under the SIR. Furthermore, the improvement, in terms of operator and maintenance personnel performance, is assumed to adequately represent the issue resolution.

As described earlier, the PNL panel felt that the effect of the issue resolution would be small. This is due to the relatively high level of training and qualification of operations personnel which currently exists. Additional licensing requirements would produce some improvement by assisting in the screening of potentially poor performers from the operations staff. The net effect is estimated to be equivalent to a two percent reduction in human error probabilities for reactor operators and maintenance personnel.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Licensing of Additional Operations Personnel (I.A.3.4)

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

   All plants are assumed to be affected.

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

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf 1 - representative BWR

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

4. Parameters Affected by SIR:

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

5. Base-Case Values for Affected Parameters:

   Original values from Appendix A of NUREG/CR-2800 are assumed (Andrews et al. 1983).

6. Affected Accident Sequences and Base-Case Frequencies:

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

TABLE 1 . (contd)

7. Affected Release Categories and Base-Case Frequencies:

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

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

$\bar{F}_{PWR}$ = 7.80E-5/ry $\bar{F}_{BWR}$ = 3.5E-5/ry[a]

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

W_{PWR} = 187.9 man-rem/ry W_{BWR} = 230 man-rem/ry[a]

10. Adjusted-Case Values for Affected Parameters:

B = C	=		0.0032
D = E	=		0.023
CH1	=	CH2 = CH3 = CH4 =	0.0050
CONST1	=		2.0E-4
CONST2	=		5.8E-4
A1 = C1	=		0.0098
B1	=		0.035
(B_3)	=		5.0E-4
K	=		2.6E-5
G1	=		0.014
HHMAN	=	HPMAN1 =	0.098
HPMAN	=		0.015
LPISCM	=		0.0029
HPRSCM	=	WXCM =	0.0029
RCSRBCM	=		3.1E-5
D·E	=		4.9E-4
W·X	=		8.7E-5
B·W	=	C·X =	2.6E-5
D·X	=	E·W =	2.1E-4
B·D	=	E·C =	6.1E-5

(a) See Attachment 1.

TABLE 1 . (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Sequence	Release category	Frequency
T_2MLU -	γ (PWR-3)	= 5.6E-7/ry
	β (PWR-5)	= 8.1E-9/ry
	ε (PWR-7)	= 5.6E-7/ry
T_1MLU -	γ (PWR-3)	= 9.1E-7/ry
	β (PWR-5)	= 1.3E-8/ry
	ε (PWR-7)	= 9.1E-7/ry
$T_1(B_3)MLU$ -	γ (PWR-3)	= 1.1E-6/ry
	β (PWR-5)	= 1.5E-8/ry
	ε (PWR-7)	= 1.1E-6/ry
T_2MQH -	γ (PWR-3)	= 5.5E-6/ry
	β (PWR-5)	= 8.1E-8/ry
	ε (PWR-7)	= 5.5E-6/ry
S_3H -	γ (PWR-3)	= 4.8E-6/ry
	β (PWR-5)	= 7.1E-8/ry
	ε (PWR-7)	= 4.8E-6/ry
S_1D -	α (PWR-1)	= 6.6E-8/ry
	γ (PWR-3)	= 1.3E-6/ry
	β (PWR-5)	= 4.8E-8/ry
	ε (PWR-7)	= 5.3E-6/ry
T_2MQFH -	γ (PWR-2)	= 2.4E-6/ry
	β (PWR-4)	= 3.5E-8/ry
	ε (PWR-6)	= 2.4E-6/ry
S_3FH -	γ (PWR-2)	= 2.0E-6/ry
	β (PWR-4)	= 3.0E-8/ry
	ε (PWR-6)	= 2.0E-6/ry
S_2FH -	α (PWR-1)	= 1.2E-8/ry
	β (PWR-4)	= 8.9E-9/ry
	ε (PWR-6)	= 9.7E-7/ry

TABLE 1 . (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

T_2MLUO -	γ(PWR-3)	= 4.0E-6/ry
	β(PWR-5)	= 5.9E-8/ry
	ε(PWR-7)	= 4.0E-6/ry
T_2KMU -	γ(PWR-3)	= 3.8E-6/ry
	β(PWR-5)	= 5.6E-8/ry
	ε(PWR-7)	= 3.8E-6/ry
S_2D -	α(PWR-1)	= 1.9E-8/ry
	γ(PWR-3)	= 3.9E-7/ry
	β(PWR-5)	= 1.4E-8/ry
	ε(PWR-7)	= 1.6E-6/ry
S_3D -	γ(PWR-3)	= 7.1E-7/ry
	β(PWR-5)	= 1.0E-8/ry
	ε(PWR-7)	= 7.1E-7/ry
T_1MLUO -	γ(PWR-3)	= 2.7E-6/ry
	β(PWR-5)	= 4.0E-8/ry
	ε(PWR-7)	= 2.7E-6/ry
T_3MLUO -	γ(PWR-3)	= 5.3E-7/ry
	β(PWR-5)	= 7.8E-9/ry
	ε(PWR-7)	= 5.3E-7/ry
T_2MOD -	γ(PWR-3)	= 7.6E-7/ry
	β(PWR-5)	= 1.1E-8/ry
	ε(PWR-7)	= 7.6E-7/ry

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

TABLE 1. (contd)

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 1.1E-7/ry

PWR-2 = 5.8E-6/ry

PWR-3 = 2.8E-5/ry

PWR-4 = 9.2E-8/ry

PWR-5 = 4.5E-7/ry

PWR-6 = 7.1E-6/ry

PWR-7 = 3.4E-5/ry

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

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

$\bar{F}^*_{PWR}$ = 7.59E-5/ry

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

W^*_{PWR} = 181.5 man-rem/ry

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

$(\Delta\bar{F})_{PWR}$ = 2.1E-6/ry $(\Delta\bar{F})_{BWR}$ = 8.6E-7/ry[a]

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

$(\Delta W)_{PWR}$ = 6.4 man-rem/ry $(\Delta W)_{BWR}$ = 7.7 man-rem/ry[a]

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.6E+4	2.3E+7	0

(a) See Attachment 1.

ATTACHMENT 1

The RSSMAP studies for Oconee 3 and Grand Gulf 1 give total core-melt frequencies ($\bar{F}_o$) of 8.2E-5/ry and 3.7E-5/ry, respectively, for these plants. Using the original release category frequencies and the public dose factors (Appendix D of NUREG/CR-2800, Andrews et al. 1983), one obtains total public risks (W_o) of 207 man-rem/ry and 250 man-rem/ry, respectively, for Oconee and Grand Gulf. For the purposes of scaling the base-case, affected core-melt frequency ($\bar{F}$) and public risk (W), and the reductions in the core-melt frequency (ΔF) and public risk (ΔW) from Oconee to Grand Gulf, the following are assumed:

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

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

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

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

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

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

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

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Licensing of Additional Operations Personnel (I.A.3.4)

2. Affected Plants (N):

All plants are assumed to be affected.

	N
PWRs	90
BWRs	44

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

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

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

PWR: (19,900 man-rem)(2.1E-6/ry) = 4.2E-2 man-rem/ry

BWR: (19,900 man-rem)(9.5E-7/ry) = 1.9E-2 man-rem/ry

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
131	2.9E+4	0

6-11. Per-Plant Utility Labor in Radiation Zones for SIR Implementation and Operation:

No change in occupational dose is expected from the implementation or operation of this issue resolution. Therefore, $D = D_O = 0$.

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
0	0	0

3.0 SAFETY ISSUE COSTS

The previously described PNL panel also estimated the costs associated with the implementation and operation of the SIR. The PNL panel assumed that the required additional effort to license the majority of the operations personnel at nuclear power plants was roughly equivalent in magnitude to the current licensing efforts for operators and senior operators.

To prepare this SIR for implementation, the NRC would have to prepare qualification criteria, licensing exams and procedures, and hold rulemakings. Clearly this would be a major undertaking. The NRC costs for development are estimated to be in the range of $15 million to $45 million. For purposes of this analysis, a value of $30 million is used as a best estimate.

To implement the SIR, the NRC would have to issue guidelines and Regulatory Guides and promulgate new regulations. This process is estimated to require $5 million.

To operate under the new licensing requirements, it is estimated that the NRC would need 50 new full-time staff members. They would prepare and conduct examinations, review training procedures and manage the NRC effort. In addition to the direct support of the new staff, funds would also be needed for travel, publication, and other functions.

Costs to industry are also expected to be significant. Operations staff would have to undergo specific training to prepare for examinations. Industry would take an active role in rulemaking and other processes. The cost to industry as a whole for implementation is estimated to be equivalent to NRC development and implementation costs, i.e., $35 million.

For operation, industry would have to provide new training staff, staff time for training and examinations, and administration of the added activities. This was estimated to cost $50,000 per plant each year.

The results of the cost analysis are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Licensing of Additional Operations Personnel (I.A.3.4)

2. Affected Plants (N):

   All plants are assumed to be affected.

	N
PWRs	90
BWRs	44
All	134

TABLE 3. (contd)

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

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

Industry Costs (Steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, $\Delta(\bar{FA})$:

PWR: (\$1.65E+9)(2.1E-6/ry) = 3.5E+3/ry

BWR: (\$1.65E+9)(9.5E-7/ry) = 1.6E+3/ry

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

Best Estimate	Upper Bound	Lower Bound
\$1.1E+7	\$2.4E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

Cost is estimated directly in Step 8.

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

Cost is estimated directly in next step.

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

NI = \$3.5E+7

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

Cost is estimated directly in next step.

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_o):

I_o = \$5.0E+4/ry

This applies to all plants.

TABLE 3. (contd)

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

$N\bar{T}I_O$ = \$1.9E+8

12. Total Industry Cost (S_T):

Best Estimate	Upper Bound	Lower Bound
\$2.3E+8	\$3.3E+8	\$1.3E+8

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Cost is estimated directly in next step.

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

C_D = \$3.0E+7

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

Cost is estimated directly in Step 17.

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

Cost is estimated directly in next step.

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

NC = \$5.0E+6

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

NRC will require 50 new full-time staff members under the new licensing requirements. Spread over all plants, the NRC labor allocation becomes: [(50 persons)(1 yr)]/[(134 reactors)(1 yr)] = 0.37 person-yr/ry. Additional travel, publication, etc. expenses would also be incurred by the NRC. This cost is estimated directly in the next step.

TABLE 3. (contd)

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

Labor = (0.37 person-yr/ry)($1.0E+5/person-yr) = $3.7E+4/ry

Travel, publication, other = $7500/ry

C_o = $4.45E+4/ry

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

$N\bar{T}C_o$ = $1.69E+8

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$2.0E+8	$2.9E+8	$1.2E+8

REFERENCES

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

U.S. NRC. 1980. NRC Action Plan Developed as a Result of the TMI-2 Accident. NUREG-0660, U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: I.A.4.2, Long-Term Training Simulator Upgrade

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Training simulators are recognized as a key tool in the training of reactor operators. This TMI action item calls for long-term improvements in simulators to enhance the quality of training provided. Operators would then be more capable of performing their safety-related functions.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	1.2E+5
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	750

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	470
SIR Operation/Maintenance =	530
Total of Above =	1000
Accident Avoidance =	62
NRC COSTS:	
SIR Development =	0
SIR Implementation Support =	1.7
SIR Operation/Maintenance Review =	8.0
Total of Above =	9.7

LONG-TERM TRAINING SIMULATOR UPGRAOE

ISSUE I.A.4.2

1.0 SAFETY ISSUE DESCRIPTION

Nuclear power plant simulators are recognized as an important part of reactor operator training. The TMI Action Plan, NUREG-0660 (NRC 1980), called for a number of actions to improve simulators and their use. There is significant interaction among the simulator-related action items, and clear separation is difficult. Action item I.A.4.2, addressed in this analysis, calls for long-term upgrades in training simulators. Specifically, this item, as described in NUREG-0660, calls for research to improve the use of simulators, develop guidance on the need for and nature of operator action during accidents, and gather data on operator performance. Specific research items mentioned include the following:

- simulator capabilities
- safety-related operator action
- simulator experiments.

The item also calls for the upgrading of training simulator standards, specifically the updating of ANSI/ANS 3.5-1979. A regulatory guide endorsing that standard and giving the criteria for acceptability is also mentioned. The final portion of I.A.4.2 calls for a review of simulators to assure their conformance to the criteria.

A significant portion of the activities to be conducted has been completed. Simulators and simulator training generally have improved since the formulation of the TMI Action Plan. A number of research studies have been completed under I.A.4.2, and others are currently underway. The ANSI/ANS standard has been revised and issued as 3.5-1981. Regulatory Guide 1.149 ("Nuclear Power Plant Simulators for Use in Operator Training"), which endorses that standard, has also been published. The outstanding portion of this item is the continuation of simulator research and the review for conformance to acceptability criteria.

The assessment of this safety issue was conducted by Pacific Northwest Laboratory (PNL) staff, with experience in reactor operator licensing, reactor operation and general reactor safety, in consultation with General Physics Corporation. General Physics Corporation provides utility training services and is greatly experienced in reactor simulators, providing procurement and startup assistance, operation and maintenance services, and simulator modifications. It was assumed that the major effect of this issue, in terms of risk reduction, dose, and cost incurred, would be in the enhancement of the level of realism imparted by simulators.

It was also assumed for this safety issue resolution (SIR) that, in order to provide the intended level of realism, site-specific simulators would be acquired. Such simulators would be significantly more realistic when compared to the specific facilities, both in layout and operation, than existing generic simulators. In addition, they are assumed to have enhanced transient and accident modeling capabilities.

Use of such simulators would significantly improve operator training in the dealing with abnormal conditions. The operator's performance under accident conditions is expected to be enhanced. Thus, potential core melts would be avoided and overall core-melt frequency reduced.

It is clear that the regulations, the ANS standard, and the regulatory guide do not require a site-specific simulator. 10 CFR 55 states that if a simulator is used in training, it "... shall accurately reproduce the operating characteristics of the facility involved and the arrangement of the instrumentation and controls of the simulator shall closely parallel that of the facility involved." ANS 3.5-1981 calls for a high degree of fidelity between the simulator and the "reference plant." However, there is no requirement that the reference plant be the same facility that the personnel in training will in fact operate. Regulatory Guide 1.149 explicitly makes the distinction, stating, "... the similarity that must exist between a simulator and the facility that the operators are being trained to operate is not addressed in the guide and should not be confused with the guidance provided that specifies the similarity that should exist between a simulator and its reference plant."

In PNL's assessment, it was clear that provision of site-specific simulators, while not explicitly required, would meet the fidelity requirements of ANS 3.5-1981 and the accurate reproduction requirements of 10 CFR 55. Less sweeping simulator enhancements might also fulfill these requirements but would have to be decided on a case-by-case basis. Therefore, for risk, dose and cost estimates we assumed the enhancement would be effected by the introduction of site-specific simulators.

It should be acknowledged that, if the intended level of realism could be achieved with existing simulators, the costs to implement the resolution of this safety issue would be significantly less.

Another caveat is that many of the TMI action items associated with operator training are interrelated. It is difficult to assess them independently. These issues strongly tie to I.A.2.6(1,2,3,5), "Long-Term Upgrading of Training and Qualification (Simulators)," which is also being assessed in this program. For the purposes of the analysis, these two issues are separated as follows.

Item I.A.2.6(1,2,3,5) deals with training improvements, including the enhanced use of existing simulators. Item I.A.4.2 deals with the improvement of simulators, providing more realistic modeling of the actual plant. Any item, by itself, would improve operator performance. There are, however, significant overlaps. If all the items were implemented, the total improvement would be less than the sum of the individual contributions as assessed in this program.

Another related item is I.E.8, "Analysis and Dissemination of Operating Experience--Human Error Rate Analysis." This item deals with the analysis of field-collected data on human reliability; the review of occurrence reports, licensee event reports (LERs), and compliance reports; development of models and identification of patterns in human errors. Such activities, while important, have no direct effect on safety. Only through the application of such data in other issues are safety benefits realized. Much of the data from this effort have been used under I.A.4.2. The risk reduction associated with I.E.8 appears, at least partly, in that reported here. Therefore the remaining costs associated with I.E.8 are also reported within this assessment.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose reduction due to accident avoidance are associated with the reduction in operator error expected to result from the training and requalification of operators on improved simulators. It is difficult to produce this estimate. Studies relating human error rates to the realism of simulator training are not available. However, based on engineering judgment, a reduction in operator error rates of 30 percent is estimated to result from the resolution of this safety issue. This recognizes that for specific instances, improvement could be much greater. The 30 percent reduction is used as an estimate of the average improvement.

Results of the analyses for public risk reduction and occupational dose are summarized in Tables 1 and 2, respectively.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Long-Term Training Simulator Upgrade (I.A.4.2)

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

   All plants are assumed to be affected.

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

TABLE 1. (contd)

3. Plants Selected for Analysis:

Oconee 3 - representative PWR

Grand Gulf 1 - representative BWR

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

4. Parameters Affected by SIR:

Oconee: B, C, D, E, CONST1, CONST2, A1, B1, C1, HHMAN, HPMAN, HPMAN1, HPRSCM, WXCM, D•E, B•W, C•X, D•X, E•W, B•D, E•C.

5. Base-Case Values for Affected Parameters:

Original values for Appendix A (Andrews et al. 1983) are assumed.

6. Affected Accident Sequences and Base-Case Frequencies:

$$T_2MLU - \begin{cases} \gamma\ (PWR\text{-}3) = 5.8E\text{-}7/py \\ \beta\ (PWR\text{-}5) = 8.5E\text{-}9/py \\ \varepsilon\ (PWR\text{-}7) = 5.8E\text{-}7/py \end{cases}$$

$$T_1MLU - \begin{cases} \gamma\ (PWR\text{-}3) = 9.8E\text{-}7/py \\ \beta\ (PWR\text{-}5) = 1.4E\text{-}8/py \\ \varepsilon\ (PWR\text{-}7) = 9.8E\text{-}7/py \end{cases}$$

$$T_1(B_3)MLU - \begin{cases} \gamma\ (PWR\text{-}3) = 1.1E\text{-}6/py \\ \beta\ (PWR\text{-}5) = 1.6E\text{-}8/py \\ \varepsilon\ (PWR\text{-}7) = 1.1E\text{-}6/py \end{cases}$$

$$T_2MQH - \begin{cases} \gamma\ (PWR\text{-}3) = 3.2E\text{-}6/py \\ \beta\ (PWR\text{-}5) = 4.7E\text{-}8/py \\ \varepsilon\ (PWR\text{-}7) = 3.2E\text{-}6/py \end{cases}$$

$$S_3H - \begin{cases} \gamma\ (PWR\text{-}3) = 2.8E\text{-}6/py \\ \beta\ (PWR\text{-}5) = 4.1E\text{-}8/py \\ \varepsilon\ (PWR\text{-}7) = 2.8E\text{-}6/py \end{cases}$$

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

S_1D -	α	(PWR-1)	=	5.3E-8/py
	γ	(PWR-3)	=	1.1E-6/py
	β	(PWR-5)	=	3.9E-8/py
	ε	(PWR-7)	=	4.3E-6/py
T_2MQFH -	γ	(PWR-2)	=	2.4E-6/py
	β	(PWR-4)	=	3.6E-8/py
	ε	(PWR-6)	=	2.4E-6/py
S_3FH -	γ	(PWR-2)	=	2.0E-6/py
	β	(PWR-4)	=	3.0E-8/py
	ε	(PWR-6)	=	2.0E-6/py
S_2FH -	γ	(PWR-1)	=	1.2E-8/py
	β	(PWR-4)	=	8.9E-9/py
	ε	(PWR-6)	=	9.8E-7/py
T_2KMU -	γ	(PWR-3)	=	3.9E-6/py
	β	(PWR-5)	=	5.7E-8/py
	ε	(PWR-7)	=	3.9E-6/py
S_2D -	α	(PWR-1)	=	7.2E-9/py
	γ	(PWR-3)	=	1.4E-7/py
	β	(PWR-5)	=	5.2E-9/py
	ε	(PWR-7)	=	5.7E-7/py
S_3D -	γ	(PWR-3)	=	6.7E-7/py
	β	(PWR-5)	=	9.8E-9/py
	ε	(PWR-7)	=	6.7E-7/py
T_2MQD -	γ	(PWR-3)	=	7.2E-7/py
	β	(PWR-5)	=	1.1E-8/py
	ε	(PWR-7)	=	7.2E-7/py

(Note: In each affected accident sequence, the contribution from the non-dominant minimal cut sets is scaled by the ratio of the sum of the affected dominant minimal cut set frequencies to the sum of all the dominant minimal cut set frequencies.)

TABLE 1. (contd)

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 8.0E-8/py

PWR-2 = 5.8E-6/py

PWR-3 = 1.6E-5/py

PWR-4 = 9.3E-8/py

PWR-5 = 2.6E-7/py

PWR-6 = 7.1E-6/py

PWR-7 = 2.0E-5/py

(Note: In each affected release category, with Sequence V excluded from PWR-2, the contribution from the non-dominant accident sequences is scaled by the ratio of the sum of the affected dominant accident sequence frequencies to the sum of all the dominant accident sequence frequencies.)

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

$\bar{F}_{PWR}$ = 4.9E-5/py $\qquad$ $\bar{F}_{BWR}$ = 2.2E-5/py[a]

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

W_{PWR} = 116 man-rem/py $\qquad$ W_{BWR} = 140 man-rem/py[a]

10. Adjusted-Case Values for Affected Parameters:

B = C	=			0.0030
D = E	=			0.022
CONST1	=			2.0E-4
CONST2	=			5.8E-4
A1 = C1	=			0.0098
B1	=			0.035
HHMAN	=	HPMAN1	=	0.07
HPMAN	=			0.0105
HPRSCM	=	WXCM	=	0.0021
D•E	=			4.4E-4
B•W	=	C•X	=	2.4E-5
D•X	=	E•W	=	2.0E-4
B•D	=	E•C	=	5.3E-5

(a) See Attachment 1.

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies:

T_2MLU -	γ	(PWR-3)	= 3.9E-7/py
	β	(PWR-5)	= 5.7E-9/py
	ε	(PWR-7)	= 3.9E-7/py
T_1MLU -	γ	(PWR-3)	= 6.4E-7/py
	β	(PWR-5)	= 9.3E-9/py
	ε	(PWR-7)	= 6.4E-7/py
$T_1(B_3)MLU$ -	γ	(PWR-3)	= 7.5E-7/py
	β	(PWR-5)	= 1.1E-8/py
	ε	(PWR-7)	= 7.5E-7/py
T_2MQH -	γ	(PWR-3)	= 2.4E-6/py
	β	(PWR-5)	= 3.5E-8/py
	ε	(PWR-7)	= 2.4E-6/py
S_3H -	γ	(PWR-3)	= 2.1E-6/py
	β	(PWR-5)	= 3.1E-8/py
	ε	(PWR-7)	= 2.1E-6/py
S_1D -	α	(PWR-1)	= 5.1E-8/py
	γ	(PWR-3)	= 1.0E-6/py
	β	(PWR-5)	= 3.7E-8/py
	ε	(PWR-7)	= 4.1E-6/py
T_2MQFH -	γ	(PWR-2)	= 1.7E-6/py
	β	(PWR-4)	= 2.5E-8/py
	ε	(PWR-6)	= 1.7E-6/py
S_3FH -	γ	(PWR-2)	= 1.4E-6/py
	β	(PWR-4)	= 2.1E-8/py
	ε	(PWR-6)	= 1.4E-6/py
S_2FH -	α	(PWR-1)	= 8.6E-9/py
	β	(PWR-4)	= 6.3E-9/py
	ε	(PWR-6)	= 6.9E-7/py

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

$$T_2KMU - \begin{cases} \gamma \text{ (PWR-3)} = 2.7E-6/py \\ \beta \text{ (PWR-5)} = 4.0E-8/py \\ \varepsilon \text{ (PWR-7)} = 2.7E-6/py \end{cases}$$

$$S_2D - \begin{cases} \alpha \text{ (PWR-1)} = 6.8E-9/py \\ \gamma \text{ (PWR-3)} = 1.4E-7/py \\ \beta \text{ (PWR-5)} = 5.0E-9/py \\ \varepsilon \text{ (PWR-7)} = 6.5E-7/py \end{cases}$$

$$S_3D - \begin{cases} \gamma \text{ (PWR-3)} = 6.7E-7/py \\ \beta \text{ (PWR-5)} = 9.8E-9/py \\ \varepsilon \text{ (PWR-7)} = 7.7E-7/py \end{cases}$$

$$T_2MQD - \begin{cases} \gamma \text{ (PWR-3)} = 7.2E-7/py \\ \beta \text{ (PWR-5)} = 1.1E-8/py \\ \varepsilon \text{ (PWR-7)} = 7.2E-7/py \end{cases}$$

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

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 7.3E-8/py

PWR-2 = 4.1E-6/py

PWR-3 = 1.2E-5/py

PWR-4 = 6.5E-8/py

PWR-5 = 2.0E-7/py

PWR-6 = 5.0E-6/py

PWR-7 = 1.6E-5/py

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

TABLE 1. (contd)

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

$\bar{F}^* = 3.7E\text{-}5/py$

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

$W^*_{PWR} = 86$ man-rem/py

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

$(\Delta\bar{F})_{PWR} = 1.2E\text{-}5/py$ $(\Delta\bar{F})_{BWR} = 5.4E\text{-}6/py$[a]

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

$(\Delta W)_{PWR} = 30$ man-rem/py $(\Delta W)_{BWR} = 36$ man-rem/py[a]

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.2E+5	1.4E+7	0

(a) See Attachment 1.

ATTACHMENT 1

The RSSMAP studies for Oconee 3 and Grand Gulf 1 give five total core-melt frequencies ($\bar{F}_o$) of 8.2E-5/py and 3.7E-5/py, respectively, for these plants. Using the original release category frequencies and the public dose factors (Appendix D of Andrews et al. 1983), one obtains total public risks (W_o) of 207 man-rem/py and 250 man-rem/py, respectively, for Oconee and Grand Gulf. For the purposes of scaling the base-case, affected core-melt frequency (F) and public risk (W), and the reductions in the core-melt frequency (ΔF) and public risk (ΔW) from Oconee to Grand Gulf, the following are assumed:

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

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

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

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

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

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

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

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Long-Term Training Simulator Upgrade (I.A.4.2)

2. Affected Plants (N):

All plants are assumed to be affected.

	N
PWRs	90
BWRs	44

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

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

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

PWR: (19,900 man-rem)(1.2E-5/py) = 0.24 man-rem/py

BWR: (19,900 man-rem)(5.4E-6/py) = 0.11 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
750	1.8E+4	0

6-12. Steps Related to Occupational Dose Increase for SIR Implementation and Operation/Maintenance:

These steps are omitted since the occupational doses for implementation and operation/maintenance are estimated to be zero. Thus, $D = D_O = G = 0$.

3.0 SAFETY ISSUE COSTS

The major effect of this SIR was assumed to be the acquisition and use of site-specific simulators. The acquisition and use of such simulators was viewed as a sufficient, if not necessary, condition to meet the objectives of the TMI Action Plan. The costs to industry of such an undertaking would be substantial. It is important to recognize that if the level of realism possible from enhancement of existing simulators were deemed adequate, the cost to industry would be substantially smaller.

Assuming that new simulators would be required, the principal industry costs for implementation of this safety issue would be the purchase of the simulators and provision of the new training materials. The capital cost of a simulator is estimated to be seven million dollars. The provision of training materials is estimated to be equivalent to a seven-person-year effort.

It was assumed that all reactors, both operating and planned, would be affected. However, not every reactor would require a simulator. Many sites have two or more reactors located together. If these reactors are sufficiently similar, a single simulator could serve them. After examination of the list of 134 operating and planned power reactors, it was estimated that 62 additional site-specific simulators would be adequate. This assumes that 20 percent of the potential simulators are not required because either a site-specific simulator already exists or the plant in question is an older facility with limited remaining lifetime.

The costs for the 62 new simulators spread over 134 reactors yield $3.2 million in capital cost per reactor and 3.2 person-year per reactor to provide new training materials.

The operation and maintenance of the new simulators is estimated to require 3 person-years of effort per simulator. Again, sharing the expense for 62 simulators over 134 reactors yields 1.4 person-years per reactor.

Industries may also experience costs stemming from their participation in simulator experiments and research. However, in comparison to the costs related to new simulators, these costs would be small.

The costs to the NRC are small in comparison to the costs to industry; however, in the NRC context they are significant. The principal costs to the NRC are the continuation of research and the conduct of the confirmatory reviews. No additional developmental costs are foreseen, as the required regulatory guide and ANS standard have been completed.

The continuing research is treated as an implementation cost. It is estimated to require one NRC person-year and $1.0E+6 in contractor support. (This includes the remaining costs associated with item I.E.8.) The confirmatory reviews are also treated as an implementation cost and are estimated to require 4 person-weeks/simulator, or 248 person-weeks in all for the assumed 62 new simulators.

The operational review cost to the NRC is minimal. It is assumed that each simulator will be audited annually to assure that reference plant updates have been adequately represented on the simulator. Such an annual review is estimated to require two person-weeks/simulator, or 124 person-weeks/year for all the 62 assumed new simulators.

The cost estimates, including the development of accident-avoidance cost savings to industry, are shown in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Long-Term Training Simulator Upgrade (I.A.4.2)

2. Affected Plants (N):

   All PWRs and BWRs are assumed to be affected.

	N
PWRs:	90
BWRs:	44

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

	$\bar{T}$ (yr)
PWR:	28.8
BWR:	27.4

Industry Costs (Steps 4 through 12)

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

   PWR = ($1.65E+9)(1.2E-5/py) = $2.0E+4/py

   BWR = ($1.65E+9)(5.4E-6/py) = $8.9E+3/py

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

Best Estimate	Upper Bound	Lower Bound
$6.2E+7	$1.5E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

   Labor: (7 person-yr/simulator)(62 simulators/134 plants)
   = 3.2 person-yr/plant

   Equipment: (62 simulators)/(134 plants) = 0.46 simulators/plant

TABLE 3. (contd)

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

Labor = (3.2)($1.0E+5) = $3.2E+5
Equipment = (0.46)($7.0E+6) = $3.2E+6
I = $3.5E+6/plant

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

(NI) = (134)($3.5E+6) = $4.7E+8

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

1.4 person-yr/py

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_o):

1.4($1.0E+5) = $1.4E+5/py

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

$N\bar{T}I_o$ = [(90)(28.8) + (44)(27.4)]($1.4E+5) = $5.3E+8

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$1.0E+9	$1.4E+9	$6.5E+8

NRC Costs (Steps 13 through 21)

13-14. Steps Related to NRC Cost for SIR Development:

C_D = 0 (development phase assumed to be completed)

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

Continuing Research:	(1 person-yr)(44 person-wk/person-yr)/134 plants	= 0.33 person-wk/plant
Initial Simulator Reviews:	248 person-wk/134 plants	= 1.9 person-wk/plant
		2.2 person-wk/plant

TABLE 3. (contd)

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

NRC Labor:	(2.2)($2270)	= $4990
Contractor Support:	($1.0E+6)/(134)	= $7460
	C	= $12,500/plant

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

NC = (134)($12,500) = $1.7E+6

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

(2 person-wks/year-simulator)(62 simulators/134 plants) = 0.93 person-wk/py

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

0.93($2270) = $2.1E+3/py

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

$N\bar{T}C_O$ = [(90)(28.8) + (44)(27.4)]($2.1E+3) = $8.0E+6

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$9.7E+6	$1.4E+7	$5.6E+6

REFERENCES

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

U.S. NRC. 1980. NRC Action Plan Developed as a Result of the TMI-2 Accident. NUREG-0660, U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: I.B.1.1 (5-7), Management for Operations: Organization and Management of Long-Term Improvements

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This TMI action item seeks to reduce human error by implementing long-term organization and management improvements at all plants.

AFFECTED PLANTS

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	7.1E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	-1.3E+5
Total of Above =	-1.3E+5
Accident Avoidance =	430

COST RESULTS ($$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	1.1
SIR Operation/Maintenance =	-11,000
Total of Above =	-11,000
Accident Avoidance =	36
NRC COSTS:	
SIR Development =	2.4
SIR Implementation Support =	0
SIR Operation/Maintenance Review =	27
Total of Above =	29

MANAGEMENT FOR OPERATIONS:
ORGANIZATION AND MANAGEMENT OF LONG-TERM IMPROVEMENTS
ISSUE I.B.1.1 (5-7)

1.0 SAFETY ISSUE DESCRIPTION

This safety issue as described in NUREG-0660 (NRC 1980) deals with implementation of long-term organization and management improvements as determined necessary by NRC-NRR. The overall objective of this TMI Action Plan (TAP) item is to "improve the licensee's groups responsible for radiation protection and plant operation. The areas to be upgraded include 1) staff size; 2) education and experience of staff members; 3) plant operating and emergency procedures; 4) management awareness of, and attention to, safety matters; and 5) numbers and types of personnel available to respond to accidents" (NRC 1980).

Four of the five areas listed above are covered by other TAP items. Operating staff size, training, and qualifications are covered by items in Section I.A. Similar issues for radiation protection are covered in Section III.D. Procedures are covered in Section I.C, and emergency preparedness is covered in Section III.A.

Therefore, the scope of this issue assessment has been modified to cover only long-term improvements in organization and management of nuclear power plants. Design, construction, startup, and operating phases of these plants are considered.

To assess this safety issue, a number of people at PNL were consulted, including those working for NRC-NRR and NRC-RES on developing the organization and administration regulatory positions. These PNL staff members have expertise in general management, utility and nuclear plant management, reactor operations, reactor operation licensing, general reactor safety, and organizational psychology. Resolution of this issue has potential for reducing the frequency of accidents resulting from the design, construction, operation, and maintenance of a nuclear power plant (NPP).

It should be emphasized that management and organization improvements have the potential to impact virtually all other safety issues. Improvements at the design/construction stage could result in better design, procurement, and installation practices. The plant could be easier to operate and maintain. Improvements at the startup/operation stage would result in a better trained, integrated, and coordinated total staff and a better maintained and functioning plant. Plant modifications would also function better through improved design, procurement, and installation practices. Thus, management and organization improvements would beneficially affect all NPP practices/actions/decisions, including those associated with other issues. Some of those issues might not exist if these improvements were already in place.

As a result of this overlap of management/organization with other safety issues, the risk reduction described below overlaps that of other issues. That is, the analyses of these other issues implicitly assumed, where appropriate, that improved management/organization practices pertaining to the issue were part of its resolution. However, the existing overlap is only partial. This management/organization issue covers more areas than those defined by other issues. Similarly, the analyses of other issues considered risk reduction that would accrue regardless of management/organization changes. Quantification of these overlaps is not feasible within the overall time and funding constraints of this issue analysis.

Resolution of this safety issue is assumed to involve the following:

1. NRC will develop and implement a more rigorous and comprehensive review procedure for operating license (OL) applicants beginning in 1985. This will be based on revisions to the Standard Review Plan (SRP, Chapter 13), a branch position document, and a detailed workbook for obtaining and recording information from written applicant submittals and site visits. Written applicant submittals will be based on internal management documents already used by most utilities.

2. Based on the NRC documents developed above, similar upgrades will be available for construction permit (CP) reviews beginning in 1986.

3. Based on the data needed at the OL stage, agreement will be reached with the Institute of Nuclear Power Operations (INPO) to obtain selected data on operating reactors from INPO's annual corporate and plant reviews. This data will be subject to NRC audit.

   INPO currently conducts, on roughly an annual basis, performance reviews of all NPPs. These include reviews both at the plant and (recently initiated) at the corporate offices. These reviews go into considerable depth in selected functional areas. They typically take about three weeks and involve a team of people representing the various functional areas considered. NRC is currently looking at INPO review criteria to determine the usefulness of these reviews in meeting NRC annual review requirements.

   The NRC will conduct additional annual reviews (integrated with the INPO effort) to obtain complementary data not provided by INPO. This will upgrade the current annual I&E inspections. The effective date is 1985.

4. Data collected will be analyzed against safety outcomes to define practices which should be prescribed, as they will predictably enhance safety. These selective presciptions will be incorporated in the above activities, effective date 1988.

2.0 SAFETY ISSUE RISK AND DOSE

Three major benefits will be obtained from resolution of this safety issue. They include reduced risk, occupational dose and cost. For purposes of this analysis, all benefits/costs are assumed to begin in 1988 when the final upgraded procedures are in place. This applies to the remaining operating life of all nuclear power plants subsequent to implementation of the resolution in 1988, i.e., 24 years.

Public risk reduction and occupational dose resulting from the safety issue resolution (SIR) are discussed below.

PUBLIC RISK REDUCTION

PNL staff experts estimate that SIR could potentially result in a 20 percent reduction in total public risk and core-melt frequency at a nuclear plant. However, many of the plants (assumed to be 25 percent) are already well-managed and organized. They would see limited further improvement. Another 50 percent would obtain only half the benefit, while the remaining 25 percent would obtain the full benefit. An average value of 10 percent for public risk reduction is used in the calculations which follow. These are summarized in Table 1.

OCCUPATIONAL DOSE

There would be substantial reduction in occupational dose, primarily from lower occupational exposure due to fewer unplanned outages. Maintenance staff would be primarily impacted; however, both operating and maintenance staff benefit from avoidance of major accidents.

The potential for exposure reduction is expected to be about 20 percent for those 25 percent of "worst case" plants, half that for the 50 percent of "intermediate case" plants, and little for the 25 percent of "best case" plants. An average value of 10 percent is used in the calculations which follow. It is estimated that 300 to 500 man-rem of occupational exposure occur annually at a typical facility. If we assume 400 man-rem as a best estimate, the 10 percent reduction results in an occupational dose reduction of 40 man-rem per plant year.

Analysis results are summarized in Table 2.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

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

TABLE 1. (contd)

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

All 134 plants. Average lives are based on a SIR effective date of 1988.

	N	$\bar{T}$ (yr)
PWR	90	24.4
BWR	44	22.7

3. Plants Selected for Analysis:

Oconee 3 - representative PWR

Grand Gulf 1 - representative BWR

4-6. Steps Related to Affected Parameters, Accident Sequences, Release Categories, and Their Base-Case Values

For this SIR, a uniform reduction of 10% is applied directly to the affected core-melt frequency and public risk. Thus, Steps 4-7 and 10-12 are omitted.

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

$\bar{F}_{PWR}$ = 8.15E-5/py

$\bar{F}_{BWR}$ = 3.67E-5/py

These are original values for Oconee and Grand Gulf, taken from Appendices A and B of NUREG/CR-2800 (Andrews et al. 1983).

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

W_{PWR} = 207 man-rem/py

W_{BWR} = 250 man-rem/py

These are original values for Oconee and Grand Gulf, taken from Appendices A and B of NUREG/CR-2800 (Andrews et al. 1983).

10-12. Steps Related to Adjusted-Case Values of Affected Parameters, Accident Sequences and Release Categories:

Analysis is not performed for these steps since the 10% reduction is applied directly to the affected core-melt frequency and public risk.

TABLE 1. (contd)

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

$\bar{F}^*_{PWR}$ = 7.34E-5/py $\bar{F}^*_{BWR}$ = 3.30E-5/py

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

W^*_{PWR} = 186 man-rem/py W^*_{BWR} = 225 man-rem/py

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

$\Delta\bar{F}_{PWR}$ = 8.2E-6/py $\Delta\bar{F}_{BWR}$ = 3.7E-6/py

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

ΔW_{PWR} = 21 man-rem/py ΔW_{BWR} = 25 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
7.1E+4	2.1E+7	0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

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

2. Affected Plants (N):

All 134 plants.

	N
PWRs	90
BWRs	44

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

	$\bar{T}$ (yr)
PWRs	24.4
BWRs	22.7

(These are based on a SIR effective date of 1988.)

TABLE 2. (contd)

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

PWR: (19,860 man-rem)(8.2E-6/py) = 0.16 man-rem/py

BWR: (19,860 man-rem)(3.7E-6/py) = 0.073 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
430	2.6E+4	0

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

These steps do not apply since the implementation occupational dose is zero; i.e., D = 0.

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

Dose increase is estimated directly in the next step.

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

D_O = -40 man-rem/py (Negative sign indicates reduction.)

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

$N\bar{T}O_O$ = [90(24.4 yr) + 44(22.7 yr)](-40 man-rem/py)
= 1.28E+5 man-rem (reduction)

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
-1.3E+5	-4.3E+4	-3.8E+5

(Negative signs indicate reductions.)

3.0 SAFETY ISSUE COSTS

There are both cost increases and cost savings involved in resolving this safety issue. BWRs and PWRs would be affected equally.

The nuclear industry has already recognized the potential for cost savings in this area. INPO gives top priority and devotes major resources to its plant and corporate performance review programs. But the INPO effort is still in its infancy. Most of the benefits lie ahead.

If the NRC links synergistically with the INPO effort, as planned in the resolution of this safety issue, large cost benefits could accrue. The NRC efforts provide a more holistic approach to organization and administration practices, complementing the functional approach used by INPO.

NRC efforts will provide a data base upon which to identify practices which predictably enhance safety and plant performance. This provides both insight and added teeth to the INPO effort. In addition, INPO is in the position to obtain industry acceptance of good practices, identified in the NRC efforts, which cannot be required by the NRC due to inadequate safety justification.

On the other hand, the INPO effort provides insights to the NRC on what constitutes good practices. These insights will help assure that NRC actions will indeed achieve the desired results and that there is consistency between the INPO and NRC efforts.

Specifically, industry costs associated with this issue are expected to be as follows:

1. The average cost of obtaining a CP or OL would increase approximately $100,000 as a result of additional rigor of this process.

2. The average cost of an annual operating reactor review could increase approximately $25,000 due to the greater depth of these reviews and the need to audit INPO data.

3. An annual cost savings of approximately $750,000 could be obtained as a result of more effective and efficient management and organization practices. Management, support staff, and/or use of outside services would be reduced in several ways. Better organization and job design would facilitate coordination and integration, thereby eliminating unnecessary overlaps and emphasis on less important activities to the detriment of higher-priority work. Better cross-training, internal communication, and review practices would improve the effectiveness of support activities and minimize rework. Better planning and external communication practices would enable problems to be anticipated and mitigated in a more cost-effective manner. Less time would be required for Licensee Event Reports (LERs) and other responses to the NRC, due to improved safety performance. In addition, improved safety performance would reduce the potential for NRC fines. This can be a significant cost savings; for example, one recent fine alone exeeded the $750,000 cost savings estimated here.

4. Average plant availability is assumed to be increased 4 percent by 1988. A more effective and better-integrated total organization would result in staff making fewer operational and maintenance errors and would provide better design, procurement, and installation of plant modifications. Similarly, for those plants impacted at the design/construction stage, a better-functioning plant would result, and initial startup activities would proceed more expeditiously. The net result would be a better-operating plant with fewer unplanned outages. Planned outage time could also would be reduced because of better outage management practices and less rework.

NRC costs associated with resolving this safety issue are expected to be as follows:

1. Approximately 22 man-years of effort by NRR and RES to develop the long-term regulatory position on management and organization after FY 1982. This includes training of NRC staff to implement this position.

2. Approximately 2 man-years to write, obtain, and issue comments on revised and new regulatory guides. The major development effort behind these guides is included in item 1.

3. Approximately 5 additional man-months for CP reviews. No additional time for OL reviews.

4. Approximately 1 additional man-month to perform an annual assessment of this SIR at each plant, including the audit of INPO data.

Table 3 summarizes the results of the cost analysis.

<u>TABLE 3.</u> Safety Issue Cost Work Sheet

1. <u>Title and Identification Number of Safety Issue</u>:

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

2. <u>Affected Plants (N)</u>:

    All 134 plants.

	N
PWRs	90
BWRs	44

TABLE 3. (contd)

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

	$\bar{T}$ (yr)
PWRs	24.4
BWRs	22.7

These are based on a SIR effective date of 1988.

Industry Costs (Steps 4 through 12)

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

PWR: (\$1.65E+9)(8.2E-6/py) = \$1.35E+4

BWR: (\$1.65E+9)(3.7E-6/py) = \$6.1E+3

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

Best Estimate	Upper Bound	Lower Bound
\$3.6E+7	\$2.1E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

Cost is estimated directly in the next step.

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

I = \$1.0E+5/plant

This applies only to the 7 PWRs and 4 BWRs obtaining operating licenses after 1988.

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

NI = (11)(\$1.0E+5/plant) = \$1.1E+6

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

Cost is estimated directly in the next step.

TABLE 3. (contd)

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_O):

Cost of operating reactor reviews would increase by \$2.5E+4/py, while an estimated 7.5E+5/py would be saved by improved management and organization practices. Further cost savings could accrue from an estimated 4% increase in plant availability. Using a best estimate of 65% for the average nuclear plant capacity factor in the U.S. and a cost of \$3.0E+5/day for replacement power, this cost savings is estimated to be

(0.04)(0.65)(\$3.0E+5/plant-day)(365 plant-days/py)

= \$2.85E+6/py (savings)

Therefore,

I_O = \$2.5E+4/py - \$7.5E+5/py - \$2.85E+6/py = -\$3.57E+6/py

(The negative sign indicates a cost savings.)

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

$N\bar{T}I_O$ = [90(24.4 yr) + 44(22.7 yr)](-\$3.57E+6/py) = -\$1.14E+10

(The negative sign indicates a cost savings.)

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$-1.1E+10	\$-5.7E+9	\$-1.7E+10

(The negative signs indicate cost savings.)

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Develop regulatory position	= 22 man-yr
Process comments on regulatory guides	= 2 man-yr
	24 man-yr

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

C_D = (24 man-yr)(\$1.0E+5/man-yr) = \$2.4E+6

TABLE 3. (contd)

15-17. Steps Related to NRC Cost for Support of SIR Implementation:

No NRC labor is foreseen. Therefore, C = 0.

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

1 man-mo/py

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

C_O = (1 man-mo/py)(1 man-yr/12 man-mo)($1.0E+5/man-yr) = $8.33E+3/py

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

$N\bar{T}C_O$ = [90(24.4 yr) + 44(22.7 yr)]($8.33E+3/py = $2.66E+7

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$2.9E+7	$4.2E+7	$1.6E+7

REFERENCES

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

U.S. NRC. 1980. NRC Action Plan Developed as a Result of the TMI-2 Accident. NUREG-0660, U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: I.C.9, Long-Term Program Plan for Upgrading Procedures

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This TMI action item calls for the development of a program plan for upgrading of plant procedures. This plan would integrate and expand current efforts to improve and coordinate procedures. Due to the currently poor state of procedures, a significant potential exists for safety improvement. Unfortunately, however, as the plan is currently envisioned, there is a certain lack of guidance which could result in strict audit and enforcement action. Thus, the improved procedures would be prevented from reaching full potential.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 2.1E+5

OCCUPATIONAL DOSES:

SIR Implementation =	250
SIR Operation/Maintenance =	-3.8E+4
Total of Above =	-3.8E+4
Accident Avoidance =	1400

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:

SIR Implementation =	67
SIR Operation/Maintenance =	380
Total of Above =	450
Accident Avoidance =	110

NRC COSTS:

SIR Development =	0.50
SIR Implementation Support =	1.0
SIR Operation/Maintenance Review =	7.2
Total of Above =	8.7

LONG-TERM PROGRAM PLAN FOR UPGRADING PROCEDURES

ISSUE I.C.9

1.0 SAFETY ISSUE DESCRIPTION

The description of this action item in NUREG-0660 (NRC 1980) called for the NRC [in an effort to be led by the Office of Nuclear Reactor Regulation (NRR), but to involve the Offices of Inspection and Enforcement (IE), Standards Development (SD), and Nuclear Regulatory Research (RES)] to develop a long-term program plan for the upgrading of plant procedures. This plan would incorporate and expand on current efforts associated with the development, review and monitoring of procedures. Consideration was to be given to studies to insure clear procedures with particular emphasis on diagnostic aids for off-normal conditions. The interrelationships of administrative, operating, maintenance, test and surveillance procedures would be considered. Emergency procedures, reliability analysis, human factors engineering, crisis management and operator training would be addressed.

The NUREG-0660 schedules called for the submittal of the plan by July 1981. The current schedule has an October 1982 completion data. Some forerunner work on emergency procedures (NUREG-0799, NRC 1981) has been completed, including a thorough audit and review of near-term operating license (NTOL) plants.

A tentative draft of the plan schedules the emergency procedure upgrading guidance for May 1982 (NUREG-0899, NRC 1982). The upgrading of other operating and maintenance procedures is scheduled in three phases:

1. Survey ongoing studies, existing procedures and practices of related industries. Assess problems, prioritize solutions (FY82-83).

2. Prepare guidance (from NUREGs and Regulatory Guides) for industry use (FY83-84).

3. Issue requirements, prepare inspection guidance, review or audit as necessary (FY85-86).

In order to assess this safety issue, a panel of Pacific Northwest Laboratory (PNL) experts was assembled. The panel included operator license examiners and members with reactor operations experience, utility field experience as well as general reactor safety backgrounds. Some members of the panel had participated in the study concerned with emergency procedures of NTOL facilities.

In their consideration of this issue, the PNL panel saw significant potential for safety improvement, based primarily on the preception that the existing procedures are poorly written and reviewed. However, the panel felt that the full extent of the potential safety improvement would not be realized. It was their understanding that the current planning at NRC turns away from the thorough review and audit approach first taken on NTOL emergency procedures. Instead, the guidance generated is to be employed by industry voluntarily and audited by the resident IE inspector. This guidance, as currently understood by PNL, is perceived to lack sufficient specificity to direct licensees to a common product. Furthermore, IE is not given criteria against which to inspect. The lack of a consistent, comprehensive audit and enforcement program, especially the lack of appreciation for applicable human factors practices, is expected to significantly reduce the potential impact of long-term upgrading of procedures.

2.0 SAFETY ISSUE RISK AND DOSE

This safety issue resolution (SIR) is expected to have a significant impact on plant procedures. The changes in procedures, in turn, are expected to improve the safety-related performance of all plant operations staff, under both routine and abnormal operating conditions. To measure the improvement in safety, the PNL panel estimated the reduction in human error probabilities for operations staff.

As was previously discussed, the PNL panel felt that the potential for improvement was largely due to the relatively poor current status of procedures. However, as the plan now is perceived, the full extent of this safety improvement is not expected to be realized.

Concurrent with the changes in human error probabilities, the resolution of this issue will result in minor changes to the occupational doses. A slight increase in occupational exposure is expected to accompany the implementation of the resolution. The increase would result from plant visits during the review of existing practices and problems.

PUBLIC RISK REDUCTION

The public risk reduction is measured in man-rem of public exposure. This is estimated through the change in core-melt frequency. The starting point in developing the estimate for core-melt frequency change is an estimation of the changes in human error probabilities for operations staff.

As discussed previously, the PNL panel noted a high potential for improvements in human error probabilities associated with procedures upgrading. Improvements from 20 percent to 70 percent were estimated. However, as the plan for procedure upgrades is currently understood, the PNL panel felt that the high potential would not be realized. The reason is the perceived lack of strong direction to licensees and strong audit and enforcement guidance for IE. A human error probability reduction of 30 percent was estimated to result from the resolution of this safety issue. While still significant, this is less than might have been otherwise expected.

Table 1 summarizes the analysis results for public risk reduction.

OCCUPATIONAL DOSE

The resolution of this issue is expected to result in small increases in occupational exposure. To implement the SIR, NRC and plant operations staff will undergo a small increase in exposure due to the plant visits in the review portion of the procedure-upgrading plan. This was estimated by the PNL panel to be 0.5 percent of the normal annual occupational exposure. The normal annual occupational exposure is estimated to be from 300 to 1000 man-rem/plant-year. Taking 700 man-rem/plant-year as a best estimate yields an estimate of 3.5 man-rem per plant for implementation. It should be noted that only plants currently operating will be visited. The information gained will be applied to all plants, but the exposure will occur only in currently completed facilities.

Resolution of this issue will produce upgraded procedures and more efficient performance of maintenance and other activities in radiation zones. Operational occupational exposures, then, should be somewhat reduced (1-2 percent estimated by the PNL panel). Using 1.5 percent as a best estimate with the 700 man-rem/plant-year exposure best estimate results in a reduction in operational dose of ~10 man-rem/plant-year.

Table 2 summarizes the analysis results for occupational dose.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Long-term Program Plan for Upgrading Procedures (I.C.9)

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

   All plants are assumed to be affected.

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

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf 1 - representative BWR

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

TABLE 1. (contd)

4. Parameters Affected by SIR:

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

5. Base-Case Values for Affected Parameters:

Original values from Appendix A (Andrews et al. 1983) are assumed.

6. Affected Accident Sequences and Base-Case Frequencies:

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

7. Affected Release Categories and Base-Case Frequencies:

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

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

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

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

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

10. Adjusted-Case Values for Affected Parameters:

B = C =	0.0024
D = E =	0.020
CH1 = CH2 = CH3 = CH4 =	0.0044
CONST1 =	1.4E-4
CONST2 =	4.3E-4
A1 = C1 =	0.0091

(a) See Attachment 1.

TABLE 1. (contd)

10. Adjusted-Case Values for Affected Parameters (contd):

B1	=		0.034
(B_3)	=		4.8E-4
K	=		1.9E-5
G1	=		0.012
HHMAN	=	HPMAN1 =	0.07
HPMAN	=		0.0105
LPISCM	=		0.0021
HPRSCM	=	WXCM =	0.0021
RCSRBCM	=		2.2E-5
D•E	=		3.8E-4
W•X	=		7.8E-5
B•W	=	C•X =	1.9E-5
D•X	=	E•W =	1.7E-4
B•D	=	E•C =	4.2E-5

11. Affected Accident Sequences and Adjusted-Case Frequencies:

T_2MLU -	γ (PWR-3)	=	2.8E-7/ry
	β (PWR-5)	=	4.1E-9/ry
	ϵ (PWR-7)	=	2.8E-7/ry
T_1MLU -	γ (PWR-3)	=	4.7E-7/ry
	β (PWR-5)	=	6.9E-9/ry
	ϵ (PWR-7)	=	4.7E-7/ry
$T_1(B_3)MLU$ -	γ (PWR-3)	=	7.1E-7/ry
	β (PWR-5)	=	1.0E-8/ry
	ϵ (PWR-7)	=	7.1E-7/ry
T_2MQH -	γ (PWR-3)	=	4.0E-6/ry
	β (PWR-5)	=	5.9E-8/ry
	ϵ (PWR-7)	=	4.0E-6/ry

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

S_3H -	γ (PWR-3)	=	3.6E-6/ry
	β (PWR-5)	=	5.2E-8/ry
	ε (PWR-7)	=	3.6E-6/ry
S_1D -	α (PWR-1)	=	5.7E-8/ry
	γ (PWR-3)	=	1.1E-6/ry
	β (PWR-5)	=	4.1E-8/ry
	ε (PWR-7)	=	4.5E-6/ry
T_2MQFH -	γ (PWR-2)	=	1.8E-6/ry
	β (PWR-4)	=	2.6E-8/ry
	ε (PWR-6)	=	1.8E-6/ry
S_3FH -	γ (PWR-2)	=	1.5E-6/ry
	β (PWR-4)	=	2.2E-8/ry
	ε (PWR-6)	=	1.5E-6/ry
S_2FH -	α (PWR-1)	=	8.9E-9/ry
	β (PWR-4)	=	6.5E-9/ry
	ε (PWR-6)	=	7.1E-7/ry
T_2MLUO -	γ (PWR-3)	=	3.5E-6/ry
	β (PWR-5)	=	5.1E-8/ry
	ε (PWR-7)	=	3.5E-6/ry
T_2KMU -	γ (PWR-3)	=	2.0E-6/ry
	β (PWR-5)	=	2.9E-8/ry
	ε (PWR-7)	=	2.0E-6/ry
S_2D -	α (PWR-1)	=	1.5E-8/ry
	γ (PWR-3)	=	3.0E-7/ry
	β (PWR-5)	=	1.1E-8/ry
	ε (PWR-7)	=	1.2E-6/ry

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

$$S_3D - \begin{cases} \gamma \text{ (PWR-3)} = 6.2E\text{-}7/ry \\ \beta \text{ (PWR-5)} = 9.1E\text{-}9/ry \\ \varepsilon \text{ (PWR-7)} = 6.2E\text{-}7/ry \end{cases}$$

$$T_1MLUO - \begin{cases} \gamma \text{ (PWR-3)} = 2.3E\text{-}6/ry \\ \beta \text{ (PWR-5)} = 3.4E\text{-}8/ry \\ \varepsilon \text{ (PWR-7)} = 2.3E\text{-}6/ry \end{cases}$$

$$T_3MLUO - \begin{cases} \gamma \text{ (PWR-3)} = 4.6E\text{-}7/ry \\ \beta \text{ (PWR-5)} = 6.7E\text{-}9/ry \\ \varepsilon \text{ (PWR-7)} = 4.6E\text{-}7/ry \end{cases}$$

$$T_2MQD - \begin{cases} \gamma \text{ (PWR-3)} = 6.7E\text{-}7/ry \\ \beta \text{ (PWR-5)} = 9.7E\text{-}9/ry \\ \varepsilon \text{ (PWR-7)} = 6.7E\text{-}7/ry \end{cases}$$

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

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 8.8E-8/ry

PWR-2 = 4.2E-6/ry

PWR-3 = 2.1E-5/ry

PWR-4 = 6.7E-8/ry

PWR-5 = 3.4E-7/ry

PWR-6 = 5.2E-6/ry

PWR-7 = 2.6E-5/ry

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

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

$\bar{F}^*_{PWR} = 5.6E\text{-}5/ry$

TABLE 1. (contd)

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

W^*_{PWR} = 135 man-rem/ry

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

$(\Delta\bar{F})_{PWR}$ = 2.2E-5/ry $(\Delta\bar{F})_{BWR}$ = 9.9E-6/ry[(a)]

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

$(\Delta W)_{PWR}$ = 53 man-rem/ry $(\Delta W)_{BWR}$ = 64 man-rem/ry[(a)]

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.1E+5	2.3E+7	0

(a) See Attachment 1.

<u>ATTACHMENT 1</u>

The RSSMAP studies for Oconee 3 and Grand Gulf 1 give total core-melt frequencies ($\bar{F}_0$) of 8.2E-5/ry and 3.7E-5/ry, respectively, for these plants. Using the original release category frequencies and the public dose factors (Appendix D, Andrews 1983), one obtains total public risks (W_0) of 207 man-rem/ry and 250 man-rem/ry, respectively, for Oconee and Grand Gulf. For the purposes of scaling the base-case, affected core-melt frequency ($\bar{F}$) and public risk (W), and the reductions in the core-melt frequency ($\Delta\bar{F}$) and public risk (ΔW) from Oconee to Grand Gulf, the following are assumed:

$$\left.\begin{array}{l} \bar{F}_{BWR}/\bar{F}_{PWR} \\ (\Delta\bar{F})_{BWR}/(\Delta\bar{W})_{PWR} \end{array}\right\} = (\bar{F}_0)_{BWR}/(\bar{F}_0)_{PWR}$$

$$\left.\begin{array}{l} W_{BWR}/W_{PWR} \\ (\Delta W)_{BWR}/(\Delta W)_{PWR} \end{array}\right\} = (W_0)_{BWR}/(W_0)_{PWR}$$

Using the original values of $\bar{F}_0$ and W_0 for Oconee and Grand Gulf, the scaling equations become

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

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

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

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

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Long-Term Program Plan for Upgrading Procedures (I.C.9)

2. Affected Plants (N):

   All plants are assumed to be affected.

	N
PWRs	90
BWRs	44

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

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

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

   PWR: (19,900 man-rem)(2.2E-5/ry) = 0.44 man-rem/ry

   BWR: (19,900 man-rem)(9.9E-6/ry) = 0.20 man-rem/ry

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.4E+3	2.9E+4	0

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

   Dose is estimated directly in Step 7.

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

   3.5 man-rem/plant (applies only to 71 operating reactors, see text)

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

   248 man-rem

TABLE 2. (contd)

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

Dose increase is estimated directly in next step.

10. Per-Plant Occupational Dose Increase for SIR Operation and Maintenance D_O:

D_O = -10 man-rem/ry (Negative sign indicates reduction)
This applies to all plants.

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

$N\bar{T}D_O$ = -3.8E+4 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
-3.8E+4	-1.3E+4	-1.1E+5

(Negative signs indicate reductions.)

3.0 SAFETY ISSUE COSTS

The PNL panel also estimated costs associated with the implementation and operation of this issue for the NRC and industry.

It was estimated that implementing the comprehensive upgrading of procedures would require a 5 person-year effort for the average facility. Assuming a utility staff cost of $1.0E+5 per person-year yields an estimated cost of $5.0E+5/plant to implement the upgrade. Operation under the new procedures as well as the continuing review and improvement of procedures, was estimated to require one person-year/year, or $1.0E+5/yr at each facility.

The NUREG-0660 (NRC 1980) description called for 4.9 person-years of NRC effort between NRR, IE, SD, and RES for the development and implementation of this issue. The panel felt that an even greater effort would be required, including some contractor support. Since the NRC staff time and contractor time are somewhat interchangeable, no estimate of direct NRC staff hours was made. The total NRC development and implementation effort was estimated to require $1.5E+6. This is assumed to be divided as $1.0E+6 for development and $5.0E+5 for implementation. The NRC effort for annual reviews and ongoing work in procedures upgrades was estimated to require 2.5 person-years, which at $1.0E+5 per person-year is equivalent to $2.5E+5 per year.

Table 3 summarizes the analysis results for industry and NRC costs.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Long-Term Program Plan for Upgrading Procedures (I.C.9)

2. Affected Plants (N):

All plants are assumed to be affected.

	N
PWR	90
BWR	44
All	134

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

	$\bar{T}$ (yr)
PWR	28.8
BWR	27.4
All	28.3

Industry Costs (Steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance $\Delta(\overline{FA})$:

PWR: ($1.65E+9)(2.2E-5/ry) = $3.6E+4/ry

BWR: ($1.65E+9)(9.9E-6/ry) = $1.6E+4/ry

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

Best Estimate	Upper Bound	Lower Bound
$1.1E+8	$2.4E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

5 person-yr/plant

This applies to all plants.

TABLE 3. (contd)

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

$I = \$5.0E+5/\text{plant}$

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

$NI = \$6.7E+7$

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

1 person-yr/ry

This applies to all plants.

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

$I_0 = \$1.0E+5/ry$

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

$N\bar{T}I_0 = \$3.8E+8$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$4.5E+8	\$6.4E+8	\$2.6E+8

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Cost is estimated directly in Step 14.

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

$C_D = \$5.0E+5$

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

Cost is estimated directly in Step 17.

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

Cost is estimated directly in Step 17.

TABLE 3. (contd)

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

NC = \$1.0E+6

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

If the total estimated annual NRC review labor of 2.5 person-yr is spread over all the plants, the following estimate is obtained:

2.5 person-yr/134 ry = 1.9E-2 person-yr/ry

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

\$1900/ry

20. Total NRC Cost for Review of SIR Operation and Maintenance ($\bar{NTC}_O$):

$\bar{NTC}_O$ = \$7.2E+6

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$8.7E+6	\$1.2E+7	\$5.1E+6

REFERENCES

Andrews, W. B., et. al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: I.D.3, Safety System Status Monitoring

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This TMI action item seeks to reduce operator error during an emergency by continuously informing the operator of the status of all important safety system components. This issue applies to all presently operating plants, and the implementation of the SIR is assumed to take place concurrently with control room redesign (Item I.D.1).

AFFECTED PLANTS	BWR:	Operating = 24	Planned = 0
	PWR:	Operating = 47	Planned = 0

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	1.2E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	1800
SIR Operation/Maintenance =	760
Total of Above =	2500
Accident Avoidance =	27

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	54
SIR Operation/Maintenance =	4.3
Total of Above =	58
Accident Avoidance =	2.3
NRC COSTS:	
SIR Development =	0.05
SIR Implementation Support =	0.65
SIR Operation/Maintenance Review =	2.2
Total of Above =	2.9

SAFETY SYSTEM STATUS MONITORING

ISSUE I.D.3

1.0 SAFETY ISSUE DESCRIPTION

This safety issue as described in NUREG-0660 (NRC 1980) calls for the NRC to study the need for all licensees and applicants not presently committed to the requirements of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," to monitor and verify operations, test, and maintenance activities by means of an automatic status monitoring system. Regulatory Guide 1.47 is already being applied to plants under construction and to license applicants. Therefore, this safety issue will involve backfitting all operating power plants.

This study is to be performed following a review of procedures and other nonautomatic actions to verify the safety system status, as required in TMI action item I.C.6, and installation of the safety monitor console (Item I.D.2). In addition, consideration should be given to the impact of other control room modifications on the need for automatic status monitoring (Item I.D.1).

To assess this safety issue, a number of staff at the Pacific Northwest Laboratory (PNL) were consulted. These staff have expertise in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas.

The PNL staff determined that the resolution of this safety issue should be undertaken after a thorough review of all components in the system to judge their importance in overall system risk. This review would be similar to one performed by EG&G Idaho in "Light Water Reactor Engineering Safety Features Status Monitoring Final Report" (NUREG/CR-2278, Brown and Vonherrmann 1981). After identification of components to be monitored and the system to be installed, this safety issue implementation should only take place as a part of a complete control room modification (Item I.D.1). Simply adding the safety system status monitoring devices in the present congested control rooms would most likely result in an increase in risk rather than a reduction. Without consolidation and simplification of the operator's job, the addition of a new system unintegrated with the other control panel systems would result in more confusion and operator error rather than less.

The new system, integrated with a new control room, would reduce operator error which, in turn, will lower the risk associated with operation of the monitored safety systems. The monitoring system envisioned takes information directly from the components and/or equipment whose failure can disable the safety function, and displays it in a coherent pattern in the control room. Any time the component is bypassed or inoperable for any reason, this status is indicated to the control room operator. Using this information during an emergency will result in a reduction in operator errors.

2.0 SAFETY ISSUE RISK AND DOSE

The analyses of public risk reduction and occupational dose resulting from SIR are described in the following two sections.

PUBLIC RISK REDUCTION

As was previously discussed, the major result of this SIR was assumed to be a reduction in operator error. For some utilities, this new system may result in a modest reduction in operator error during an emergency, whereas in others the system may have no discernible effect. An average of about 2 percent was arrived at to apply to all presently operating plants. Thus, this issue assumes the installation of the safety status monitoring system at the 47 operating PWRs and 24 operating BWRs.

Table 1 summarizes the results of the public risk reduction analysis.

OCCUPATIONAL DOSE

Occupational dose will be accumulated by personnel installing the safety status monitoring system when working in radiation zones. In addition, some maintenance work will be needed on the new monitoring system, which will also result in occupational dose being received.

The implementation of this SIR would require the installation of a safety status monitoring system at each operating plant. This installation was estimated to require 17 man-months of labor per plant, part of which would take place in highly radioactive zones. An average dose rate of 10 millirems per hour was assumed for installation labor. This average is estimated from the different radiation environments the workers would encounter during the installation process. The labor in a radiation zone during maintenance of the safety status monitoring system was estimated at one man-week per plant-year. Again, an average of 10 millirems per hour was assumed as the dose rate for the maintenance personnel.

Table 2 summarizes the results of the occupational dose analysis.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Safety System Status Monitoring (I.D.3)

TABLE 1. (contd)

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

All operating plants are assumed to be affected.

	N	$\bar{T}$ (yr)
PWRs	47	27.7
BWRs	24	25.2

3. Plants Selected for Analysis:

Oconee 3 - representative PWR

Grand Gulf 1 - representative BWR

(This analysis is conducted for Oconee 3, and the results are scaled for Grand Gulf 1, as discussed in Attachment 1.)

4. Parameters Affected by SIR:

Oconee: B, C, D, E, CONST1, CONST2, A1, C1, B1, HHMAN, HPMAN1, HPMAN, HPRSCM, WXCM, D·E, B·W, C·X, D·X, E·W, B·D, E·C

5. Base-Case Values for Affected Parameters:

Original values from Appendix A are assumed (Andrews et al. 1983).

6. Affected Accident Sequences and Base-Case Frequencies:

Sequence		Frequency (1/py)
T_2MLU -	γ (PWR-3)	5.8E-7
	β (PWR-5)	8.5E-9
	ε (PWR-7)	5.8E-7
T_1MLU -	γ (PWR-3)	9.8E-7
	β (PWR-5)	1.4E-8
	ε (PWR-7)	9.8E-7
$T_1(B_3)$MLU -	γ (PWR-3)	1.1E-6
	β (PWR-5)	1.6E-8
	ε (PWR-7)	1.1E-6

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

Sequence		Frequency (1/py)
T_2MQH -	γ (PWR-3)	3.2E-6
	β (PWR-5)	4.7E-8
	ϵ (PWR-7)	3.2E-6
S_3H -	γ (PWR-3)	2.8E-6
	β (PWR-5)	4.1E-8
	ϵ (PWR-7)	2.8E-6
S_1D -	α (PWR-1)	5.3E-8
	γ (PWR-3)	1.1E-6
	β (PWR-5)	3.9E-8
	ϵ (PWR-7)	4.3E-6
T_2MQFH -	γ (PWR-2)	2.4E-6
	β (PWR-4)	3.6E-8
	ϵ (PWR-6)	2.4E-6
S_3FH -	γ (PWR-2)	2.0E-6
	β (PWR-4)	3.0E-8
	ϵ (PWR-6)	2.0E-6
S_2FH -	α (PWR-1)	1.2E-8
	β (PWR-4)	8.9E-9
	ϵ (PWR-6)	9.8E-7
T_2KMU -	γ (PWR-3)	3.9E-6
	β (PWR-5)	5.7E-8
	ϵ (PWR-7)	3.9E-6
S_2D -	α (PWR-1)	7.2E-9
	γ (PWR-3)	1.4E-7
	β (PWR-5)	5.2E-9
	ϵ (PWR-7)	5.7E-7

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

Sequence		Frequency (1/py)
S_3D -	γ (PWR-3)	6.7E-7
	β (PWR-5)	9.8E-9
	ε (PWR-7)	6.7E-7
T_2MQD -	γ (PWR-3)	7.2E-7
	β (PWR-5)	1.1E-8
	ε (PWR-7)	7.2E-7

Note: In each affected accident sequence, the contribution from the non-dominant minimal cut sets is scaled by the ratio of the sum of the affected dominant minimal cut set frequencies to the sum of all the dominant minimal cut set frequencies.

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 8.0E-8/py

PWR-2 = 5.8E-6/py

PWR-3 = 1.6E-5/py

PWR-4 = 9.3E-8/py

PWR-5 = 2.6E-7/py

PWR-6 = 7.1E-6/py

PWR-7 = 2.0E-5/py

Note: In each affected release category, with Sequence V excluded from PWR-2, the contribution from the non-dominant accident sequences is scaled by the ratio of the sum of the affected dominant accident sequence frequencies to the sum of all the dominant accident sequence frequencies.

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

$\bar{F}_{PWR}$ = 4.906E-5/py $\bar{F}_{BWR}$ = 2.2E-5/py[(a)]

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

W_{PWR} = 116.3 man-rem/py W_{BWR} = 140 man-rem/py[(a)]

(a) See Attachment 1.

TABLE 1. (contd)

10. Adjusted-Case Values for Affected Parameters:

All affected parameters are assumed to experience a 2% decrease in their failure probabilities as a result of SIR. However, this decrease is evident to two significant figures only for the following parameters:

HHMAN = HPMAN1 = 0.098

HPRSCM = WXCM = 0.0029

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Sequence		Frequency (1/py)
T_2MQH -	γ (PWR-3)	3.1E-6
	β (PWR-5)	4.6E-8
	ε (PWR-7)	3.1E-6
S_3H -	γ (PWR-3)	2.8E-6
	β (PWR-5)	4.0E-8
	ε (PWR-7)	2.8E-6
T_2MQFH -	γ (PWR-2)	2.4E-6
	β (PWR-4)	3.4E-8
	ε (PWR-6)	2.4E-6
S_3FH -	γ (PWR-2)	2.0E-6
	β (PWR-4)	2.9E-8
	ε (PWR-6)	2.0E-6
S_2FH -	α (PWR-1)	1.2E-8
	β (PWR-4)	8.6E-9
	ε (PWR-6)	9.5E-7
T_2KMU -	γ (PWR-3)	3.8E-6
	β (PWR-5)	5.6E-8
	ε (PWR-7)	3.8E-6

Note: Only affected accident sequences containing HHMAN, HPMAN1, HPRSCM, or WXCM exhibit a change in frequency from the base to the adjusted case to two significant figures and are shown here. The contributions from the non-dominant minimal cut sets are assumed to decrease in the same proportions as those from the dominant minimal cut sets in all affected accident sequences.

TABLE 1. (contd)

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 8.0E-8/py

PWR-2 = 5.7E-6/py

PWR-3 = 1.5E-5/py

PWR-4 = 8.9E-8/py

PWR-5 = 2.6E-7/py

PWR-6 = 6.9E-6/py

PWR-7 = 2.0E-5/py

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

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

$\bar{F}^* = 4.819E-5/py$

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

$W^*_{PWR} = 110.4$ man-rem/py

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

$(\Delta\bar{F})_{PWR} = 8.7E-7/py$ $(\Delta\bar{F})_{BWR} = 3.9E-7/py$[(a)]

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

$(\Delta W)_{PWR} = 5.9$ man-rem/py $(\Delta W)_{BWR} = 7.1$ man-rem/py[(a)]

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.2E+4	7.1E+6	0

(a) See Attachment 1.

ATTACHMENT 1

The RSSMAP studies for Oconee 3 and Grand Gulf 1 give total core-melt frequencies ($\bar{F}_0$) of 8.2E-5/py and 3.7E-5/py, respectively, for these plants (Andrews et al. 1983). Using the original release category frequencies and the public dose factors (Appendix D of PNL-4297), one obtains total public risks (W_0) of 207 man-rem/py and 250 man-rem/py, respectively, for Oconee and Grand Gulf. For the purpose of scaling the base-case, affected core-melt frequency ($\bar{F}$) and public risk (W), and the reductions in the core-melt frequency ($\delta\bar{F}$) and public risk (ΔW) from Oconee to Grand Gulf, the following are assumed:

$$\left.\begin{array}{l} \bar{F}_{BWR}/\bar{F}_{PWR} \\ (\Delta\bar{F})_{BWR}/(\Delta\bar{F})_{PWR} \end{array}\right\} = (\bar{F}_0)_{BWR}/(\bar{F}_0)_{PWR}$$

$$\left.\begin{array}{l} W_{BWR}/W_{PWR} \\ (\Delta W)_{BWR}/(\Delta W)_{PWR} \end{array}\right\} = (W_0)_{BWR}/(W_0)_{PWR}$$

Using the original values of $\bar{F}_0$ and W_0 for Oconee and Grand Gulf, the scaling equations become:

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

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

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

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

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Safety System Status Monitoring (I.D.3)

2. Affected Plants (N):

   All 71 completed plants (47 PWRs and 24 BWRs).

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

	$\bar{T}$ (yr)
47 operating PWRs	27.7
24 operating BWRs	25.2
All 71 operating LWRs	26.9

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

   PWR (19,860 man-rem)(8.7E-7/py) = 1.7E-2 man-rem/py

   BWR (19,860 man-rem)(3.9E-7/py) = 7.7E-3 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
27	9.2E+3	0

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

   Installation = (17 man-mo/plant)(40 man-hr/man-wk)
   = (1 man-yr/12 man-mo)(44 man-wk/man-yr)
   = 2490 man-hr/plant

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

   An average dose rate of 10 mR/hr is assumed.

   D = (2490 man-hr/plant)(D.010 R/hr)
   = 24.9 man-rem/plant (same for PWRs and BWRs)

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

   ND = (71)(24.9 man-rem/plant) = 1.77E+3 man-rem

TABLE 2. (contd)

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

   Maintenance = 1 man-wk/py = 40 man-hr/py

10. Per-Plant Occupational Dose Increase for SIR Operation and Maintenance (D_0):

    An average dose of 10 mR/hr is assumed.

    D_0 = (40 man-hr/py)(0.010 R/hr) = 0.40 man-rem/py

    (Same for PWRs and BWRs)

11. Total Occupational Dose Increase for SIR Operation and Maintenance ($N\bar{T}D_0$):

    $N\bar{T}D_0$ = (0.40 man-rem/py)(71)(26.9 yr) = 764 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.5E+3	7.5E+3	8.3E+2

3.0 SAFETY ISSUE COSTS

The PNL staff, with the aid of consultants, estimated the costs associated with the implementation and operation of this SIR. The industry cost per plant for implementation of this issue is equal to the installation cost of the safety status monitoring system. The installation cost is the sum of labor, equipment, and replacement power costs. Equipment and labor costs were estimated specifically for this issue. As previously discussed, it was assumed that the installation of this system would be undertaken only as a part of a complete control room modification (Item I.D.1). Thus, the replacement power costs likely to be required as part of the control room modification are not included for this issue, but it is anticipated that they would be quite substantial if this issue were undertaken separately.

The industry equipment and labor costs per plant for implementation of this SIR are presented below.

Equipment:	Cable - ~30 miles @ $6.00/100 L.F.	$ 9,500
	Electrical Penetration Limitations	300,000
	Cable Tray and Additional Termination	10,000
	Intermediate Logic Panel	100,000
	Control Room Alarms	10,000
	Total Equipment	$429,500
Labor and Other Costs:		
	Design Labor = 12 man-mos.	$100,000
	Installation Labor = 17 man-mos.	142,000
	QA Cost	40,000
	Class IE Qualification	50,000
	Total Labor	$332,000
Total Implementation Costs to Industry/Plant		$761,500

Maintenance of the safety status monitoring system by industry was estimated at one man-week per plant per year. All of the above costs would be the same for both BWRs and PWRs.

Development of this SIR by the NRC was estimated to take 0.5 man-years as given in the TMI Action Plan, NUREG-0660 (NRC 1980). The review of industry implementation of the SIR was estimated to take 4 man-weeks per plant. The NRC labor to review operation/maintenance of the SIR was estimated at 0.5 man-weeks per plant per year of operation.

Table 3 summarizes the results of the industry and NRC cost analyses.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Safety System Status Monitoring (I.D.3)

2. Affected Plants (N):

   All 71 completed plants (47 PWRs and 24 BWRs)

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

	$\bar{T}$ (yr)
47 operating PWRs	27.7
24 operating BWRs	25.2
All 71 operating LWRs	26.9

TABLE 3. (contd)

Industry Costs (Steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, $\Delta(\overline{FA})$:

PWR ($1.65E+9)(8.7E-7/py) = $1.4E+3/py

BWR ($1.65E+9)(3.9E-7/py) = $6.4E+2/py

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

Best Estimate	Upper Bound	Lower Bound
$2.3E+6	$7.6E+8	0

6. Per-Plant Industry Resources for SIR Implementation:

Equipment per plant: Cable, cable trays, logic panel, control room alarms, and electrical penetration limitations

Labor per plant: 12 man-months of design labor and 17 man-months of installation labor

QA and Class IE qualification costs are costed directly.

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

I = $7.61E+5/plant (Same for BWRs and PWRs)

(See text, Section 3.0 for details)

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

NI = (71)($7.61E+5/plant) = $5.40E+7

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

1 man-wk/py

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_O):

I_O = (1 man-wk/py)($2270/man-wk) = $2270/py

(Same for BWRs and PWRs)

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

$N\bar{T}I_O$ = ($2270/py)(71)(26.9 yr) = $4.34E+6

TABLE 3. (contd)

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$5.8E+7	\$8.5E+7	\$3.1E+7

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

0.5 man-yr = 22 man-wk

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

C_D = (22 man-wk)(\$2270/man-wk) = \$5.0E+4

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

4 man-wk/plant

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

(4 man-wk/plant)(\$2270/man-wk) = \$9.08E+3/plant

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

NC = (71)(\$9080/plant) = 6.45E+5

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

0.5 man-wk/py

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

C_o = (0.5 man-wk/py)(\$2270/man-wk) = \$1.14E+3/py

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

$N\bar{T}C_o$ = (\$1140/py)(71)(26.9 yr) = \$2.18E+6

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$2.9E+6	\$4.0E+6	\$1.8E+6

REFERENCES

Andrews, W. B., et. al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development, NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

Brown, R. G., and J. Vonherrmann. 1981. Light Water Reactor Engineering Safety Features Status Monitoring: Final Report. NUREG/CR-2278, prepared for the U.S. Nuclear Regulatory Commission by EG&G, Inc., Idaho Falls, Idaho.

U.S. NRC. 1980. NRC Action Plan Developed as a Result of the TMI-2 Accident. NUREG-0660, Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: I.D.4, Control Room Design Standard

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

The purpose of this issue is to develop guidance on the design of control rooms to incorporate human factors. The proposed resolution for this issue is to construct control rooms in accordance with an NRC regulatory guide (to be issued in FY86) at plants to be completed after 1986.

AFFECTED PLANTS	BWR:	Operating = 0	Planned = 5
	PWR:	Operating = 0	Planned = 10

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	1.9E+3
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	5.7

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	1.0
SIR Operation/Maintenance =	0
Total of Above =	1.0
Accident Avoidance =	0.48
NRC COSTS:	
SIR Development =	0.30
SIR Implementation Support =	0.09
SIR Operation/Maintenance Review =	0
Total of Above =	0.39

CONTROL ROOM DESIGN STANDARD

ISSUE I.D.4

1.0 SAFETY ISSUE DESCRIPTION

The purpose of this issue is to develop guidance on the design of control rooms to incorporate human factors considerations. To meet this goal, a proposed regulatory guide is to be completed and issued, based on an evaluation of industry standards (IEEE 566 and 567) and including consideration of the applicability of these standards to plants under construction.

The IEEE subcommittees are working in three areas. One working group is preparing a document on human performance evaluation. This document will be a guide that recommends methods of evaluating the control room with regard to human factors deficiencies. The document will 1) provide ways to evaluate proposed solutions to these deficiencies, and 2) discuss ways of evaluating these changes prior to the installation of any hardware changes. Another IEEE working group is preparing a white paper that deals with a human engineering plan to include human factors principles throughout all phases of the control room. This plan is to cover both existing and new control rooms. A third subcommittee is preparing a revision to IEEE 566 that covers the issues of control room facilities. This revision is expected to be completed in 1983.

The NRC regulatory guide is to be issued in FY86. It will identify criteria for control room design and be more comprehensive and up-to-date than the current guidelines.

PROPOSED RESOLUTION

The proposed safety issue resolution (SIR) for this study is to construct control rooms in accordance with the NRC regulatory guide at plants to be completed after 1986.

AFFECTED PLANTS

This issue affects only those PWRs and BWRs to be completed after 1986.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose are estimated in this section. Results are summarized in Tables 1 and 2, respectively.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Control Room Design Standard (I.D.4)

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

   All PWRs and BWRs to be completed after CY86

	N	$\bar{T}$ (yr)
PWRs	10	30
BWRs	5	30

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf 1 - representative BWR

4. Parameters Affected by SIR:

   All those parameters requiring direct operator actions are considered affected.

   PWR: HHMAN, HPMAN, HPMAN1, HPRSCM, WXCM

   BWR: C, OP

5. Base-Case Values for Affected Parameters:

   All affected parameters have the original values as given in Tables A.4 (PWR) and B.4 (BWR) (Andrews et al. 1983).

6. Affected Accident Sequences and Base-Case Frequencies:

Sequence			Base-Case Frequency (1/py)
PWR:			
T_2MLU	-	γ (PWR-3)	5.3E-7
		β (PWR-5)	7.8E-9
		ε (PWR-7)	5.3E-7
T_1MLU	-	γ (PWR-3)	9.8E-7
		β (PWR-5)	1.4E-8
		ε (PWR-7)	9.8E-7

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

Sequence			Base-Case Frequency (1/py)
PWR: (contd)			
$T_1(B_3)MLU$	-	γ (PWR-3)	1.1E-6
		β (PWR-5)	1.6E-8
		ε (PWR-7)	1.1E-6
T_2MQH	-	γ (PWR-3)	2.3E-6
		β (PWR-5)	3.3E-8
		ε (PWR-7)	2.3E-6
S_3H	-	γ (PWR-3)	2.0E-6
		β (PWR-5)	2.8E-8
		ε (PWR-7)	2.0E-6
T_2MQFH	-	γ (PWR-2)	2.3E-6
		β (PWR-4)	3.3E-8
		ε (PWR-6)	2.3E-6
S_3FH	-	γ (PWR-2)	2.0E-6
		β (PWR-4)	2.8E-8
		ε (PWR-6)	2.0E-6
S_2FH	-	α (PWR-1)	1.2E-8
		β (PWR-4)	8.8E-9
		ε (PWR-6)	9.6E-7
T_2KMU	-	γ (PWR-3)	3.9E-6
		β (PWR-5)	5.7E-8
		ε (PWR-7)	3.9E-6
BWR:			
T_1PQE	-	γ (BWR-3)	1.4E-8
		δ (BWR-4)	1.4E-8
$T_{23}PQE$	-	γ (BWR-3)	2.3E-7
		δ (BWR-4)	2.3E-7
$T_{23}C$	-	δ (BWR-2)	5.4E-6

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

Sequence		Base-Case Frequency (1/py)
BWR: (contd)		
T_1QUV -	γ (BWR-3)	1.2E-7
	δ (BWR-4)	1.2E-7

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 1.2E-8/py	BWR-2=5.4E-6/py
PWR-2 = 4.2E-6/py	BWR-3=3.7E-7/py
PWR-3 = 1.1E-5/py	BWR-4=3.7E-7/py
PWR-4 = 7.0E-8/py	
PWR-5 = 1.6E-7/py	
PWR-6 = 5.2E-6/py	
PWR-7 = 1.1E-5/py	

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

$\bar{F}_{PWR}$ = 3.1026E-5/py $\bar{F}_{BWR}$ = 6.12E-6/py

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

W_{PWR} = 8.08E+1 man-rem/py W_{BWR} = 4.045E+1 man-rem/py

10. Adjusted-Case Values for Affected Parameters:

It is assumed for this study that the probability of operator error for the affected parameters is decreased by 3% based on the SIR. This 3% value is based on comparison with previously evaluated issues involving operator error. This decrease is evident to two significant figures only for the following parameters:

Parameter	Affected Value
PWR:	
HHMAN	9.7E-2
HPMAN1	9.7E-2
HPRSCM	2.9E-3
WXCM	2.9E-3

TABLE 1. (contd)

10. Adjusted-Case Values for Affected Parameters (contd):

Parameter	Affected Value
BWR:	
C	7.5E-7

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Sequence		Adjusted-Case Frequency (1/py)
PWR:		
T_2MQH	γ (PWR-3)	2.2E-6
	β (PWR-5)	3.2E-8
	ε (PWR-7)	2.2E-6
S_3H	γ (PWR-3)	1.9E-6
	β (PWR-5)	2.8E-8
	ε (PWR-7)	1.9E-6
T_2MQFH	γ (PWR-2)	2.2E-6
	β (PWR-4)	3.2E-8
	ε (PWR-6)	2.2E-6
S_3FH	γ (PWR-2)	1.9E-6
	β (PWR-4)	2.8E-8
	ε (PWR-6)	1.9E-6
S_2FH	α (PWR-1)	1.2E-8
	β (PWR-4)	8.5E-9
	ε (PWR-6)	9.3E-7
T_2KMU	γ (PWR-3)	3.8E-6
	β (PWR-5)	5.5E-8
	ε (PWR-7)	3.8E-6
BWR:		
$T_{23}C$	δ (BWR-2)	5.3E-6

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

Note: Only affected accident sequences containing HHMAN, HPMAN1, HPRSCM, and WXCM for PWRs and C for BWRs exhibit a change in frequency from the base to the adjusted case to two significant figures and are shown here.

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1	= 1.2E-8/py	BWR-2	= 5.3E-6/py
PWR-2	= 4.1E-6/py	BWR-3	= 3.7E-7/py
PWR-3	= 1.0E-5/py	BWR-4	= 3.7E-7/py
PWR-4	= 6.8E-8/py		
PWR-5	= 1.5E-7/py		
PWR-6	= 5.0E-6/py		
PWR-7	= 1.0E-5/py		

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

$\bar{F}^*_{PWR}$ = 3.0132E-5/py $\qquad$ F^*_{BWR} = 5.98E-6/py

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

W^*_{PWR} = 7.49E+1 man-rem/py $\qquad$ W^*_{BWR} = 3.974+1 man-rem/py

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

PWR: 8.9E-7/py $\qquad$ BWR: 1.4E-7/py

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

PWR: 5.9 man-rem/py $\qquad$ BWR: 0.71 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.9E+3	9.1E+5	0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Control Room Design Standard (I.D.4)

TABLE 2. (contd)

2. Affected Plants (N):

All PWRs and BWRs to be completed after 1986 (10 PWRs and 5 BWRs).

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

Average remaining life is 30 years for both PWRs and BWRs.

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

PWR: (19,900 man-rem)(8.9E-7/py) = 1.8E-2 man-rem/py

BWR: (19,900 man-rem)(1.4E-7/py) = 2.8E-3 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
5.7	1.2E+3	0

6-12. Steps Related to Occupational Dose Increase for SIR Implementation and Operation/Maintenance, and Total Occupational Dose Increase:

Construction of control rooms at new plants involves no radiation zone work. The operation and maintenance of a control room involves no radiation zone work. Therefore, there is no occupational dose increase due to this SIR, and $D = D_O = G = 0$.

3.D SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. Results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Control Room Design Standard (I.D.4)

2. Affected Plants (N):

All PWRs and BWRs to be completed after 1986 (10 PWRs and 5 BWRs).

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

Average remaining life is 30 years for both PWRs and BWRs.

TABLE 3. (contd)

Industry Costs (Steps 4 through 12)

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

PWR: ($1.65E+9)(8.9E-7/py) = $1.5E+3/py

BWR: ($1.65E+9)(1.4E-7/py) = $2.3E+2/py

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

Best Estimate	Upper Bound	Lower Bound
$4.8E+5	$1.0E+8	0

6. Per-Plant Industry Resources for SIR Implementation:

Estimates are included directly in the next step.

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

For purposes of this analysis, it is assumed that 1) the industry will continue to modify plant control rooms for human factors considerations through 1986 independent of this issue, and 2) these modifications are already planned. Therefore, no costs are assigned to this issue from those efforts. For those plants completed between 1987 and 1990, the cost to modify the control room to meet the standard is assumed to be $100,000 per plant. This figure is based on the assumption that the control rooms will be designed according to draft standards until 1986 and that only minor changes will then need to be made in the control room designs when the final standards come out. For those plants completed after 1990, it is assumed that the cost to design new control rooms upgraded to the standard will be part of the basic plant cost.

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

There are 10 plants to be completed between 1987 and 1990. Therefore, the total industry cost for SIR implementation is $1.0E+6.

9-11. Steps Related to SIR Operation and Maintenance:

It is assumed that the operation and maintenance of newly designed control rooms requires no additional expenditure beyond that for present control rooms ($I_0 = 0$).

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$1.0E+6	$1.5E+6	$5.0E+5

TABLE 3. (contd)

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Estimates are included directly in the next step.

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

It is assumed for this study that $300,000 will be needed to develop the regulatory guide that will define the new control room design standard (C_D = \$3.0E+5).

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

It is assumed for this study that the NRC will spend 4 man-weeks/plant checking the control room design modifications that will be required at the 10 plants built between 1987 and 1990. For those plants completed in or beyond 1990, the NRC labor to check to control room designs will be that normally expended in initial design review. No additional labor is anticipated from the SIR.

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

C = (4 man-wk/plant)($2270/man-wk) = $9.1E+3/plant

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

NC = ($9.1E+3/plant)(10 plants) = $9.1E+4

18-20. Steps Related to NRC Cost for Review of SIR Operation and Maintenance:

It is assumed for this study that there will be no additional NRC cost for review of control room operation and maintenance beyond what may be required currently (C_o = 0).

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$3.9E+5	$5.5E+5	$2.3E+5

REFERENCE

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: I.D.5(3-5), Control Room Design: Improved Control Room Instrumentation Research

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

The overall objective of Task I.D, "Control Room Design," is to improve the ability of nuclear power plant control room operators to prevent and cope with accidents if they occur. Part 5 of this task is aimed at developing new instrumentation to enhance the performance of the control room operator. The proposed resolution is to implement an advanced diagnostic system, including a continuous on-line surveillance system and an in-vessel, liquid-level detection system.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 5.1E+4

OCCUPATIONAL DOSES:

SIR Implementation =	14
SIR Operation/Maintenance =	480
Total of Above =	490
Accident Avoidance =	280

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:

SIR Implementation =	270
SIR Operation/Maintenance =	-730
Total of Above =	-460
Accident Avoidance =	23

NRC COSTS:

SIR Development =	2.0
SIR Implementation Support =	0.65
SIR Operation/Maintenance Review =	0
Total of Above =	2.6

CONTROL ROOM DESIGN--

IMPROVED CONTROL ROOM INSTRUMENTATION RESEARCH

ISSUE I.D.5(3-5)

1.0 SAFETY ISSUE DESCRIPTION

The overall objective of Task I.D.5, Control Room Design, is to improve the ability of nuclear power plant control room operators to prevent and cope with accidents, if they occur, by improving the availability and presentation of information. Part 5 of this task is aimed at developing new instrumentation to enhance the performance of the control room operator.

The purpose of subpart 3 of I.D.5 is to construct and test a continuous on-line reactor surveillance system, based on noise-diagnostic and pattern-recognition techniques, to evaluate selected plant signals for abnormalities in operation. The purpose of subpart 4 of I.D.5 is to assess the reliability of light water reactor (LWR) in-vessel, liquid-level detection techniques. Subpart 4 is considered complete as of the end of FY83. The purpose of subpart 5 of I.D.5 is to provide the technical basis for developing design requirements, developing review criteria, and assessing the need, feasibility and adequacy of advanced diagnostic systems. This involves research underway at the Idaho Nuclear Engineering Laboratory to evaluate Safety Parameter Display Systems (SPDS). Reactor data generated at the Loss-of-Fluid Test (LOFT) facility are used to generate various monitor screen displays.

PROPOSED RESOLUTION

The proposed safety issue resolution (SIR) for this study is to incorporate an advanced diagnostic system in each LWR. This advanced diagnostic system includes a continuous on-line reactor surveillance system and more reliable in-vessel, liquid-level detection system. The effectiveness of this advanced diagnostic system in aiding control room operators would be greater if implemented with a complete control room design. However, for this study, the effectiveness of the advanced diagnostic system excludes the implementation of a complete control room design.

AFFECTED PLANTS

This issue affects all PWRs and BWRs, both completed and under construction.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose are estimated in this section and summarized in Tables 1 and 2, respectively.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Control Room Design--Improved Control Room Instrumentation Research [I.D.5(3-5)].

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

   All PWRs and BWRs are assumed to be affected.

	N	$\bar{T}$ (yr)
PWR	90	28.8
BWR	44	27.4

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf 1 - representative BWR

4. Parameters Affected by SIR:

   All parameters requiring direct operator action are assumed to be affected. Also, the SIR will result in reduced transient frequencies because operator actions (based on diagnostic information) will presumably terminate transients before the need for shutdown. The reduced transient frequencies are calculated in Attachment 1. The reduced transient frequencies divided by the total transient frequencies (9.80/py for PWRs and 8.90/py for BWRs) give the percent transient reductions. Thus, the parameters T_2 and T_3 for PWRs and T_{23} for BWRs are also considered affected parameters for this study.[a]

   PWR: HHMAN, HPMAN, HPMAN1, HPRSCM, WXCM, T_2, T_3

   BWR: C, OP, T_{23}

5. Base-Case Values for Affected Parameters:

   All parameters have the original values as given in Tables A.4 (PWR) and B.4 (BWR) in Andrews et al. 1983.

(a) Transients induced by loss of offsite power (parameters T, for both PWRs and BWRs) are assumed to remain unaffected by SIR.

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies:

Sequence		Base-Case Frequency (1/py)
PWR:		
T_2MLU -	γ (PWR-3)	5.3E-7
	β (PWR-5)	7.8E-9
	ϵ (PWR-7)	5.3E-7
T_1MLU -	γ (PWR-3)	9.8E-7
	β (PWR-5)	1.4E-8
	ϵ (PWR-7)	9.8E-7
$T_1(B_3)MLU$ -	γ (PWR-3)	1.1E-6
	β (PWR-5)	1.6E-8
	ϵ (PWR-7)	1.1E-6
T_2MQH -	γ (PWR-3)	5.2E-6
	β (PWR-5)	7.6E-8
	ϵ (PWR-7)	5.2E-6
S_3H -	γ (PWR-3)	2.0E-6
	β (PWR-5)	2.8E-8
	ϵ (PWR-7)	2.0E-6
T_2MQFH -	γ (PWR-2)	2.4E-6
	β (PWR-4)	3.4E-8
	ϵ (PWR-6)	2.4E-6
S_3FH -	γ (PWR-2)	2.0E-6
	β (PWR-4)	2.8E-8
	ϵ (PWR-6)	2.0E-6
S_2FH -	α (PWR-1)	1.2E-8
	β (PWR-4)	8.8E-9
	ϵ (PWR-6)	9.6E-7

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

Sequence		Base-Case Frequency (1/py)
T_2MLUO -	γ (PWR-3)	4.0E-6
	β (PWR-5)	5.8E-8
	ε (PWR-7)	4.0E-6
T_2KMU -	γ (PWR-3)	3.9E-6
	β (PWR-5)	5.7E-8
	ε (PWR-7)	3.9E-6
T_3MLUO -	γ (PWR-3)	5.3E-7
	β (PWR-5)	7.8E-9
	ε (PWR-7)	5.3E-7
T_2MQD -	γ (PWR-3)	7.6E-7
	β (PWR-5)	1.1E-8
	ε (PWR-7)	7.6E-7
BWR:		
$T_{23}PQI$ -	α (BWR-1)	3.7E-8
	δ (BWR-2)	3.7E-6
T_1PQE -	γ (BWR-3)	1.4E-8
	δ (BWR-4)	1.4E-8
$T_{23}PQE$ -	γ (BWR-3)	2.7E-7
	δ (BWR-4)	2.7E-7
$T_{23}QW$ -	δ (BWR-2)	1.1E-5
$T_{23}C$ -	δ (BWR-2)	5.4E-6
T_1QUV -	γ (BWR-3)	1.2E-7
	δ (BWR-4)	1.2E-7

TABLE 1. (contd)

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 1.2E-8/py	BWR-1 = 3.7E-8/py
PWR-2 = 4.3E-6/py	BWR-2 = 2.0E-5/py
PWR-3 = 1.9E-5/py	BWR-3 = 4.0E-7/py
PWR-4 = 7.2E-8/py	BWR-4 = 4.0E-7/py
PWR-5 = 2.8E-7/py	
PWR-6 = 5.3E-6/py	
PWR-7 = 1.9E-5/py	

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

$\bar{F}_{PWR}$ = 4.78E-5/py $\bar{F}_{BWR}$ = 2.08E-5/py

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

W_{PWR} = 1.25E+2 man-rem/py W_{BWR} = 1.44E+2 man-rem/py

10. Adjusted-Case Values for Affected Parameters:

It is assumed for this study that the probability of operator error for the affected parameters involving direct operator action decreased by 2% due to the SIR. This 2% value is based upon comparison with previously evaluated issues involving operator error.

Based on the percent transient reductions discussed in Step 4 and Attachment 1, T_2 and T_3 for PWRs are reduced by 9.4%, and T_{23} for BWRs is reduced by 12%. These are calculated as follows:

[(4.63 transients/py)(0.80)(0.25)]/(9.80 transients/py) = 0.094, or 9.4% for PWRs

[(5.20 transients/py)(0.80)(0.25)]/(8.90 transients/py = 0.12, or 12% for BWRs

One effect of the on-line monitoring systems is related to reliability. System monitoring can indicate the need for maintenance, thereby enabling repairs to be effected before failure. This will reduce the frequency of transients, hence reducing demands on the safety systems and reducing core-melt frequency. This effect is assumed to be represented in the core-melt reduction indicated by transient reduction.

TABLE 1. (contd)

10. Adjusted-Case Values for Affected Parameters (contd):

The adjusted-case values of the affected parameters are as follows, showing only those where the decrease from the base case is evident to two significant figures:

Parameter	Adjusted Value
PWR:	
HHMAN	9.8E-2
HPMAN1	9.8E-2
HPRSCM	2.9E-3
WXCM	2.9E-3
T_2	2.7/py
T_3	3.6/py
BWR:	
C	7.5E-7
T_{23}	6.2/py

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Sequence		Adjusted-Case Frequency (1/py)
PWR:		
T_2MLU -	γ (PWR-3)	4.8E-7
	β (PWR-5)	7.0E-9
	ε (PWR-7)	4.8E-7
T_2MQH -	γ (PWR-3)	4.6E-6
	β (PWR-5)	6.7E-8
	ε (PWR-7)	4.6E-6
S_3H -	γ (PWR-3)	1.9E-6
	β (PWR-5)	2.8E-8
	ε (PWR-7)	1.9E-6

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

Sequence		Adjusted-Case Frequency (1/py)
T_2MQFH -	γ (PWR-2)	2.1E-6
	β (PWR-4)	3.0E-8
	ε (PWR-6)	2.1E-6
S_3FH -	γ (PWR-2)	1.9E-6
	β (PWR-4)	2.8E-8
	ε (PWR-6)	1.9E-6
S_2FH -	α (PWR-1)	1.2E-8
	β (PWR-4)	8.5E-9
	ε (PWR-6)	9.3E-7
T_2MLUO -	γ (PWR-3)	3.6E-6
	β (PWR-5)	5.2E-8
	ε (PWR-7)	3.6E-6
T_2KMU -	γ (PWR-3)	3.4E-6
	β (PWR-5)	5.0E-8
	ε (PWR-7)	3.4E-6
T_3MLUO	γ (PWR-3)	4.8E-7
	β (PWR-5)	7.0E-9
	ε (PWR-7)	4.8E-7
T_2MQD -	γ (PWR-3)	6.9E-7
	β (PWR-5)	1.0E-8
	ε (PWR-7)	6.9E-7

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

Sequence		Adjusted-Case Frequency (1/py)
BWR:		
$T_{23}PQI$ -	α (BWR-1)	3.3E-8
	δ (BWR-2)	3.3E-6
T_1PQE -	γ (BWR-3)	1.4E-8
	δ (BWR-4)	1.4E-8
$T_{23}PQE$ -	γ (BWR-3)	2.4E-7
	δ (BWR-4)	2.4E-7
$T_{23}QW$ -	δ (BWR-2)	9.6E-6
$T_{23}C$ -	δ (BWR-2)	4.7E-6
T_1QUV -	γ (BWR-3)	1.2E-7
	δ (BWR-4)	1.2E-7

Note: Only affected accident sequences containing HHMAN, HPMAN1, HPRSCM, WXCM, T_2, T_3 for PWRs and C, T_{23} for BWRs exhibit a change in frequency from the base to the adjusted case to two significant figures and are shown here.

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 1.2E-8/py	BWR-1 = 3.3E-8/py
PWR-2 = 3.9E-6/py	BWR-2 = 1.8E-5/py
PWR-3 = 1.7E-5/py	BWR-3 = 3.7E-7/py
PWR-4 = 6.6E-8/py	BWR-4 = 3.7E-7/py
PWR-5 = 2.5E-7/py	
PWR-6 = 4.9E-6/py	
PWR-7 = 1.7E-5/py	

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

$\bar{F}^*_{PWR}$ = 4.36E-5/py $\qquad$ $\bar{F}_{BWR}$ = 1.83E-5/py

TABLE 1. (contd)

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

W^*_{PWR} = 1.12E+2 man-rem/py W^*_{BWR} = 1.30E+2 man-rem/py

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

$\Delta\bar{F}_{PWR}$ = 4.2E-6/py $\Delta\bar{F}_{BWR}$ = 2.5E-6/py

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

ΔW_{PWR} = 13 man-rem/py ΔW_{BWR} = 14 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
5.1E+4	1.5E+7	0

ATTACHMENT 1

The reduction in the number of unscheduled outages related to transient shutdowns is calculated below. Transients that are assumed to be potentially affected by operator actions (intended to prevent the need for shutdown) are chosen from those listed in a recent EPRI analysis (McClymont and Poehlman 1982). The transients listed in this EPRI review are based on the historical record. These transients were categorized by initiating cause, and transient frequencies were developed from statistical analyses of these data. Forty-one categories of PWR transients and 37 categories of BWR transients were identified. These results have been reviewed in this study to identify categories of transients that could potentially be affected by operator actions directed to prevent the need for shutdown. The transients that are assumed to potentially contribute to shutdown frequencies are listed below.

TRANSIENTS LEADING TO SHUTDOWN THAT ARE ASSUMEDLY AFFECTED BY APPROPRIATE OPERATOR ACTIONS

EPRI Category		Title	Frequency (Events/py)
PWR	4	Leakage from Control Rods through Drive Mechanism	0.2
	5	Leakage from Primary System	0.08
	6	Low Pressurizer Pressure	0.03
	7	Pressurizer Leakage	0.01
	8	High Pressurizer Pressure	0.03
	9	Inadvertent Safety Injection Signal	0.06
	10	Containment Pressure Problems	0.01
	11	CVCS Malfunction--Boron Dilution	0.04
	12	Pressure/Temperature/Power Imbalance--Rod Position Error	0.16
	15	Loss or Reduction in Feedwater Flow (1 Loop)	1.88
	19	Increase in Feedwater Flow (1 Loop)	0.69
	20	Increase in Feedwater Flow (All Loops)	0.01
	21	Feedwater Flow Instability--Operator Error	0.15
	22	Feedwater Flow Instability--Miscellaneous Mechanical Causes	0.21
	23	Loss of Condensate Pumps (1 Loop)	0.08
	25	Loss of Condenser Vacuum	0.02
	26	Steam Generator Leakage	0.04

ATTACHMENT 1 (contd)

EPRI Category		Title	Frequency (Events/py)
	27	Condenser Leakage	0.05
	28	Miscellaneous Leakage in Secondary System	0.08
	36	Pressurizer Spray Failure	0.04
	38	Spurious Trips--Cause Unknown	0.14
	40	Manual Trip--No Transient Condition	0.62
		Total =	4.63
BWR	8	Loss of Normal Condenser Vacuum	0.45
	9	Pressure Regulator Fails Open	0.17
	10	Pressure Regulator Fails Closed	0.17
	12	Turbine Bypass Fails Open	0.06
	13	Turbine Bypass or Control Valves Cause Increased Pressure (Closed)	0.42
	14	Recirculation Control Failure--Increasing Flow	0.23
	15	Recirculation Control Failure--Decreasing Flow	0.10
	16	Trip of One Recirculation Pump	0.08
	20	Feedwater--Increasing Flow at Power	0.16
	21	Loss of Feedwater Heater	0.04
	23	Trip of One Feedwater Pump (or Condensate Pump)	0.14
	24	Feedwater--Low Flow	0.52
	25	Low Feedwater Flow During Startup or Shutdown	0.21
	26	High Feedwater Flow During Startup or Shutdown	0.07
	29	Inadvertent Insertion of Rod or Rods	0.12
	35	Spurious Trip via Instrumentation, RPS Fault	1.21
	36	Manual Scram--No Out of Tolerance Condition	1.05
		Total =	5.20

The basis for choosing these transients is as follows. Either the detection time leading up to a transient or the time from the transient occurrence to shutdown was perceived to be longer than 30 minutes, enabling the advanced diagnostic system to diagnose the problem and provide possible solutions. Transients chosen were those for which it was perceived that operator actions could conceivably prevent the need for shutdown.

ATTACHMENT 1 (contd)

For the purpose of this study, it was assumed that the operator could only respond with actions to 80 percent of the transients listed above that would occur during the remaining lifetimes of the subject power plants. Of this 80 percent, only 25 percent of the operator's actions is assumed to prevent the need for shutdowns. The average shutdown is assumed to last 0.75 day. Therefore, the reduction in unscheduled outages is calculated as follows:

PWR: (4.63 transients/py)(0.80)(0.25)(0.75 day/shutdown)

= 0.69 day/py (reduction)

BWR: (5.20 transients/py)(0.80)(0.25)(0.75 day/shutdown)

= 0.78 day/py (reduction)

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

    Control Room Design--Improved Control Room Instrumentation Research [I.D.5(3-5)].

2. Affected Plants (N):

PWR:	operating	47
	planned	43
BWR:	operating	24
	planned	20
		134

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

		$\bar{T}$ (yr)
PWR:	41 operating	27.7
	43 planned	30
	All 90	28.8
BWR:	24 operating	25.2
	20 planned	30
	All 44	27.4

4. Per-Plant Occupational Dose Reduction Due to Accident Avoidance, $\Delta(\overline{FD}_R)$:

    PWR: (19,900 man-rem)(4.2E-6/py) = 8.4E-2 man-rem/py

    BWR: (19,900 man-rem)(2.5E-6/py) = 5.0E-2 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.8E+2	1.8E+4	0

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

    It is assumed that 80 man-hours are required to install instrumentation in radiation zones on operating LWRs.

TABLE 2. (contd)

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

Since installation will involve labor outside of containment, a dose rate of 2.5 mR/hr is assumed:

$$D = (0.0025R/hr)(80 \text{ man-hr/plant}) = 0.20 \text{ man-rem/plant}$$

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

$$ND = (71)(0.20 \text{ man-rem/plant}) = 14 \text{ man-rem}$$

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

It is assumed that 50 man-hours in radiation zones are required per plant-year to operate and maintain the advanced diagnostic system above that is currently required to operate and maintain the control room instrumentation.

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

Again, a dose rate of 2.5 mR/hr is assumed.

$$D_o = (0.0025R/hr)(50 \text{ man-hr/py}) = 0.125 \text{ man-rem/py}$$

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

$$N\bar{T}D_o = [90(28.8 \text{ yr}) + 44(27.4\text{yr})](0.125 \text{ man-rem/py}) = 475 \text{ man-rem}$$

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
4.9E+2	1.5E+3	1.6E+2

3.0 SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. Results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Control Room Design--Improved Control Room Instrumentation Research [I.D.5(3-5)].

2. Affected Plants (N):

PWR:	operating	47
	planned	43
BWR:	operating	24
	planned	20
		134

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

		$\bar{T}$ (yr)
PWR:	47 operating	27.7
	43 planned	30
BWR:	24 operating	25.2
	20 planned	30

Industry Costs (Steps 4 through 12)

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

PWR: (\$1.65E+9)(4.2E-6/py) = \$6.9E+3/py

BWR: (\$1.65E+9)(2.5E-6/py) = \$4.1E+3/py

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

Best Estimate	Upper Bound	Lower Bound
\$2.3E+7	\$1.5E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

Estimates are included directly in the next step.

TABLE 3. (contd)

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

It is assumed for this study that the average cost for SIR implementation is \$2.0E+6/plant. This is broken down as follows:

- \$1.5E+6 for advanced diagnostic system and in-vessel, liquid-level detection system

- \$5.0E+5 for on-line surveillance system, including

  \$1.25E+5 for hardware cost and

  \$3.75E+5 for installation (assumed to be 3 times hardware cost).

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

NI = 134 (\$2.0E+6/plant) = \$2.68E+8

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

For purposes of this study, it is assumed that 10 man-weeks per plant-year are required for operation and maintenance of the advanced diagnostic system beyond that currently required for the control room instrumentation operation and maintenance. The advanced diagnostic system is also presumed to reduce the number of unscheduled outages over the lifetimes of the plants. As developed in Attachment 1, these reductions are 0.69 day/py and 0.78 day/py for PWRs and BWRs, respectively.

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_O):

The industry cost for operation and maintenance consists of costs for the operation and maintenance of the advanced diagnostic system and resultant cost savings over the lifetimes of all plants due to a reduction in the number of unscheduled outages (and, thus, a savings in replacement power costs) related to transient shutdowns.

I_O(PWR) = (10 man-wk/py)(\$2270/man-wk) - (0.69 day/py) (\$3.0E+5/day) = -\$1.84E+5/py

I_O(BWR) = (10 man-wk/py)(\$2270/man-wk) - (0.78 day/py) (\$3.0E+5/day) = -\$2.11E+5/py

(Negative signs indicate cost savings.)

TABLE 3. (contd)

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

$$N\bar{T}I_O = (90 \text{ PWRs})(28.8 \text{ yr})(-\$1.84E+5/\text{py}) + (44 \text{ BWRs})(27.4 \text{ yr})(-\$2.11E+5/\text{py}) = -\$7.32E+8$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$-4.6E+8	$-7.4E+7	$-8.5E+8

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Estimates are included directly in the next step.

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

Based on the average budgeted figure for 1) development of the on-line surveillance system for FY83 of $2.5E+5/yr; 2) the assessment of the reliability of in-vessel, liquid-level detection techniques for FY83 of $2.5E+5/yr; 3) development of the advanced diagnostic system for FY83 and FY84 of $7.4E+5/yr; the cost for SIR resolution is estimated to be

$$C_D = (\$2.5E+5/\text{yr})(1 \text{ yr}) + (\$2.5E+5/\text{yr})(1 \text{ yr}) + (\$7.4E+5/\text{yr})(2 \text{ yr}) = \$1.98E+6$$

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

It is assumed for this study that an average of 4 man-weeks/operating plant is required to approve and monitor hardware changes.

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

$$C = (4 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$9.1E+3/\text{plant}$$

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

$$NC = 71\ (\$9.1E+3/\text{plant}) = \$6.45E+5$$

TABLE 3. (contd)

18-20 Steps Related to Per-Plant NRC Cost for Review of SIR Operation and Maintenance:

No additional NRC review of operation and maintenance above that currently required is anticipated. Thus, $C_O = 0$.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$2.6E+6	\$3.7E+6	\$1.6E+6

REFERENCES

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG-CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

McClymont, A. S., and B. W. Poehlman. 1982. ATWS: A Reappraisal Part 3: Frequency of Anticipated Transients. EPRI-NP-2230, Electric Power Research Institute, Palo Alto, California.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: I.F.2, Detailed QA Criteria for Design, Construction and Operation

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Concern exists that several systems important to the safety of TMI Unit 2 may not have been designed, fabricated, and maintained at a level equivalent to their safety importance. This condition may exist at other plants, resulting primarily from the lack of clarity in NRC guidance for graded protection. The proposed resolution is the development and implementation of more detailed QA criteria.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	8500
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	51

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	56
SIR Operation/Maintenance =	760
Total of Above =	810
Accident Avoidance =	4.2
NRC COSTS:	
SIR Development =	0.20
SIR Implementation Support =	1.0
SIR Operation/Maintenance Review =	3.8
Total of Above =	5.0

DETAILED QA CRITERIA FOR DESIGN, CONSTRUCTION AND OPERATIONS

ISSUE I.F.2

1.0 SAFETY ISSUE DESCRIPTION

Several systems important to the safety of TMI Unit 2 were not designed, fabricated, and maintained at a level equivalent to their safety importance. This condition exists at other plants and results primarily from the lack of clarity in NRC guidance for graded protection. This situation and other quality assurance problems relating to the quality assurance organization, authority, reporting, and inspection have been identified by the various TMI accident investigations and inquiries (NRC 1980).

The overall objective of this safety issue resolution (SIR) is the improvement of the quality assurance (QA) program for design, construction, and operations to provide greater assurance that plant design, construction, and operational activities are conducted in a manner commensurate with their importance to safety. To achieve this objective, the NRC will develop more detailed criteria for various aspects of quality assurance for design, construction, and operations.

PROPOSED RESOLUTION

More detailed criteria for QA related to design, construction, and operations are proposed. The detailed criteria will consider the following (NRC 1980):

(1) Assure the independence of the organization performing the checking functions from the organization responsible for performing the tasks. For the construction phase, consider options for increasing the independence of the QA function. Include an option to require that licensees perform the entire quality assurance/quality control (QA/QC) function at construction sites. Consider using the third-party concept for accompanying the NRC review and audit and making the QA/QC personnel agents of the NRC. Consider using the Institute of Nuclear Power Operations (INPO) to enhance QA/QC independence.

(2) Include the QA personnel in the review and approval of plant operational maintenance and surveillance procedures, and quality-related procedures associated with design, construction, and installation.

(3) Include the QA personnel in all activities involved in design, construction, installation, pre-operational and startup testing, and operation.

(4) Establish criteria for determining QA requirements for specific classes of equipment, such as instrumentation, mechanical equipment, and electrical equipment.

(5) Establish qualification requirements for QA and QC personnel.

(6) Increase the size of the licensee's QA staff.

(7) Clarify that the QA program is a condition of the construction permit and operating license and that substantive changes to an approved program must be submitted to NRC for review.

(8) Compare NRC QA requirements with those of other agencies (i.e., NASA, FAA, DOD) to improve NRC requirements.

(9) Clarify organizational reporting levels for the QA organization.

(10) Clarify requirements for maintenance of "as built" documentation.

(11) Define role of QA in design and analysis activities. Obtain views on prevention of design errors from licensees, architect-engineers, and venders.

AFFECTED PLANTS

This SIR presumably affects all 134 PWRs and BWRs, both operating and planned.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose are estimated in this section. Results are summarized in Tables 1 and 2, respectively.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Detailed QA Criteria for Design, Construction, and Operation (I.F.2)

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

   All plants

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

TABLE 1. (contd)

3. Plants Selected for Analysis:

Oconee 3 - representative PWR

Grand Gulf 1 - representative BWR

4-7. Steps Related to Affected Parameters, Accident Sequences, Release Categories, and Their Base-Case Values:

Estimates of the reduction in core-melt frequency and public risk due to issue resolution are calculated directly from the base-case values. Thus, these steps (and Steps 10-14) are omitted.

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

PWR = 8.2E-5/py (original Oconee value)

BWR = 3.7E-5/py (original Grand Gulf value)

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

PWR = 2.1E+2 man-rem/py (original Oconee value)

BWR = 2.5E+2 man-rem/py (original Grand Gulf value)

10-14. Steps Related to Adjusted-Case Values of Affected Parameters, Accident Sequences, Release Categories, Core-Melt Frequency, and Public Risk:

These steps are omitted (see explanation, Steps 4-7).

15. Reduction in Core-Melt Frequency (ΔF):

The safety impact of this issue resolution is expected to be somewhat indirect since its effect is mainly on surveillance. Thus, the SIR is assumed to decrease the base-case core-melt frequency by only 1%.

PWR = (8.2E-5/py)(0.01) = 8.2E-7/py

BWR = (3.7E-5/py)(0.01) = 3.7E-7/py

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

As for the core-melt frequency, the public risk will likewise be reduced by 1%.

PWR = (2.1E+2 man-rem/py)(0.01) = 2.1 man-rem/py

BWR = (2.5E+2 man-rem/py)(0.01) = 2.5 man-rem/py

TABLE 1. (contd)

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
8.5E+3	2.5E+7	0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Detailed QA Criteria for Design, Construction, and Operation (I.F.2)

2. Affected Plants (N):

   All plants

	N
PWRs:	90
BWRs:	44
All	134

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

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

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

   PWR = (19,900 man-rem)(8.2E-7/py) = 1.6E-2 man-rem/py

   BWR = (19,9D0 man-rem)(3.7E-7/py) = 7.4E-3 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
5.1E+1	3.1E+4	0

6-12. Steps Related to Occupational Dose from SIR Implementation and Operation/Maintenance and Total Dose Increase:

The issue resolution is assumed to involve procedural changes that would not result in increased occupational dose. Thus, $D = D_0 = G = 0$.

3.0 SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. Results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

    Detailed QA Criteria for Design, Construction and Operation (I.F.2)

2. Affected Plants (N):

    All plants

		N			N
PWRs	Operating	47	BWRs	Operating	24
	Planned	43		Planned	20
	All	90		All	44

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

		$\bar{T}$ (yr)			$\bar{T}$ (yr)
PWRs	Operating	27.7	BWRs	Operating	25.2
	Planned	30		Planned	30
	All	28.8		All	27.4

For all 134 plants, $\bar{T}$ = 28.3 yr.

Industry Costs (Steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance Δ(FA):

    PWR = (\$1.65E+9)(8.2E-7/py) = \$1.4E+3/py

    BWR = (\$1.65E+9)(3.7E-7/py) = \$6.1E+2/py

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

Best Estimate	Upper Bound	Lower Bound
\$4.2E+6	\$2.5E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

    For this study the following assumptions are made:

    - For operating plants, 0.5 man-yr/plant is required to rewrite QA procedures based on more detailed QA criteria.

TABLE 3. (contd)

6. Per-Plant Industry Resources for SIR Implementation (contd):

- For planned plants, a total of 8.3 man-yr/plant of additional time is required, based on two assumptions:

  1) 0.3 man-yr/plant of additional time to write QA procedures based on the more detailed QA criteria

  2) 8 man-yr/plant of additional time for more detailed QA during the remaining years of design and construction, based on an average remaining construction time of 4 yr/plant.

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

For operating plants:

$$I = (0.5 \text{ man-yr/plant})(\$1.0E+5/\text{man-yr}) = \$5.0E+4/\text{plant}$$

For planned plants:

$$I = (8.3 \text{ man-yr/plant})(\$1.0E+5/\text{man-yr}) = \$8.3E+5/\text{plant}$$

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

$$NI = (71 \text{ operating plants})(\$5.0E+4/\text{plant}) + (63 \text{ planned plants})(\$8.3E+5/\text{plant}) = \$5.58E+7$$

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

For purposes of this study, it is assumed that the 2 man-yr/py additional time is required to carry out the more detailed QA procedures.

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_O):

$$I_O = (2 \text{ man-yr/py})(\$1.0E+5/\text{man-yr}) = \$2.0E+5/\text{py}$$

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

$$N\bar{T}I_O = (134 \text{ plants})(28.3 \text{ yr})(\$2.0E+5/\text{py}) = \$7.58E+8$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$8.1E+8	\$1.2E+9	\$4.3E+8

TABLE 3. (contd)

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

For purposes of this study, it is assumed that 2 man-years are required to develop the more detailed QA criteria.

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

C_D = (2 man-yr)($1.0E+5/man-yr) = $2.0E+5

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

For purposes of this study, it is assumed that 0.1 man-yr/plant is required to review each operating plant's rewritten QA procedures and that 0.05 man-yr/plant additional time is required to review each planned plant's more detailed QA procedures.

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

For operating plants:

C = (0.1 man-yr/plant)($1.0E+5/man-yr) = $1.0E+4/plant

For planned plants:

C = (0.05 man-yr/plant)($1.0E+5/man-yr) = $5.0E+3/plant

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

NC = (71 operating plants)($1.0E+4/plant) + (63 planned plants)
($5.0E+3/plant) = $1.03E+6

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

It is assumed for this study that 0.01 man-yr/py is required to review compliance with QA procedures.

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

C_O = (0.01 man-yr/py)($1.0E+5/man-yr) = $1.0E+3/py

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

$N\bar{T}C_O$ = (134 plants)(28.3 yr)($1.0E+3/py) = $3.79E+6

TABLE 3. (contd)

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$5.0E+6	$7.0E+6	$3.1E+6

REFERENCE

U.S. NRC. 1980. NRC Action Plan Developed as a Result of the TMI-2 Accident. NUREG-0660, U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: II.B.5(1,2)/II.B.8, Research on Phenomena Associated with Core Degradation and Fuel Melting: Behavior of Severely Damaged Fuel, Behavior of Core Melt; Severely Oamaged Core Rulemaking

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Phenomenological uncertainties exist in connection with core degradation and fuel melting. Current efforts focus on prescribing containment features to mitigate the potential release of radioactivity from core-melt accidents. Several options have been and are being considered, including core retention devices, hydrogen control features, and filtered venting of containment. Installation of the last is assumed as the resolution in this issue analysis.

AFFECTED PLANTS			
	BWR:	Operating = 24	Planned = 20
	PWR:	Operating = 26	Planned = 26

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 2.4E+5

OCCUPATIONAL DOSES:

SIR Implementation =	7800
SIR Operation/Maintenance =	1900
Total of Above =	9700
Accident Avoidance =	0

COST RESULTS ($$10^6$)

INOUSTRY COSTS:

SIR Implementation =	480
SIR Operation/Maintenance =	20
Total of Above =	500
Accident Avoidance =	0

NRC COSTS:

SIR Development =	0.60
SIR Implementation Support =	0.85
SIR Operation/Maintenance Review =	0.99
Total of Above =	2.4

RESEARCH ON PHENOMENA ASSOCIATED WITH CORE DEGRADATION AND FUEL MELTING: BEHAVIOR OF SEVERELY DAMAGED FUEL, BEHAVIOR OF CORE MELT; SEVERELY DAMAGED CORE RULEMAKING

ISSUE II.B.5(1,2)/II.B.8

In May 1980, the NRC established TMI Action Plan (TAP) Task II.B, "Consideration of Degraded or Melted Cores In Safety Review" (NRC 1980). As part of this task, subtasks II.B.5(1), "Behavior of Severely Damaged Fuel," and II.B.5(2), "Behavior of Core Melt," were defined. For this analysis, it was decided to combine those subtasks and their effects on containment, and to relate them to Task II.B.8, "Rulemaking Proceeding on Degraded Core Accidents."

1.0 SAFETY ISSUE DESCRIPTION

RESEARCH ON PHENOMENA ASSOCIATED WITH CORE DEGREDATION AND FUEL MELTING (II.B.5)

The description of TAP Task II.B.5 is as follows:

> For a number of key severe accident sequences, there are critical phenomenological unknowns or uncertainties that impact containment integrity assessments and judgments regarding the desirability of certain mitigating features. The phenomena fall into three broad categories: 1) the behavior of severely damaged fuel, including oxidation and hydrogen generation; 2) the behavior of the core melt in its interaction with water, concrete, and core retention materials; and 3) the effect of potential hydrogen burning and/or explosions on containment integrity. Steam explosions will also be considered in this category. Previous work in these several areas has received less attention, since these areas relate to accidents beyond the design basis [of power plants]... RES [is] conducting major programs to support the basis for rulemaking and to confirm certain licensing decisions. Complementary efforts conducted within NRR will address specific licensing issues related to the subject research (NRC 1980; NRC 1982b).

Behavior of Severely Damaged Fuel

(a) In-pile studies: Fuel behavior research will include in-pile testing to help evaluate the effects of conditions leading to severe fuel damage. Such tests were scheduled for the INEL Power Burst Facility (PBF) in FY82 and later in the ACRR at Sandia and in the NRU reactor at Chalk River National Laboratory, Canada.

In the PBF and NRU, RES will perform a series of in-reactor fuel experiments to determine the effect of heating and cooling rates on damage to the bundle, rod fragmentation, distortion, and debris formation. Fission product release and hydrogen generation will also be measured during the test.

Separate effects studies will be conducted on rubble beds in the ACRR at Sandia.

(b) Hydrogen studies: The objective of this work is to increase understanding of the formation of hydrogen in a reactor from metal-water reactions, radiolytic decomposition of coolant, and corrosion of metals, and to determine their consequences in terms of pressure-time histories and hydrogen deflagration and detonation. This work will also include 1) the preparation of a compendium of information related to hydrogen as it affects reactor safety; 2) analysis of radiolysis under accident conditions; 3) a review of hydrogen sampling and analysis methods; 4) a study of the effects of hydrogen embrittlement on reactor vessel materials; and 5) a review of means of handling accident-generated hydrogen, with recommendations on improving current methods. Results of these studies were considered to support issue A-48, "Hydrogen Control and Effect of Burn," and were not considered further in this issue.

(c) Studies of post-accident coolant chemistry: The RES objective in this area is the development of a relationship between fission product release and fuel failure, and the improvement of post-accident sampling and analysis techniques. This will be accomplished by the investigation of fission product release in a variety of fuel failure experiments.

(d) Modeling of severe fuel damage: The effort in this area is the development of models for fuel rods operating beyond 2200°F which suffer a loss in geometry in order to compute extensive damage phenomena (such as eutectic liquid formation, fuel slumping, oxidation and hydrogen generation, fission product release and interaction with the coolant, rubble-bed particle size, extent of fuel and clad melting, and flow blockage).

Behavior of Core Melt

The RES fuel-melt research program will develop and verify a methodology for assessing the consequences and mitigation of fuel-melt accidents. The program addresses the range of severe reactor accident phenomena from the time when extensive fuel damage and major core geometry changes have occurred until the containment has failed and/or the molten core materials have attained a semipermanent configuration and further movement is terminated. Studies of improvements in containment design to reduce the risk of core-melt accidents are also included.

The program is composed of integrated tasks that include scoping, phenomenological and separate effects tests, and demonstration experiments that provide results for the development and verification of analytical models and codes. These codes and supporting data are then used for the analysis of thermal, mechanical, and radiological consequences of accidents and for decisions related to requirements of design features for mitigation and performance confirmation.

The technical scope of the program includes work in the following areas:

- fuel debris behavior
- fuel interactions with structure and soil
- radiological source term
- fuel-coolant interactions
- systems analysis codes
- mitigation features.

Effect of Hydrogen or Steam Explosions on Containment Integrity

A method will be developed to predict the response of containment structures to hydrogen or steam explosions. Both the loading associated with the explosion and the structural response will be included.

NRC will systematically study the uncertainties involved in the prediction of containment response to hydrogen or steam explosions. The staff will then assess the bounds of uncertainty associated with current technology. These results will support issue A-48 and were not considered in this analysis.

RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS (II.B.8)

TAP Task II.B.8 addresses a need to conduct both a short-term and a long-term rulemaking to establish policy, goals, and requirements with regard to accidents resulting in core damage greater than the present design basis. As part of the short-term effort, an Advance Notice of Proposed Rulemaking and an Interim Rule were issued. The Advance Notice was issued on December 30, 1980 (45FR65474). The Interim Rule was issued in two parts: The first was issued in effective form in October 1981 (46FR58484), and the second was issued as a proposed rule on December 23, 1981 (46FR62281).

On January 4, 1982, NRC staff sent a policy paper (NRC 1982a) to the Commission for action. In the paper, the Commission was asked to reconsider the approach to long-term rulemaking. The substance of the paper was that the uncertainty associated with long-term rulemaking was--and is--an inhibiting force on the industry. The paper then recommended that, since new applications are to be standardized anyway, licensing could proceed on these standardized designs, using the information presently available. Probabilistic risk assessments and the safety goal would be used to assess plant safety and, if the plant needed safety features beyond the present requirements to meet the safety goal, they could be included. This approach would not need rulemaking specifically directed at severe accident mitigation.

The Commission directed the staff to make several changes in the policy paper.[a] The staff then submitted revised papers incorporating the changes directed by the Commission, including ACRS input (NRC 1982b). The revised papers are still under Commission consideration.

The long-term rulemaking is intended to require means for dealing with a damaged core. This translates into preventing the release of radioactivity and providing means for recovering from the accident. Specific items to be considered include the following: use of filtered, vented containment; hydrogen control measures; core retention devices ("core catchers"); re-examination of design criteria for decay heat removal, and other systems; post-accident recovery plans; criteria for locating highly radioactive systems; effects or accidents at multi-unit sites; and comprehensive review and evaluation of related guides and regulations.

PROPOSED ISSUE RESOLUTION

While this issue encompasses several areas of concern with regard to degraded core behavior and post-accident conditions, it is assumed that potential mitigative features will take the form of one or more of the following:

1. core retention devices
2. filtered venting of containment
3. hydrogen control features.

The effects of implementing hydrogen control features are considered in Issue A-48 and are excluded from this analysis.

Some preliminary assessments have been made of the effectiveness of core retention devices and filtered venting of containment. Their relative effectiveness will most likely vary with plant type, and the decision as to which, if any, of these features a plant should install will be based on plant-specific assessments. To obtain estimates of the risk reduction, dose, and cost associated with resolution of Issue II.B.5(1,2)/II.B.8, it is assumed that all operating and planned BWRs, ice condenser PWRs, and PWRs with low-pressure containment designs, and half of all remaining PWRs (operating and planned) will install filtered venting of their containments. This assumption is based on the discussion below.

In its earlier policy paper, the NRC made the following statement with regard to filtered venting of containment and core retention devices:

> In future CP (construction permit) applications for both PWRs and BWRs, filtered-vented containment systems, or a variation of such systems, should be provided if these yield a cost-effective reduction in risk. Some recent information indicates these systems may not be

(a) S. Chilk, "Staff Requirements--Briefing on Status and Plan for Severe Accident Rulemaking" (SECY-82-1). January 29, 1982, Memorandum to W. J. Dircks, U.S. Nuclear Regulatory Commission, Washington, D.C.

> cost-effective for large, dry containments while other studies indicate these may be of value for some pressure suppression containments such as the Mark III design of General Electric.... These preliminary conclusions need to be addressed and final conclusions reached for new designs before they are applied to future plants.
>
> Over the past several years, studies of large, dry containment buildings ... indicate that classical core retention devices are probably not cost-effective in reducing atmospheric release of radiation. Post-accident flooding of the reactor cavity may be all that is necessary to establish a coolable debris bed and prevent basemat penetration. However, unique basemat designs and unique or undesirable liquid-pathway characteristics should be carefully weighed in future CP applications before deciding that this concept can safely be dismissed (NRC 1982a).

In the Zion risk assessment, the effects upon risk of adding a core ladle and filtered venting of containment were evaluated. The following conclusions were reached:

> Provision of a core ladle has no risk reduction benefit.... Provision of a filtered-vented containment yields a marginal reduction in risk of a factor of approximately two (Commonwealth Edison Co. 1981).

In a "Risk Assessment for Filtered-Vented Containment Options for a BWR Mark I Containment," Benjamin et al. concluded:

> The results (of providing high/low volume containment relief, with high pressure service water core cooling and drywell spray tie-ins, and a crushed rock filter for the low volume vent path) indicate risk reduction factors on the order of 40 to 400 (Benjamin et al. 1981).

These findings seem to indicate that filtered venting of containment has potentially significant value in reducing risk for pressure-suppression-type containments, while being of marginal value for large, dry containments. Core retention devices probably have marginal value, at best, in reducing risk. Based on these findings, it seems reasonable to assume a resolution for this issue which involves installation of filtered venting of containments at all operating and planned BWRs, ice condenser PWRs (pressure-suppression-type containments), and PWRs with low-pressure containment designs, and at half of all remaining operating and planned PWRs (large, dry-type containments). While the possibility of a plant's opting for a core retention device is recognized, it is assumed not to be part of the safety issue resolution (SIR) for this analysis.

Because final determination of the long-term rulemaking on degraded core behavior is several years from attainment, installation of filtered venting of containment at the assumed affected plants is postulated not to begin until 1988. This delay allows for additional research, NRC comments, and industry feedback (including the IDCOR results) prior to final determination.

2.0 SAFETY ISSUE RISK AND DOSE

Public risk reduction and occupational dose are estimated in this section. Results are summarized in Tables 1 and 2, respectively.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Research on Phenomena Associated with Core Degradation and Fuel Melting: Behavior of Severely Damaged Fuel, Behavior of Core Melt; Severely Damaged Core Rulemaking [II.B.5(1,2)/II.B.8]

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

   Based on an assumed SIR implementation date of 1988,

	N	$\bar{T}$ (yr)
PWR LPPs[(a)]	13	26.7
BWR LPPs	44	22.7
PWR HPPs[(a)]	39	24.1

   The development of these plant groupings and estimates is discussed in Attachment 1.

3. Plants Selected for Analysis:

Affected Plant Group	Representative Plant
PWR LPPs	Sequoyah 1
BWR LPPs	Peach Bottom 2
PWR HPPs	Zion 1

4-8. Steps Related to Affected Parameters, Accident Sequences, Release Categories, Core-Melt Frequency, and Their Base-Case Values:

   Estimates of the base-case, affected public risk are made directly in the next step. Thus, these steps are omitted.

   Note that the core-melt frequency is assumed to be unaffected since the filtered venting of containment serves to mitigate radioactive release subsequent to core melt rather than to reduce the core-melt frequency.

(a) LPPs = low-pressure plants; HPPs = high-pressure plants (see Attachment 1 for definitions).

TABLE 1. (contd)

4-8. Steps Related to Affected Parameters, Accident Sequences, Release Categories, Core-Melt Frequency, and Their Base-Case Values (contd):

Benjamin et al. (1981) acknowledge the potential to reduce core-melt frequency for many of their filtered venting options. However, this potential is attributable to the additional features included with each filtered venting option rather than to the filtered venting itself.

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

The base-case, affected public risks are calculated from the release category frequencies in the original studies and their corresponding dose factors in the Guidelines (Andrews et al. 1983).[a] These frequencies are as follows:

Plant	Release Categories	Frequencies (1/py)
Sequoyah 1	PWR-1	1.0E-7
	PWR-2	1.0E-5
	PWR-3	2.6E-5
	PWR-4	1.6E-5
	PWR-5	4.0E-6
Peach Bottom 2	BWR-1	1.0E-6
	BWR-2	6.0E-6
	BWR-3	2.0E-5
	BWR-4	2.0E-6
Zion 1[b]	Z-1	2.7E-11
	2	1.1E-7
	2R	5.9E-6
	Z-3	2.2E-10

(a) Note the special modification necessary for the Zion release category dose factors discussed in Addendum 1 to Attachment 1.
(b) Release category frequencies are a summation of those due to internal events (Table 8.4.2 of the Zion study) and major external events, i.e., earthquakes and fires (while these can be calculated directly from Section 7 of the Zion study, they are more readily derived from the draft report "Prioritization of Safety Issues Project: A Methodology for Estimating the Public Risk for Seismic and Fire External Events" (June 1983).

TABLE 1. (contd)

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

Plant	Release Categories	Frequencies (1/py)
Zion 1 (contd)	5R	5.6E-9
	Z-5	4.8E-10
	6	1.1E-9
	7	8.3E-9
	8A	9.2E-9
	8B	4.6E-5

When combined with the appropriate dose factors, these release category frequencies yield the following base-case, affected public risks:

Plant	W (man-rem/py)
Sequoyah 1	236
Peach Bottom 2	151
Zion 1	28.3

10-14. Steps Related to Adjusted-Case Values of Affected Parameters, Accident Sequences, Release Categories, Core-Melt Frequency, and Public Risk:

The public risk reduction is estimated directly in Step 16. Thus, these steps are omitted.

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

The core-melt frequency is unaffected. Thus, $\Delta\bar{F} = 0$.

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

Based on the discussion in Attachment 1, the public risk is assumed to be reduced by a factor of 40 at the LPPs (represented by Sequoyah and Peach Bottom) and by a factor of two at the HPPs (represented by Zion). Thus,

Plant	ΔW (man-rem/py)
Sequoyah 1	230
Peach Bottom 2	147
Zion 1	14

TABLE 1. (contd)

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.4E+5	7.8E+6	0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Research on Phenomena Associated with Core Degradation and Fuel Melting: Behavior of Severely Oamaged Fuel, Behavior of Core Melt; Severely Damaged Core Rulemaking [II.B.5(1,2)/II.B.8]

2. Affected Plants (N):

Based on a SIR implementation date of 1988 [(a)]

	LPPs		HPPs		
	PWRs	8WRs	PWRs	BWRs	Total
Operating	5	24	21	0	5D
Pre-1988 Planned	8	16	14	0	38
Post-1987 Planned	0	4	4	0	8
Total	13	44	39	0	96

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

Based on a SIR implementation date of 1988:[(a)]

	LPPs		HPPs	
	PWRs	8WRs	PWRs	8WRs
Operating	26.4 yr	19.2 yr	21.1 yr	--
Pre-1988 Planned	26.9 yr	26.1 yr	27.0 yr	--
Post-1987 Planned	--	30.0 yr	30.0 yr	--
Group Average	26.7 yr	22.7 yr	24.1 yr	--

(a) See Attachment 1.

TABLE 2. (contd)

4-5 Steps Related to Occupational Dose Reduction Due to Accident Avoidance:

There is no reduction in core-melt frequency for this SIR. Thus, ΔU = D.

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

While some of the labor involved in installing filtered venting will occur in and around containment (installing new piping or modifying existing lines), much of it will take place away from the containment structures (installation of filters and laying of new pipe routes). It is assumed that one-third of the labor involved in the installation occurs in radiation zones in and around the immediate containment vicinity. The remainder is assumed to take place outside of radiation zones.

PNL staff with expertise in nuclear reactor decommissioning estimate that 20% to 30% of the total cost of decommissioning can typically be attributed to dedicated staff labor. For this analysis, it is assumed that an average of 25% represents the contribution of dedicated staff labor to the cost of installing filtered venting. For an average implementation cost (excluding license amendment fee) of \$5.0E+6 at a plant operational prior to 1988 (see Step 8 of Table 3), the amount of labor is estimated to be (using \$1.0E+5/man-yr)

$$\text{Labor} = (0.25)(\$5.0\text{E}+6/\text{plant})/(\$1.0\text{E}+5/\text{man-yr})$$
$$= 12.5 \text{ man-yr/plant}$$

Assuming that one-third takes place in radiation zones, the amount of dedicated staff labor in radiation zones (including a 75% utilization factor) becomes

$$\begin{matrix}\text{Radiation}\\ \text{Zone Labor}\end{matrix} = (0.75)(12.5 \text{ man-yr/plant})/3 = 3.13 \text{ man-year/plant}$$

This is presumed to apply both to PWRs and BWRs, LPPs and HPPs, operational prior to 1988.

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

During refueling or testing outages, the average radiation field inside a BWR drywell is 0.10 R/hr, while the average in the reactor building just outside the drywell is 2.5 mR/hr.[a] For labor in and around the conainment at BWRs, an average radiation field of 25 mR/hr is assumed (this average is shifted more toward the lower end of the range since it is anticipated that work inside containment will be limited as much as possible). This average field is also assumed to apply to PWR LPPs.

(a) Based on information in Chapters 12 of the Grand Gulf and Palo Verde FSARs.

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

During refueling or testing outages at PWR HPPs, the average radiation field just inside and immediately around the containment is 2.5 mR/hr (higher fields exist near the reactor vessel and reactor cavity, but work in these areas is not anticipated for filtered venting installation). This average field is assumed for labor in and around the containment at PWR HPPs.

Based on the above, the occupational dose increase for SIR implementation is

$$\begin{aligned} D(LPPs) &= (3.13 \text{ man-yr/plant})(44 \text{ man-wk/man-yr}) \\ &\quad (40 \text{ man-hr/man-wk})(0.025 \text{ R/hr}) \\ &= (138 \text{ man-rem/plant} \end{aligned}$$

$$\begin{aligned} D(HPPs) &= (3.13 \text{ man-yr/plant})(44 \text{ man-wk/man-yr}) \\ &\quad (40 \text{ man-hr/man-wk})(0.0025 \text{ R/hr}) \\ &= 13.8 \text{ man-rem/plant} \end{aligned}$$

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

$$\begin{aligned} ND &= (5+24+8+16)(138 \text{ man-rem/plant}) \\ &\quad + (21+14)(13.8 \text{ man-rem/plant}) \\ &= 7770 \text{ man-rem} \end{aligned}$$

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

It is assumed that one-third of the labor associated with operation and maintenance of the filtered venting features (4 man-wk/py, see Step 9 of Table 3) will involve work in radiation zones. Thus,

$$\begin{aligned} \text{Labor} &= (4 \text{ man-wk/py})/3 \\ &= 1.33 \text{ man-wk/py} \end{aligned}$$

This is presumed applicable to all affected plants.

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

Again using the radiation field estimates of 25 mR/hr at LPPs and 2.5 mR/hr at HPPs,

TABLE 2. (contd)

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

D_o (LPPs) = (1.33 man-wk/py)(40 man-hr/man-wk)(0.025 R/hr)
= 1.33 man-rem/py

D_o (HPPs) = (1.33 man-wk/py)(40 man-hr/man-wk)(0.0025 R/hr)
= 0.133 man-rem/py

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

$N\bar{T}D_o$ = [13(26.7 yr) + 44(22.7 yr)](1.33 man-rem/py)
+ 39(24.1 yr)(0.133 man-rem/py)
= 1920 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
9700	2.9E+4	3200

3.0 SAFETY ISSUE COSTS

The costs to the industry and the NRC of resolving Issue II.B.5(1,2)/ II.B.8 are estimated in this section. Results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Research on Phenomena Associated with Core Degradation and Fuel Melting: Behavior of Severely Damaged Fuel, Behavior of Core Melt; Severely Damaged Core Rulemaking [II.B.5(1,2)/II.B.8]

2. Affected Plants (N):

Based on a SIR implementation date of 1988:[a]

(a) See Attachment 1.

TABLE 3. (contd)

2. Affected Plants (N) (contd):

	LPPs		HPPs		
	PWRs	BWRs	PWRs	BWRs	Total
Operating	5	24	21	0	50
Pre-1988 Planned	8	16	14	0	38
Post-1987 Planned	0	4	4	0	8
Total	13	44	39	0	96

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

Based on a SIR implementation date of 1988:[(a)]

	LPPs		HPPs		Group
	PWRs	BWRs	PWRs	BWRs	Average
Operating	26.4 yr	19.2 yr	21.1 yr	--	20.7 yr
Pre-1988 Planned	26.9 yr	26.1 yr	27.0 yr	--	26.6 yr
Post-1987 Planned	--	30.0 yr	30.0 yr	--	30.0 yr
			Overall Average	=	22.6 yr

Industry Costs (Steps 4 through 12)

4-5. Steps Related to Industry Cost Savings Due to Accident Avoidance:

There is no reduction in core-melt frequency for this SIR. Thus, $\Delta H = 0$.

6. Per-Plant Industry Resources for SIR Implementation:

Cost is estimated directly in the next step.

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

Carlson and Hickman (1978) estimated the cost of installing a relatively simple filtered venting option to be "on the order of a few million dollars." This option involved the addition of a modest-sized water tank external to the existing structure, plus addition of piping

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(a) See Attachment 1.

TABLE 3. (contd)

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

equipped with relief valves. The tank would not be elaborate; no unscheduled outages would be necessary, nor would unique technological advances be required. However, to allow for installation of more sophisticated filtered venting schemes, it is assumed for this analysis that the cost of installation at an operating plant will, on the average, be $5.0E+6. For a planned plant (operational after 1987), the cost to install filtered venting is assumed to be 10% less, or $4.5E+6, since the containment can be initially designed to accommodate this option.

Plants becoming operational prior to 1988 will also incur a license amendment fee, assumed to be $12,300 (Class IV). Thus, the costs per plant to implement the SIR become as follows:

	Cost ($/Plant)		
Plant Group	Filtered Venting	License Amendment	Total (I)
Operating	5.0E+6	12,300	5.01E+6
Pre-1988 Planned	5.0E+6	12,300	5.01E+6
Post-1987 Planned	4.5E+6	--	4.5E+6

These are taken to be average costs applicable to each plant in the various groups, both PWRs and BWRs, LPPs and HPPs.

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

$$NI = 88(\$5.01E+6/\text{plant}) + 8(\$4.5E+6/\text{plant})$$
$$= \$4.77E+8$$

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

Carlson and Hickman (1978) state that "increased maintenance and testing would be associated with the (filtered venting) system" discussed in Step 7. An additional 4 man-wk/py is assumed necessary for operation and maintenance of this system. This is presumed applicable to all affected plants.

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

$$I_0 = (4\ \text{man-wk/py})(\$2270/\text{man-wk}) = \$9080/\text{py}$$

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

$$N\bar{T}I_0 = 96(22.6\ \text{yr})(\$9080/\text{py}) = \$1.97E+7$$

TABLE 3. (contd)

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$5.0E+8	\$7.4E+8	\$2.6E+8

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Because this issue is still several years from final resolution, both NRC staff labor and contractor support will continue to be needed. Estimates are as follows:

Labor = 1 man-yr

Contractor Support = 5 man-yr

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

C_D = (6 man-yr)(\$1.0E+5/man-yr) = \$6.DE+5

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

The filtered venting features will presumably vary somewhat from plant to plant. Backfit installations will inevitably be slightly more complex. The NRC is expected to perform plant-specific reviews of these installations, expending slightly more effort to review backfits. NRC staff labor estimates are as follows:

Operating and pre-1988 planned plants	= 4 man-wk/plant
Post-1987 planned plants	= 3 man-wk/plant

These are taken to be average estimates applicable to both PWRs and BWRs.

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

Plant Group	C (\$/plant)
Operating	4(2270) = 9080
Pre-1988 Planned	4(2270) = 908D
Post-1987 Planned	3(2270) = 6810

TABLE 3. (contd)

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

$$NC = 88(\$9080/\text{plant}) + 8(\$6810/\text{plant}) = \$8.54E+5$$

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

The filtered venting features will be inspected as part of the NRC's routine plant inspection activities. An additional 1 man-day/py is presumed necessary. This applies to all affected plants.

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

$$C_O = (1\ \text{man-day/py})(1\ \text{man-wk}/5\ \text{man-day})(\$2270/\text{man-wk}) = \$454/\text{py}$$

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

$$N\bar{T}C_O = 96(22.6\ \text{yr})(\$454/\text{py}) = \$9.85E+5$$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$2.4E+6	\$3.2E+6	\$1.7E+6

ATTACHMENT 1

The installation of filtered venting of containment at the affected plants is assumed to have varied effects on the risk, depending upon the containment type. Significant risk reduction is expected at all operating and planned BWRs, ice condenser PWRs, and PWRs with low-pressure containment designs (< 2.0 kg/cm^2). There are 13 such PWRs:

Catawba 1 and 2
Comanche Peak 1 and 2
Cook 1 and 2
McGuire 1 and 2
Millstone 3
Sequoyah 1 and 2
Watts Bar 1 and 2.

These plants are heretofore referred to as "low-pressure plants" (LPPs). Marginal risk reduction is expected at half of all remaining PWRs (operating and planned), i.e., those with high-pressure containment designs. These plants are heretofore referred to as "high-pressure plants" (HPPs).

Based on the above, the numbers of affected plants in each group are a follows:

	LPPs		HPPs	
	PWRs	BWRs	PWRs	BWRs
Operating	5	24	21	0
Planned (operational prior to 1988)	8	16	14	0
Planned (operational in 1988 or beyond)	0	4	4	0
	13	44	39	0

The distinction regarding operation in 1988 results from the assumption that filtered venting is not installed until then. The average remaining operating lives of the above groups of plants, relative to 1988, are as follows:

ATTACHMENT 1 (contd)

	LPPs		HPPs	
	PWRs	BWRs	PWRs	BWRs
Operating	26.4 yr	19.2 yr	21.1 yr	--
Pre-1988 Planned	26.9 yr	26.1 yr	27.0 yr[a]	--
Post-1987 Planned	--	30.0 yr	30.0 yr	--
Group Average	26.7 yr	22.7 yr	24.1 yr	--

The above calculations are based on Appendix C of the Guidelines (Andrews et al. 1983). Containment specifications are taken from the nuclear power plant technical information in the August 1983 Supplement of Nuclear Engineering International.

The following three plants for which risk/reliability studies exist are selected to represent the three groups of affected plants:

1. Sequoyah 1 PWR--represents all PWR LPPs
2. Peach Bottom 2 BWR--represents all BWR LPPs
3. Zion 1 PWR--represents all PWR HPPs.

The base-case, affected public risk for each of these plants is calculated from the release category frequencies as provided in the risk/reliability studies (NRC 1975; Carlson et al. 1981; Commonwealth Edison Co. 1981). The Public Risk Reduction Work Sheet details these calculations. Note that the Zion release categories differ somewhat from those used in WASH-1400. These have been normalized to correspond to the dose factors used in the Guidelines (Andrews et al. 1983). Details of this normalization are provided in Addendum 1 to this Attachment, taken from the draft report "Prioritization of Safety Issues Project: A Methodology for Estimating the Public Risk for Seismic and Fire External Events," June 1983.

Estimates of the reduction in public risk resulting from installation of filtered venting at each type of affected plant are based on two assessments performed specifically for this design modification. The first estimated this reduction for various filtered-vented containment options at a Mark I BWR (Benjamin et al. 1981). Peach Bottom 2 was selected as the reference plant. For the most promising option of providing high/low volume containment relief, with high-pressure service water core cooling and drywell spray tie-ins, and a crushed rock filter for the low-volume vent path, a risk reduction factor of 40 to 400 is expected.

(a) Assumes average operating date of 1985.

For the Issue II.B.5(1,2)/II.B.8 analysis, the lower limit on this risk reduction factor (40) is assumed to be applicable at all affected LPPs, i.e., both PWRs and BWRs. The lower limit is chosen since the Benjamin et al. analysis included additional options besides filtered venting (service water core cooling and spray tie-ins) which may have contributed to this risk reduction. Since these options are not included as part of this issue resolution, use of the lower limit of 40 seems more appropriate. The analysis for the Peach Bottom 2 BWR is assumed to be applicable to PWR LPPs since both are designed to withstand the lower containment pressures (i.e., < 2.0 kg/cm^2).

The second estimate of the public risk reduction from installation of filtered venting is based on the analysis of this option performed for the Zion plant (Commonwealth Edison Co. 1981). A risk reduction factor of approximately two was estimated and is assumed to apply at all affected HPPs since all are designed, as is Zion, to withstand the higher containment pressures.

In summary, the risk reduction factors assumed for this analysis are as follows:

Affected Plant Group	Risk Reduction Factor
PWR LPPs	40
BWR LPPs	40
HPPs (all PWRs)	2

The risk reduction calculations are summarized in the Public Risk Reduction Work Sheet.

ADDENDUM 1

APPENDIX A

DOSE FACTOR ESTIMATION FOR RELEASE CATEGORIES OF INTEREST FROM THE ZION SAFETY STUDY

The release categories defined in the Zion Safety Study (ZSS, Commonwealth Edison Co. 1981) were based in part on the the PWR release categories identified in the Reactor Safety Study (RSS, NRC 1975). However modifications were introduced such that the ZSS effectively has its own set of unique release categories. Section 5 of the ZSS discusses the development of the release categories. The assumptions and techniques employed are presented there. These are beyond the scope of this report; the reader is referred to the ZSS for further information. However, for easy reference, part of Section 5 of the ZSS is included here as Addendum A.1 and summarizes the ZSS release categories of interest in this report, along with their source terms and associated parameters.[a]

The ZSS analysts characterized consequences by several different damage indices. To maintain consistency with the consequence index chosen for the Prioritization of Safety Issues Project (PSIP, Andrews et al. 1983) the man-rem index from the ZSS is used here to characterize release category consequences (i.e., dose factors). Table A.1 presents a tabular display of the complementary cumulative density functions for the doses in each ZSS release category of interest in this report. Since the ZSS analysts employed different sets of atmospheric conditions, population distributions, etc.

(a) This additional addendum has not been provided in this issue analysis report.

TABLE A.1. Complementary Cumulative Density Functions for Doses in ZSS Release Categories of Interest (from ZSS Table 8.3-4e)

PROBABILITY OF EXCEEDING DOSE FOR EACH RELEASE CATEGORY

DOSE[(a)]	Z-1	2	2R	Z-3	5R	Z-5	6	7	8A	8B
1.000E+03	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00	6.771E-01	9.167E-01	8.021E-01
2.000E+03	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00	5.625E-01	8.229E-01	5.208E-01
3.000E+03	1.000E+00	1.000E+00	1.000E+00	1.000E+00	9.896E-01	1.000E+00	9.687E-01	4.583E-01	7.396E-01	3.333E-01
5.000E+03	1.000E+00	1.000E+00	1.000E+00	1.000E+00	9.896E-01	1.000E+00	9.479E-01	3.646E-01	6.667E-01	3.021E-01
7.000E+03	1.000E+00	1.000E+00	1.000E+00	1.000E+00	9.896E-01	1.000E+00	9.479E-01	2.917E-01	6.042E-01	2.083E-01
1.000E+04	1.000E+00	1.000E+00	1.000E+00	1.000E+00	9.896E-01	1.000E+00	9.271E-01	2.083E-01	4.375E-01	1.458E-01
2.000E+04	1.000E+00	1.000E+00	1.000E+00	1.000E+00	9.557E-01	1.000E+00	8.542E-01	4.167E-02	3.542E-01	7.292E-02
3.000E+04	1.000E+00	1.000E+00	1.000E+00	1.000E+00	9.583E-01	1.000E+00	8.125E-01	1.042E-02	2.917E-01	2.083E-02
5.000E+04	1.000E+00	1.000E+00	1.000E+00	1.000E+00	8.854E-01	9.965E-01	7.187E-01	0.	2.198E-01	1.042E-02
7.000E+04	9.896E-01	9.965E-01	9.931E-01	1.000E+00	8.646E-01	9.931E-01	6.562E-01	0.	1.771E-01	0.
1.000E+05	9.792E-01	9.931E-01	9.931E-01	1.000E+00	8.542E-01	9.931E-01	6.042E-01	0.	1.042E-01	0.
2.000E+05	9.687E-01	9.931E-01	9.896E-01	9.792E-01	7.604E-01	9.825E-01	4.792E-01	0.	4.167E-02	0.
3.000E+05	9.583E-01	9.861E-01	9.861E-01	9.792E-01	7.187E-01	9.826E-01	3.750E-01	0.	0.	0.
5.000E+05	9.583E-01	9.826E-01	9.826E-01	9.271E-01	6.667E-01	9.583E-01	2.292E-01	0.	0.	0.
7.000E+05	9.375E-01	9.653E-01	9.653E-01	8.854E-01	6.042E-01	9.479E-01	1.875E-01	0.	0.	0.
1.000E+06	9.375E-01	9.549E-01	9.549E-01	8.333E-01	5.208E-01	9.167E-01	9.375E-02	0.	0.	0.
2.000E+06	8.958E-01	9.062E-01	9.062E-01	8.125E-01	3.333E-01	8.611E-01	1.042E-02	0.	0.	0.
3.000E+06	8.542E-01	8.681E-01	8.611E-01	7.708E-01	2.604E-01	8.403E-01	0.	0.	0.	0.
5.000E+06	8.437E-01	8.333E-01	8.229E-01	6.458E-01	1.667E-01	7.465E-01	0.	0.	0.	0.
7.000E+06	8.021E-01	7.951E-01	7.778E-01	5.937E-01	1.250E-01	6.979E-01	0.	0.	0.	0.
1.000E+07	7.604E-01	7.292E-01	7.153E-01	5.208E-01	5.208E-02	6.215E-01	0.	0.	0.	0.
2.000E+07	6.667E-01	5.729E-01	5.660E-01	3.854E-01	1.042E-02	4.792E-01	0.	0.	0.	0.
3.000E+07	4.792E-01	4.792E-01	4.722E-01	1.979E-01	0.	3.368E-01	0.	0.	0.	0.
5.000E+07	2.813E-01	3.090E-01	3.021E-01	1.354E-01	0.	1.736E-01	0.	0.	0.	0.
7.000E+07	1.453E-01	1.910E-01	1.736E-01	7.292E-02	0.	1.111E-01	0.	0.	0.	0.
1.000E+08	6.250E-02	1.076E-01	1.111E-01	3.125E-02	0.	5.556E-02	0.	0.	0.	0.
2.000E+08	0.	3.472E-03	3.472E-03	0.	0.	0.	0.	0.	0.	0.
3.000E+08	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
5.000E+08	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
7.000E+08	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
1.000E+09	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

(a) Whole-body man-rem

etc. from those employed in calculating dose factors for the PSIP, it is necessary that the ZSS release category doses be normalized to those of the PSIP for consistency. This is accomplished in the following manner.

Since the PSIP utilizes a point estimate dose factor for each release category, it is necessary to calculate a point estimate dose factor (the mean value will suffice) for each of the ZSS release categories. Given a complementary cumulative density function Q(x), which is just 1-P(x) where P(x) is the cumulative density function, the mean value of x is defined as follows.

$$\bar{x} \text{ (mean)} = \int_{-\infty}^{\infty} xf(x)dx \Big/ \int_{-\infty}^{\infty} f(x)dx$$

where f(x) is the probability density function.

$$f(x) = dP(x)/dx = -dQ(x)/dx$$

$$\bar{x} = \int_{-\infty}^{\infty} x[dQ(x)/dx]dx \Big/ \int_{-\infty}^{\infty} [dQ(x)/dx]dx$$

$$= \int_{-\infty}^{\infty} x[dQ(x)/dx]dx/[Q(\infty) - Q(-\infty)]$$

If the data of Table A.1 were plotted (on a log-log display, as would befit the orders of magnitude indicated in Table A.1), the resulting curves would follow no strict analytical functions. A reasonable approximation for each curve would be a series of line segments joining the various data points.

Assume each segment has endpoints (x_a, Q_a) and (x_b, Q_b). For any one segment, $Q(x)$ could be expressed as follows:

$$[\ln Q(x) - \ln Q_a]/[\ln x - \ln x_a] = [\ln Q_b - \ln Q_a]/[\ln x_b - \ln x_a] = m$$

$$\ln Q(x) = \ln Q_a + m \ln(x/x_a)$$

$$Q(x) = Q_a(x/x_a)^m$$

The term m is recognized as the slope of the line on a log-log plot. Taking the derivative of $Q(x)$ enables an approximation for $\bar{x}$ to be developed.

$$dQ(x)/dx = (mQ_a/x_a{}^m)x^{m-1}$$

$$\bar{x} = [(\int_{-\infty}^{x_1} + \int_{x_1}^{x_2} + \ldots + \int_{x_n}^{\infty})(mQ_a/x_a{}^m)x^m dx]/[Q(\infty) - Q(-\infty)]$$

$$= [mQ_a x^{m+1}/(m+1)x_a{}^m]\left(\Big|_{-\infty}^{x_1} + \Big|_{x_1}^{x_2} + \ldots + \Big|_{x_n}^{\infty}\right)/[Q(\infty) - Q(-\infty)]$$

where $mQ_a x^{m+1}/(m+1)x_a{}^m$ must be evaluated over each of the intervals.

Relaxing the restriction that $f(x)$ be defined from $-\infty$ to ∞ and designating its range of defintion to extend from some minimum value x_α to some maximum value x_β (with corresponding ordinates Q_α and Q_β), one obtains the following equation for $\bar{x}$:

$$x = [mQ_a x^{m+1}/(m+1)x_a^{\,m}]\left(\Big|_{x_\alpha}^{x_1} + \Big|_{x_1}^{x_2} + \ldots + \Big|_{x_n}^{x_\beta}\right)\Big/(Q_\beta - Q_\alpha)$$

This may be more simply expressed as:

$$x = \left\{\sum_{\substack{\text{all}\\ \text{intervals}}} mQ_a(x_b^{\,m+1} - x_a^{\,m+1})/(m+1)x_a^{\,m}\right\}\Big/(Q_\beta - Q_\alpha)$$

where the subscripts a and b represent the interval endpoints and

$$m = \ln(Q_b/Q_a)/\ln(x_b/x_a)$$

Using this formulation, the mean values for the dose factors of each ZSS release category given in Table A.1 become:

ZSS Release Category	Mean Dose Factors (man-rem)
Z-1	3.09+7
2	3.87+7
2R	3.79+7
Z-3	1.81+7
5R	2.37+6
Z-5	2.27+7
6	3.25+5
7	7.26+3
8A	2.86+4
8B	5.91+3

Clearly, these are generally much higher than the values used in the PSIP, a consequence of the different set of site conditions applied in the ZSS. To normalize these ZSS dose factors to those of the PSIP, it is assumed that the dose factor for ZSS category 2 (3.87+7 man-rem, which happens to be the maximum ZSS category dose factor) can be scaled directly to the PSIP dose factor for PWR-2 (4.8+6 man-rem). This is a conservative assumption which is also reasonable since the source term for ZSS category 2 was taken directly from the RSS. Based on this normalization, the dose factors for the ZSS release categories become:

ZSS Release Category	Mean Dose Factors (man-rem)
Z-1	3.83+6
2	4.80+6
2R	4.70+6
Z-3	2.24+6
5R	2.94+5
Z-5	2.82+6
6	4.03+4
7	9.00+2
8A	3.55+3
8B	7.33+2

For convenience, the distinction between ZSS categories 2 and 2R can be omitted since their dose factors differ by only 2%. Both can be viewed as release category 2 with dose factor 4.80+6 man-rem.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: II.D.2, Research on Relief and Safety Valve Test Requirements

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Concern exists regarding the ability of relief and safety valves, block valves and associated piping to provide sufficient primary system depressurization during anticipated transients without scram. NRC is currently monitoring industry's testing and evaluation of valve performance and will presumably issue recommendations for equipment upgrade as a result of these findings.

AFFECTED PLANTS	BWR:	Operating = 12	Planned = 10
	PWR:	Operating = 24	Planned = 21

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	1300
OCCUPATIONAL DOSES:	
SIR Implementation =	4700
SIR Operation/Maintenance =	0
Total of Above =	4700
Accident Avoidance =	9.9

COST RESULTS ($$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	21
SIR Operation/Maintenance =	0
Total of Above =	21
Accident Avoidance =	0.82
NRC COSTS:	
SIR Development =	0.40
SIR Implementation Support =	0.25
SIR Operation/Maintenance Review =	0
Total of Above =	0.65

RESEARCH ON RELIEF AND SAFETY VALVE TEST REQUIREMENTS

ISSUE II.D.2

1.0 SAFETY ISSUE DESCRIPTION

The objective of Task II.D of the TMI Action Plan (NUREG-0660, NRC 1980b) on "Reactor Coolant System Relief and Safety Valves" is to "demonstrate by testing and analysis that the relief and safety valves, block valves and associated piping in the reactor coolant system are qualified for the full range of operating and accident conditions. Anticipated transients without scram (ATWS) may be considered later in the program." Subtask II.D.2 requires that NRC "technically monitor and analyze the planned industry valve test and analytical program at EPRI (Electric Power Research Institute) and collect, analyze and compare information from foreign tests; develop, improve or verify available flow discharge and structural response models using the above information; determine the need for a valve-testing program, with the main focus to be on subcooled and two-phase discharge and on determining operability; and conduct additional tests, as necessary, to assure that the response to the full spectrum of fluid conditions that would be expected to result from anticipated operational occurrences and ATWS events has been adequately characterized."

The above work, with the exception of that related to the ATWS events, has been performed in conjunction with Subtask II.D.1, "Reactor Coolant System Relief and Safety Valves--Testing Requirements," as defined in NUREG-0737 (NRC 1980a). In this regard, Subtask II.D.1 incorporates all aspects of Subtask II.D.2, with the exception of the ATWS-related research. Thus, risk, dose, and cost estimation is performed only for the ATWS-related aspects of Subtask II.D.2 in this issue analysis.

The Idaho National Engineering Laboratory has been performing the technical monitoring and evaluation of the industry valve testing and analysis program for the NRC. Valve testing for BWRs is complete, but some testing remains to be done for PWRs (primarily for block valves). Many of the results to date remain proprietary, although some initial findings indicate that certain block valves fail to perform under full flow conditions at maximum differential pressure.

As considered here, Issue II.D.2 addresses the ability of relief and safety valves (R&SVs), block valves and associated piping to provide sufficient primary system depressurization during ATWS sequences. Coupled with failure of the reactor protection system (RPS) following a transient, inadequate depressurization could result in rupture of the reactor coolant pressure boundary (RCPB), producing a LOCA.

The assumed safety issue resolution (SIR) for II.D.2 consists of two parts:

1. The testing and evaluation will be completed for R&SVs, block valves and associated piping regarding their ability to provide sufficient primary system depressurization during ATWS sequences.

2. Industry will implement any recommended modifications to enhance the above ability as a result of the tests and analysis. For this issue analysis, these modifications are presumed to involve increased sizing of R&SVs and their associated block valves at half of all plants. Only half of all plants are presumed affected since depressurization capability will vary from plant to plant.

2.0 SAFETY ISSUE RISK AND DOSE

PUBLIC RISK REDUCTION

The depressurization capability of the R&SVs, block valves and associated piping can affect public risk via the ATWS sequences. Using Oconee 3 as the representative PWR, one observes that the only ATWS sequence appearing among the dominant core-melt sequences is T_2KMU (Andrews et al. 1983), where

T_2 = loss-of-power-conversion-system (PCS) transient caused by other than a loss-of-offsite power

K = failure of the RPS

M = interruption of the PCS (a certainty given the T_2 initiator)

U = failure of the high-pressure injection system.

As discussed in the Oconee RSSMAP study (Kolb et al. 1981), this sequence assumes that the RCPB remains intact, despite a high potential peak pressure (4000 psi).

Inadequate capability of the R&SVs, block valves and associate piping to depressurize the primary system in an ATWS sequence can impact core melt if the RCPB subsequently ruptures. The Oconee RSSMAP study states:

> For...ATWS, all three (pressurizer safety and relief) valves are needed to limit RCS (reactor coolant system) pressure to less than 150% of the design pressure. It is not clear whether this requirement can be met (Kolb et al. 1981).

The Oconee study did not postulate an ATWS sequence in which the RCPB ruptured. However, to assess the impact of this issue resolution on public risk, one must be postulated.

It is assumed that, given a T_2 transient and failure of the RPS, there is some potential for rupture of the RCPB. Assuming this rupture to be more likely in smaller than in larger pipes, a new failure event (S_{33}) is designated, corresponding to the rupture of an RCPB pipe with diameter $\leq$4" given T_2K. Based on the dominant core-melt sequences for the S_3 LOCA initiator in Oconee, the following accident sequences result:

$$T_2KS_{33}D - (\gamma, \beta, \varepsilon)$$
$$T_2KS_{33}H - (\gamma, \beta, \varepsilon)$$
$$T_2KS_{33}FH - (\gamma, \beta, \varepsilon)$$

where the containment failure modes are assumed to lead to the same PWR release categories as for the corresponding S_3 sequences. These ATWS sequences are introduced into the Oconee plant risk equation as defined in Appendix A (Andrews et al. 1983).

Using Grand Gulf 1 as the representative BWR, one observes that the only ATWS sequence appearing among the dominant core-melt sequences is $T_{23}C$ (Andrews et al. 1983), where

T_{23} = transient other than loss-of-offsite power which requires a reactor shutdown

C = failure to achieve reactor subcriticality.

As discussed in the Grand Gulf RSSMAP study (Hatch et al. 1981), this sequence assumes that the RCPB remains intact.

As for Oconee, no Grand Gulf ATWS sequence was postulated in which the RCPB ruptured. However, to assess the impact of this resolution on public risk, the following one is postulated due to inadequate depressurization by the R&SVs, block valves and associated piping.

It is assumed that, given a T_{23} transient and failure of the RPS, there is some potential for rupture of the RCPB. Assuming this rupture to be more likely in smaller than in larger pipes, a new failure event (S_s) is designated corresponding to rupture of the RCPB with an area <1 ft^2 (the Grand Gulf "small LOCA") given $T_{23}C$. Based on the dominant core-melt sequences for the S LOCA initiator in Grand Gulf, the following accident sequences result:

$$T_{23}CS_sI - (\alpha, \delta)$$

where the containment failure modes are assumed to lead to the same BWR release categories as for the SI-(α, δ) sequences. These ATWS sequences are introduced into the Grand Gulf plant risk equation as defined in Appendix B of Andrews et al. 1983.

The results of the analysis for public risk reduction due to resolution of Issue II.D.2 are summarized in Table 1.

OCCUPATIONAL DOSE

Occupational dose will be accumulated during Part Two of the assumed SIR: Industry implementation of any recommended modifications to enhance the ability of R&SVs and block valves to provide sufficient primary system depressurization during ATWS sequences. This dose will accrue only at half of the operating plants assumed to be affected.

The assumed modification is an increased sizing of R&SVs and associated block valves. This will presumably not alter the R&SV reliability nor change the existing operation/maintenance schedule. Thus, no change in occupational dose received during SIR operation/maintenance is anticipated.

The results of the analysis for occupational dose due to resolution of Issue II.D.2 are summarized in Table 2.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Research on Relief and Safety Valve Test Requirements (II.D.2)

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

   Half of all PWRs and BWRs are assumed to be affected.

	N	$\bar{T}$ (yr)
PWRs	45	28.8
BWRs	22	27.4

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf 1 - representative BWR

4. Parameters Affected by SIR:

   PWR: S_{33} (ATWS-induced rupture of an RCPB pipe with diameter <4")

   BWR: S_s (ATWS-induced rupture of the RCPB with an area <1 ft^2)

   These terms are defined earlier in this section as part of "new" ATWS sequences.

TABLE 1. (contd)

5. Base-Case Values for Affected Parameters:

It is unclear whether ATWS-induced overpressure of the RCPB beyond its design pressure will automatically cause rupture. A likelihood of 0.5 is assumed for both S_{33} and S_s.

PWR: $S_{33} = 0.5$ BWR: $S_s = 0.5$

6. Affected Accident Sequences and Base-Case Frequencies:

"New" ATWS sequences are developed earlier in this section to model RCPB rupture. Except for the affected parameters, all parameters have their original values as their base-case values, i.e., their values from Appendices A and B of Andrews et al. (1983).

Sequence		Frequency (1/py)
PWR: $T_2KS_{33}D$ -	γ (PWR-3)	2.1E-8
	β (PWR-5)	3.1E-10
	ε (PWR-7)	2.1E-8
$T_2KS_{33}H$ -	γ (PWR-3)	1.5E-7
	β (PWR-5)	2.2E-9
	ε (PWR-7)	1.5E-7
$T_2KS_{33}FH$ -	γ (PWR-2)	6.3E-8
	β (PWR-4)	9.2E-10
	ε (PWR-6)	6.3E-8
BWR: $T_{23}CS_sI$ -	α (BWR-1)	8.9E-11
	δ (BWR-2)	8.9E-9

TABLE 1. (contd)

7. Affected Release Categories and Base-Case Frequencies:

PWR-2 = 6.3E-8/py
PWR-3 = 1.7E-7/py
PWR-4 = 9.2E-10/py
PWR-5 = 2.5E-9/py
PWR-6 = 6.3E-8/py
PWR-7 = 1.7E-7/py

BWR-1 = 8.9E-11/py
BWR-2 = 8.9E-9/py

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

PWR: 4.7E-7/py   BWR: 8.9E-9/py

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

PWR: 1.2 man-rem/py   BWR: .063 man-rem/py

10. Adjusted-Case Values for Affected Parameters:

Resolution of this issue (increased sizing of R&SVs and any associated block valves) is assumed to decrease the likelihood of an ATWS-induced rupture of the RCPB by a factor of 5. Increased sizing would enhance the depressurization capability, reducing the potential for RCPB rupture.

PWR: $S_{33} = 0.5/5 = 0.1$ BWR: $S_S = 0.5/5 = 0.1$

11. Affected Accident Sequences and Adjusted-Case Frequencies:

	Sequence	Frequency (1/py)
PWR:	$T_2KS_{33}Q$ - γ (PWR-3)	4.2E-9
	$T_2KS_{33}Q$ - β (PWR-5)	6.1E-11
	$T_2KS_{33}Q$ - ϵ (PWR-7)	4.2E-9
	$T_2KS_{33}H$ - γ (PWR-3)	3.0E-8
	$T_2KS_{33}H$ - β (PWR-5)	4.4E-10
	$T_2KS_{33}H$ - ϵ (PWR-7)	3.0E-8

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

$$T_2KS_{33}FH - \begin{cases} \gamma \ (PWR\text{-}2) & 1.3E\text{-}8 \\ \beta \ (PWR\text{-}4) & 1.8E\text{-}10 \\ \epsilon \ (PWR\text{-}6) & 1.3E\text{-}8 \end{cases}$$

$$BWR: \quad T_{23}CS_sI - \begin{cases} \alpha \ (BWR\text{-}1) & 1.8E\text{-}11 \\ \delta \ (BWR\text{-}2) & 1.8E\text{-}9 \end{cases}$$

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-2 = 1.3E-8/py
PWR-3 = 3.4E-8/py
PWR-4 = 1.8E-10/py
PWR-5 = 5.0E-10/py
PWR-6 = 1.3E-8/py
PWR-7 = 3.4E-8/py

BWR-1 = 1.8E-11/py
BWR-2 = 1.8E-9/py

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

$\bar{F}^*_{PWR}$ = 9.4E-8/py $\bar{F}^*_{BWR}$ = 1.8E-9/py

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

W^*_{PWR} = 0.25 man-rem/py W^*_{BWR} = .013 man-rem/py

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

$\Delta\bar{F}_{PWR}$ = 3.8E-7/py $\Delta\bar{F}_{BWR}$ = 7.1E-9/py

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

ΔW_{PWR} = 0.99 man-rem/py ΔW_{BWR} = .051 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1300	4.8E+4	0

TABLE 2. Occupational Oose Work Sheet

1. Title and Identification Number of Safety Issue:

   Research on Relief and Safety Valve Test Requirements (II.D.2)

2. Affected Plants (N):

   Half of all PWRs and BWRs are presumed to be affected.

		N
PWRs:	Operating	24
	Planned	21
		45
BWRs:	Operating	12
	Planned	10
		22

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

		$\bar{T}$ (yr)
PWRs:	Operating	27.7
	Planned	30.0
	All	28.8
BWRs:	Operating	25.2
	Planned	30.0
	All	27.4

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

$$\Delta(\bar{FD}_R)_{PWR} = (19{,}9D0 \text{ man-rem})(3.8E\text{-}7/py) = .D076 \text{ man-rem/py}$$
$$\Delta(\bar{FD}_R)_{BWR} = (19{,}900 \text{ man-rem})(7.1E\text{-}9/py) = 1.4E\text{-}4 \text{ man-rem/py}$$

TABLE 2. (contd)

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
9.9	73	0

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

It is assumed that 3500 man-hr/plant (with a 75% utilization factor) will be required in radiation zones to implement the modifications on the R&SVs and associated block valves assumed in the SIR. This applies only to operating plants.

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

The radiation field for the above labor is assumed to be .050 R/hr (as in Issue B-55, "Improved Reliability of Target Rock Safety Relief Valves"). A 75% utilization factor on the above labor is assumed.

D = (3500 man-hr/plant)(.050 R/hr)(0.75) = 130 man-rem/plant

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

ND = (36 operating plants)(130 man-rem/plant) = 4700 man-rem

9-11. Steps Related to Occupational Dose Increase for SIR Operation and Maintenance:

As discussed earlier in this section, no change in occupational dose received during SIR operation/maintenance is anticipated. Thus, $D_o = 0$.

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
4700	1.4E+4	1600

3.0 SAFETY ISSUE COSTS

Industry costs are estimated only for Part Two of the SIR--hardware modifications. The testing and evaluation for R&SVs, block valves and associated piping is essentially complete for BWRs, although some remains for PWRs. It is assumed that these industry costs have been committed. Thus, they are not included in the issue analysis.

Hardware modifications will presumably be incorporated at half of all plants. However, for planned plants which will be constructed subsequent to final recommendations from Part One of this SIR, these modifications (if applicable) will be incorporated during initial installation of the R&SVs and associated block valves. No retrofit is involved since these modifications will be treated as design changes. Thus, no SIR implementation costs will be incurred as a result of Issue II.D.2 for these planned plants. Since Issue II.D.2 is scheduled for completion in FY-1985, it is assumed that only half of the planned plants scheduled to begin operation prior to 1986 will incur SIR implementation costs. Review of Appendix C (Andrews et al. 1983) indicates that there are 24 planned PWRs and 13 planned BWRs in this category. Thus, SIR implementation costs will be incurred at only 12 planned PWRs and 7 planned BWRs under the previous assumptions.

NRC costs are estimated for both parts of the SIR. Further SIR development is anticipated in evaluating industry test results, especially for the PWRs. The NRC will also support SIR implementation at the appropriate plants (assumed to be 36 operating plants and 19 planned plants) by monitoring hardware modifications.

As discussed previously in Section 2.0, no change in the existing operation/maintenance schedule is anticipated for this SIR. Thus, neither industry nor the NRC will incur any additional costs related to SIR operation/maintenance.

Table 3 summarizes the analysis results for the industry and NRC costs due to resolution of Issue II.D.2.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Research on Relief and Safety Valve Test Requirements (II.0.2)

2. Affected Plants (N):

   Half of all PWRs and BWRs are presumed to be affected.

		N			N
PWRs:	Operating	24	BWRs:	Operating	12
	Planned	21		Planned	10
	All	45[a]		All	22[a]

(a) For industry and NRC costs related to SIR implementation, the affected numbers of planned plants are 12 PWRs and 7 BWRs.

TABLE 3. (contd)

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

		$\bar{T}$ (yr)
PWRs:	Operating	27.7
	Planned	30
	All	28.8
BWRs:	Operating	25.2
	Planned	30
	All	27.4

Industry Costs (Steps 4 through 12)

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

$$\Delta(\bar{F}A)_{PWR} = (\$1.65E+9)(3.8E-7/py) = \$630/py$$
$$\Delta(\bar{F}A)_{BWR} = (\$1.65E+9)(7.1E-9/py) = \$12/py$$

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

Best Estimate	Upper Bound	Lower Bound
\$8.2E+5	\$5.9E+6	0

6. Per-Plant Industry Resources for SIR Implementation:

Labor (engineering, crafts, etc.) = 125 man-wk/plant

Equipment (cost estimated directly in next step)

Additional down-time = none

These resources are needed only at 36 operating plants and 19 planned plants.

TABLE 3. (contd)

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

Labor = (125 man-wk/plant)($2270/man-wk) = $2.84E+5/plant

Equipment = $1E+5/plant

License Amendment (Class III, see 10 CFR 170.22) = $4000/plant (operating only)

$$I = \begin{cases} \$3.88E+5/\text{plant (operating)} \\ \$3.84E+5/\text{plant (planned)} \end{cases}$$

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

NI = (36 operating plants)($3.88E+5/plant) + (19 planned plants) ($3.84E+5/plant) = $2.1E+7

9-11. Steps Related to Industry Cost for SIR Operation and Maintenance:

As discussed earlier in this section, no additional industry cost will be incurred for SIR operation/maintenance. Thus, $I_0 = 0$.

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$2.1E+7	$3.2E+7	$1.1E+7

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

SIR development is scheduled for completion in FY-1985. Given that valve testing for BWRs and some of the valve testing for PWRs are complete, it is assumed that further NRC SIR development will require two man-years of staff labor plus an equal amount of contractor support.

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

Labor = (2 man-yr)($1.0E+5/man-yr) = $2.0E+5

Contractor Support = (2 man-yr)($1.0E+5/man-yr) = $2.0E+5

C_D = $4.0E+5

TABLE 3. (contd)

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

Monitoring of hardware modifications at the affected plants (36 operating plants and 19 planned plants) is assumed to require 2 man-wk/plant of NRC staff labor.

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

C = (2 man-wk/plant)($2270/man-wk) = $4540/plant

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

NC = (55 plants)($4540/plant) = $2.5E+5

18-20. Steps Related to NRC Cost for Review of SIR Operation and Maintenance:

As discussed earlier in this section, no additional NRC cost will be incurred for review of SIR operation/maintenance. Thus, $C_O = 0$.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$6.5E+5	$8.9E+5	$4.1E+5

REFERENCES

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U.S. NRC. 1980a. Clarification of TMI Action Plan Requirements. NUREG-0737, U.S. Nuclear Regulatory Commission, Washington, D.C.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: II.E.2.2, Research on Small-Break LOCAs and Anomalous Transients

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

The response of LWRs to small-break LOCAs and anomalous transients is being studied in the loss-of-fluid-test (LOFT) facility. Means are being investigated to enhance the operator's ability to respond to upset conditions. It is assumed that additional operator training and advanced instrumentation result in reduced likelihood of operator error during upset conditions. Program results may ultimately reduce LOCA and transient frequencies and/or severities.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	3.9E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	110
SIR Operation/Maintenance =	0
Total of Above =	110
Accident Avoidance =	290

COST RESULTS ($$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	36
SIR Operation/Maintenance =	0
Total of Above =	36
Accident Avoidance =	24
NRC COSTS:	
SIR Development =	3.1
SIR Implementation Support =	0.2
SIR Operation/Maintenance Review =	1.7
Total of Above =	5.0

RESEARCH ON SMALL-BREAK LOCAS AND ANOMALOUS TRANSIENTS

ISSUE II.E.2.2

The objective of this issue is to study the response of LWRs to various sizes of small-break, loss-of-coolant accidents (LOCAs) and anomalous transients. Means to enhance the operator's ability to respond to upset conditions are being investigated.

1.0 SAFETY ISSUE DESCRIPTION

This issue is directly linked to experiments performed at the loss of fluid test (LOFT) facility at Idaho Falls. The major goal of this program is to provide experimental data for verification of methodologies used in thermal hydraulics modeling of nuclear reactors. This includes data for coolant flow and heat transfer under a variety of conditions.

One part of this program has been the examination of small-break LOCAs and anomalous transients. Specifically, the ability of typical process instruments to provide accurate and sufficient information to operating personnel is being assessed. Advanced control room and diagnostic instrumentation is being used as part of the augmented operator capabilities program to assess operator needs to mitigate the consequences of LOCA and transient sequences.

In addition, the NRC has allocated funds to sponsor a study on the effects of localized thermal shock coincident with internal pressure on vessel crack propagation. Previous thermal-shock tests have been conducted without internal pressure to simulate the large LOCA. The pressurized thermal-shock tests will provide a licensing basis for postulated material condition, flaw size, and accident loads in small breaks.

Research on analytical methods development and assessment is directed toward improving current computer codes, development and application of advanced computer codes for small-break LOCA and other accident analyses, and analyses of thermal hydraulic phenomena in LWRs in the presence of severe core damage.

Note that the experiments dealing with small-break LOCAs are only a part of those being performed at the LOFT facility. As a first estimate, it is assumed that ~20 percent of funding is utilized for this purpose. The work for small-break LOCAs was completed primarily in FY81 and FY82, with final data analysis being the primary function of FY83 funding. The methodology used in this analysis considers only future costs, but the reader should be aware that this is an ongoing program with sunk costs. Costs are developed further in Attachment 1, presented in Section 3.0.

The primary goal of the small-break and transient research at LOFT is to improve operator performance during these off-normal events. In the evaluation of risk for this issue resolution, operator error failure likelihoods found in sequences initiated by small-break LOCAs or transients are assumed to be reduced by some amount. This applies primarily to PWRs; however, it is felt that it will also find applications in BWR LOCA sequences. Consequently, BWR risk reduction is also examined using the above approach.

In addition to improving operator performance, it is possible that the LOFT program will ultimately provide information useful in reducing the frequency or severity of small-break LOCAs or transients. However, it should be recognized that although this potential exists and further risk reduction may be possible, it cannot be quantified at this time. For purposes of this analysis, only reduction in operator error during LOCA and transient sequences is assumed for the safety issue resolution (SIR).

2.0 SAFETY ISSUE RISK AND DOSE

Public risk reduction and occupational dose are estimated in this section. Results are summarized in Tables 1 and 2, respectively.

It is assumed that small-break LOCAs or transients leading to a LOCA, typically via a jammed open pressure relief valve, represent the initiating events applicable to this issue. Using Oconee 3 and Grand Gulf 1 as the respresentative PWR and BWR, respectively,

Oconee	Grand Gulf
S_3 LOCA	S LOCA
T_2Q transient	T_1P transient
	$T_{23}P$ transient

Operator error failure likelihoods in such sequences are assumed to be reduced by one-third as a result of a combination of operator training and improved instrumentation. These operator error failures for PWRs are HPRSCM and WXCM, and OP for BWR sequences.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Research on Small-Break LOCAs and Anomalous Transients (II.E.2.2)

TABLE 1. (contd)

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

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

3. Plants Selected for Analysis:

Oconee 3 - representative PWR

Grand Gulf 1 - representative BWR

4. Parameters Affected by SIR:

PWR

HPRSCM: common-cause failure of the operator to align suction of the high-pressure recirculation system to the suction of the low-pressure recirculation system.

WXCM: common-cause failure of the operator to open both containment sump suction valves in the low-pressure containment spray recirculation system at the start of recirculation.

BWR

OP: failure of operator to manually initiate the automatic depressurization system.

5. Base-Case Values for Affected Parameters:

PWR	BWR
HPRSCM = 0.003	OP = 0.0015
WXCM = 0.003	

6. Affected Accident Sequences and Base-Case Frequencies:

PWR

$$T_2MQH - \begin{cases} \gamma \text{ (PWR-3)} = 2.25E-6/py \\ \beta \text{ (PWR-5)} = 3.29E-8/py \\ \varepsilon \text{ (PWR-7)} = 2.25E-6/py \end{cases}$$

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

PWR (contd)

$$S_3H - \begin{cases} \gamma\ (PWR\text{-}3) = 1.95E\text{-}6/py \\ \beta\ (PWR\text{-}5) = 2.85E\text{-}8/py \\ \varepsilon\ (PWR\text{-}7) = 1.95E\text{-}6/py \end{cases}$$

$$T_2MQFH - \begin{cases} \gamma\ (PWR\text{-}2) = 2.25E\text{-}6/py \\ \beta\ (PWR\text{-}4) = 3.29E\text{-}8/py \\ \varepsilon\ (PWR\text{-}6) = 2.25E\text{-}6/py \end{cases}$$

$$S_3FH - \begin{cases} \gamma\ (PWR\text{-}2) = 1.95E\text{-}6/py \\ \beta\ (PWR\text{-}4) = 2.85E\text{-}8/py \\ \varepsilon\ (PWR\text{-}6) = 1.95E\text{-}6/py \end{cases}$$

BWR

$$T_1PQE - \begin{cases} \gamma\ (BWR\text{-}3) = 1.32E\text{-}8/py \\ \delta\ (BWR\text{-}4) = 1.32E\text{-}8/py \end{cases}$$

$$T_{23}PQE - \begin{cases} \gamma\ (BWR\text{-}3) = 2.35E\text{-}7/py \\ \delta\ (BWR\text{-}4) = 2.35E\text{-}7/py \end{cases}$$

7. Affected Release Categories and Base-Case Frequencies:

PWR-2 = 4.20E-6/py	BWR-3 = 2.48E-7/py
PWR-3 = 4.20E-6/py	BWR-4 = 2.48E-7/py
PWR-4 = 6.13E-8/py	
PWR-5 = 6.13E-8/py	
PWR-6 = 4.20E-6/py	
PWR-7 = 4.20E-6/py	

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

PWR: $\bar{F}$ = 1.69E-5/py BWR: $\bar{F}$ = 5.0E-7/py

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

PWR: W = 44 man-rem/py BWR: W = 1.42 man-rem/py

TABLE 1. (contd)

10. Adjusted-Case Values for Affected Parameters:

A reduction of 33% in operator error is assumed as a result of improved instrumentation and operator response.

PWR	BWR
HPRSCM = 0.002	OP = 0.0010
WXCM = 0.002	

11. Affected Accident Sequences and Adjusted-Case Frequencies:

PWR

$$T_2MQH - \begin{cases} \gamma = 1.50E\text{-}6/py \\ \beta = 2.19E\text{-}8/py \\ \varepsilon = 1.50E\text{-}6/py \end{cases}$$

$$S_3H - \begin{cases} \gamma = 1.30E\text{-}6/py \\ \beta = 1.90E\text{-}8/py \\ \varepsilon = 1.30E\text{-}6/py \end{cases}$$

$$T_2MQFH - \begin{cases} \gamma = 1.50E\text{-}6/py \\ \beta = 2.19E\text{-}6/py \\ \varepsilon = 1.50E\text{-}6/py \end{cases}$$

$$S_3FH - \begin{cases} \gamma = 1.30E\text{-}6/py \\ \beta = 1.90E\text{-}8/py \\ \varepsilon = 1.30E\text{-}6/py \end{cases}$$

BWR

$$T_1PQE - \begin{cases} \gamma = 8.77E\text{-}9/py \\ \delta = 8.77E\text{-}9/py \end{cases}$$

$$T_{23}PQE - \begin{cases} \gamma = 1.57E\text{-}7/py \\ \delta = 1.57E\text{-}7/py \end{cases}$$

TABLE 1. (contd)

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-2 = 2.80E-6/py	BWR-3 = 1.65E-7/py
PWR-3 = 2.80E-6/py	BWR-4 = 1.65E-7/py
PWR-4 = 4.09E-8/py	
PWR-5 = 4.09E-8/py	
PWR-6 = 2.80E-6/py	
PWR-7 = 2.80E-6/py	

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

PWR: $\bar{F}^*$ = 1.13E-5/py   BWR: $\bar{F}^*$ = 3.3E-7/py

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

PWR: W^* = 29 man-rem/py   BWR: W^* = 0.95 man-rem/py

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

PWR: $\Delta\bar{F}$ = 5.6E-6/py   BWR: $\Delta\bar{F}$ = 1.7E-7/py

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

PWR: ΔW = 15 man-rem/py   BWR: 0.47 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
3.9E+4	3.4E+6	0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Research on Small-Break LOCAs and Anomalous Transients (II.E.2.2)

TABLE 2. (contd)

2. Affected Plants (N):

PWRs:	Operating	47	BWRs:	Operating	24
	Planned	43		Planned	20
		90			44

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

		$\bar{T}$ (yr)			$\bar{T}$ (yr)
PWRs:	Operating	27.7	BWRs:	Operating	25.2
	Planned	30		Planned	30
	All	28.8		All	27.4

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

$$\Delta(\bar{F}D_R)_{PWR} = (19{,}900 \text{ man-rem})(5.6E\text{-}6/py) = 0.11 \text{ man-rem/py}$$
$$\Delta(\bar{F}D_R)_{BWR} = (19{,}900 \text{ man-rem})(1.7E\text{-}7/py) = 0.0034 \text{ man-rem/py}$$

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
290	5300	0

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

100 man-hr/plant is assumed, in a 15 mR/hr radiation field, to modify instrumentation in operating plants.

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

$$D = (100 \text{ man-hr/plant})(0.015 \text{ R/hr}) = 1.5 \text{ man-rem/plant}$$

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

$$ND = 71(1.5 \text{ man-rem/plant}) = 107 \text{ man-rem}$$

TABLE 2. (contd)

9- 11. Steps Related to Occupational Dose Increase for SIR Operation and Maintenance:

No additional labor in radiation zones is foreseen for SIR operation and maintenance. Thus $D_O = 0$.

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
107	320	36

3.0 SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. Results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Research on Small-Break LOCAs and Anomalous Transients (II.E.2.2)

2. Affected Plants (N):

PWRs:	Operating	47	BWRs:	Operating	24
	Planned	43		Planned	20
		90			44

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

		$\bar{T}$ (yr)			$\bar{T}$ (yr)
PWRs:	Operating	27.7	BWRs:	Operating	25.2
	Planned	30		Planned	30
	All	28.8		All	27.4

TABLE 3. (contd)

Industry Costs (Steps 4 through 12)

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

$$\Delta(\bar{F}A)_{PWR} = (\$1.65E+9)(5.6E-6/py) = \$9200/py$$
$$\Delta(\bar{F}A)_{BWR} = (\$1.65E+9)(1.7E-7/py) = \$280/py$$

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

Best Estimate	Upper Bound	Lower Bound
\$2.4E+7	\$4.4E+8	0

6. Per-Plant Industry Resources for SIR Implementation:

Training and installation = 3 man-yr/plant

Equipment (cost estimated directly in next step)

(These apply to operating plants only; see Attachment 1.)

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

Training and installation = (3 man-yr/plant)(\$1.0E+5/man-yr)	=	\$3.0E+5/plant
Equipment	=	\$2.0E+5/plant
	I =	\$5.0E+5/plant

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

NI = 71(\$5.0E+5/plant) = \$3.6E+7

9-11. Steps Related to Industry Cost for SIR Operation and Maintenance:

Training requirements are assumed to be integrated with or simply to replace existing requirements. No additional labor is assumed; thus, $I_o = 0$.

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$3.6E+7	\$5.3E+7	\$1.8E+7

TABLE 3. (contd)

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Costs are estimated directly in next step.

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

$3.1E+6 (See Attachment 1.)

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

Costs are estimated directly in Step 17.

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

See Step 17.

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

NC = $2.0E+5 (see Attachment 1.)

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

1 man-day/py

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

C_O = (1 man-day/py)(1 man-wk/5 man-days)($2270/man-wk) = $454/py

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

$N\bar{T}C_O$ = [90(28.8 yr) + 44(27.4 yr)]($454/py) = $1.7E+6

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$5.0E+6	$6.8E+6	$3.2E+6

ATTACHMENT 1

TMI Task II.E.2.2 (Research on Small-Break LOCAs and Anomalous Transients) is ongoing and, therefore, already has sunk costs associated with it. Cost and manpower information taken from the TMI Action Plan (NUREG-0660 1980) and from discussions with NRC technical contacts for this issue is shown below. Estimates are identified as such.

Organization	FY80	FY81	FY82	FY83
RES Contractor (LOFT)	$5.3E+7	$4.6E+7	$4.2E+7	$1.5E+7
RES	8.2 man-yr	8.0 man-yr	7.0 man-yr[a]	2.0 man-yr[a]
Total NRR	0.3 man-yr	0.5 man-yr	0.3 man-yr[a]	0.1 man-yr[a]
Total AOM	$600K	$800K	$600K[a]	$300K[a]

(a) Estimate.

Assuming $100,000/man-year for NRC costs, the FY83 expenditure comes to $1.55E+7. However, these costs represent total budget costs to the LOFT facility. It is estimated that the small-break LOCA program represents approximately 20 percent of the total research effort currently underway, or $3.1E+6.

For implementation of this SIR, it is assumed that an additional $200,000 is required to establish new criteria for reactor instrumentation and operator training. Annual review requirements by the NRC, beyond those already required, are assumed to be minimal. One man-day per plant-year is assumed here.

For the estimate of utility costs, it is assumed that 2 man-years of effort (at $100,000/man-yr) are required for training at each facility, plus $200,000 for upgrades in advanced control room equipment. One additional man-year is assumed for equipment installation, bringing the estimate to $500,000 per facility for upgrade of operator capabilities and instrumentation. It is assumed that equipment installation occurs primarily in the control room, with no increase in radiation exposure. These costs are applied to operating plants only, since the change will presumably be incorporated into the initial design of new plants.

The costs assumed above can be compared with those assumed in safety issue I.C.9, Long-Term Program Plan for Upgrading of Procedures. Here, a utility effort of 5 man-year per facility is estimated to produce a 30 percent reduction in all operator and maintenance error probabilities. In this issue, primary emphasis is placed on operator performance alone, and further limited to errors during transient and LOCA sequences.

REFERENCE

U.S. NRC. 1980. NRC Action Plan Developed as a Result of the TMI-2 Accident. NUREG-0660, U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: II.E.6, In-Situ Testing of Valves

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

The purpose of this issue is to establish the adequacy of current requirements for valve testing in providing assurance of safety-related valve functions. For purposes of issue resolution, it is assumed that a study is conducted and the results indicate that additional valve testing is recommended. It is further assumed that a program of additional valve testing and maintenance is instituted for all reactors.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	3.1E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	8.4E+4
Total of Above =	8.4E+4
Accident Avoidance =	180

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	2.8
SIR Operation/Maintenance =	42
Total of Above =	44
Accident Avoidance =	15
NRC COSTS:	
SIR Development =	0.17
SIR Implementation Support =	0.30
SIR Operation/Maintenance Review =	7.3
Total of Above =	7.7

IN-SITU TESTING OF VALVES

ISSUE II.E.6

1.0 SAFETY ISSUE DESCRIPTION

The purpose of this issue is to establish the adequacy of current requirements for valve testing in providing assurance of safety-related valve functions. Valve performance is critical to the successful functions of a large number of the plants' safety systems. A study was proposed which would result in recommendations for alternate means of verifying performance requirements.

For issue resolution, it is assumed that a study is conducted for both PWRs and BWRs which results in a recommendation that additional testing and maintenance be performed on all safety-related valves. It is further assumed that a program is instituted for this purpose at all reactors.

This issue was analyzed as an independent item. However, significant overlap appears to exist between this issue and II.D.2, "Test Requirements for Coolant System Valves;" B-58, "Passive Mechanical Failures;" and C-11, "Assessment of Failure and Reliability of Pumps and Valves." It is suggested that these issues be considered together to avoid double-counting of risk reductions.

2.0 SAFETY ISSUE RISK AND DOSE

The analyses of public risk reduction and occupational dose associated with the safety issue resolution (SIR) are estimated in this section. The results are summarized in Tables 1 and 2, respectively.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

In-Situ Testing of Valves (II.E.6)

TABLE 1. (contd)

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

All PWRs and BWRs are affected.

		N	$\bar{T}$(yr)	
PWR:	planned	43	30.0	28.8
	operating	47	27.7	
BWR:	planned	20	30.0	27.4
	operating	24	25.2	
	Total	134		

3. Plants Selected for Analysis:

Oconee 3 - representative PWR

Grand Gulf 1 - representative BWR

4. Parameters Affected by SIR:

All safety-related valves are assumed to be affected by issue resolution. These valves are designed to perform their intended functions under all postulated plant conditions (Seabrook Station FSAR 1981). The following list includes all affected elements of the dominant minimal cut sets for Oconee and Grand Gulf dominant accident sequences:

Oconee 3: B, C, D, E, CONST1, CONST2, A1, B1, C1, O, O·E, W·X, B·W, C·X, D·X, E·W, B·O, E·C

Grand Gulf 1: H, P,, R, L, LAZ, LB1, LB2, LC, VGA1, VGA2, VGB1, VGB2, SA, SB, SSA, SSB, SSC, V1, V2, V3, SCVA, SCVB.

5. Base-Case Values for Affected Parameters:

Base-case values remain unchanged from original values - refer to Guidelines, Tables A.4 and B.4 (Andrews et al. 19B3).

6. Affected Accident Sequences and Base-Case Frequencies:

Oconee:

$$T_2MLU - \begin{cases} \gamma(\text{PWR-3}) = 4.8\text{E-}7/\text{py} \\ \beta(\text{PWR-5}) = 6.9\text{E-}9/\text{py} \\ \varepsilon(\text{PWR-7}) = 4.8\text{E-}7/\text{py} \end{cases}$$

$$T_1MLU - \begin{cases} \gamma(\text{PWR-3}) = 9.5\text{E-}7/\text{py} \\ \beta(\text{PWR-5}) = 1.4\text{E-}8/\text{py} \\ \varepsilon(\text{PWR-7}) = 9.5\text{E-}7/\text{py} \end{cases}$$

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

$$T_2MQH - \begin{cases} \gamma(\text{PWR-3}) & = & 5.2\text{E-6/py} \\ \beta(\text{PWR-5}) & = & 7.6\text{E-8/py} \\ \varepsilon(\text{PWR-7}) & = & 5.2\text{E-6/py} \end{cases}$$

$$S_3H - \begin{cases} \gamma(\text{PWR-3}) & = & 5.9\text{E-7/py} \\ \beta(\text{PWR-5}) & = & 8.6\text{E-9/py} \\ \varepsilon(\text{PWR-7}) & = & 5.9\text{E-7/py} \end{cases}$$

$$S_1D - \begin{cases} \alpha(\text{PWR-1}) & = & 5.3\text{E-8/py} \\ \gamma(\text{PWR-3}) & = & 1.1\text{E-6/py} \\ \beta(\text{PWR-5}) & = & 3.8\text{E-8/py} \\ \varepsilon(\text{PWR-7}) & = & 4.2\text{E-6/py} \end{cases}$$

$$T_2MQFH - \begin{cases} \gamma(\text{PWR-2}) & = & 2.4\text{E-6/py} \\ \beta(\text{PWR-4}) & = & 3.4\text{E-8/py} \\ \varepsilon(\text{PWR-6}) & = & 2.4\text{E-6/py} \end{cases}$$

$$S_3FH - \begin{cases} \gamma(\text{PWR-2}) & = & 9.0\text{E-8/py} \\ \beta(\text{PWR-4}) & = & 1.3\text{E-9/py} \\ \varepsilon(\text{PWR-6}) & = & 9.0\text{E-8/py} \end{cases}$$

$$S_2FH - \begin{cases} \alpha(\text{PWR-1}) & = & 5.7\text{E-10/py} \\ \beta(\text{PWR-4}) & = & 4.2\text{E-10/py} \\ \varepsilon(\text{PWR-6}) & = & 4.6\text{E-8/py} \end{cases}$$

$$S_2D - \begin{cases} \alpha(\text{PWR-1}) & = & 6.9\text{E-9/py} \\ \gamma(\text{PWR-3}) & = & 1.4\text{E-7/py} \\ \beta(\text{PWR-5}) & = & 5.1\text{E-9/py} \\ \varepsilon(\text{PWR-7}) & = & 5.5\text{E-7/py} \end{cases}$$

$$S_3D - \begin{cases} \gamma(\text{PWR-3}) & = & 6.3\text{E-7/py} \\ \beta(\text{PWR-5}) & = & 9.2\text{E-9/py} \\ \varepsilon(\text{PWR-7}) & = & 6.3\text{E-7/py} \end{cases}$$

$$T_2MQD - \begin{cases} \gamma(\text{PWR-3}) & = & 7.6\text{E-7/py} \\ \beta(\text{PWR-5}) & = & 1.1\text{E-8/py} \\ \varepsilon(\text{PWR-7}) & = & 7.6\text{E-7/py} \end{cases}$$

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies (contd):

Grand Gulf:

T_1PQI - $\begin{cases} \alpha\,(BWR\text{-}1) = 1.5E\text{-}8/py \\ \gamma\,(BWR\text{-}2) = 1.5E\text{-}6/py \end{cases}$

$T_{23}PQI$ - $\begin{cases} \alpha\,(BWR\text{-}1) = 3.7E\text{-}8/py \\ \delta\,(BWR\text{-}2) = 3.7E\text{-}6/py \end{cases}$

T_1PQE - $\begin{cases} \alpha\,(BWR\text{-}3) = 9.5E\text{-}8/py \\ \delta\,(BWR\text{-}4) = 9.5E\text{-}8/py \end{cases}$

$T_{23}PQE$ - $\begin{cases} \alpha\,(BWR\text{-}3) = 2.6E\text{-}7/py \\ \delta\,(BWR\text{-}4) = 2.6E\text{-}7/py \end{cases}$

SI - $\begin{cases} \alpha\,(BWR\text{-}1) = 4.6E\text{-}8/py \\ \delta\,(BWR\text{-}2) = 4.6E\text{-}6/py \end{cases}$

T_1QW - $\delta\,(BWR\text{-}2) = 4.5E\text{-}6/py$

$T_{23}QW$ - $\delta\,(BWR\text{-}2) = 1.1E\text{-}5/py$

T_1QUV - $\begin{cases} \gamma\,(BWR\text{-}3) = 9.2E\text{-}7/py \\ \delta\,(BWR\text{-}4) = 9.2E\text{-}7/py \end{cases}$

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 6.0E-8/py
PWR-2 = 2.5E-6/py
PWR-3 = 9.9E-6/py
PWR-4 = 3.6E-8/py
PWR-5 = 1.7E-7/py
PWR-6 = 2.5E-6/py
PWR-7 = 1.3E-5/py

BWR-1 = 9.8E-8/py
BWR-2 = 2.5E-5/py
BWR-3 = 1.3E-6/py
BWR-4 = 1.3E-6/py

TABLE 1. (contd)

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

$\bar{F}_{PWR}$ = 2.85E-5/py $\bar{F}_{BWR}$ = 2.79E-5/py

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

W_{PWR} = 66.1 man-rem/py W_{BWR} = 185 man-rem/py

10. Adjusted-Case Values for Affected Parameters:

It is assumed that the proposed testing program results in a 5% reduction in safety-related valve failure probabilities. Because such contributory modes as hardware failures, control circuitry failures, plugging and test outages could be affected, the reduction in probability was applied to the valve failure probabilities as a whole in the dominant minimal cut set elements.

The following is a list of the adjusted case values for affected parameters:

Oconee:

B = C	=	3.14E-3
D = E	=	2.19E-2
CONST1	=	1.83E-4
CONST2	=	5.70E-4
A1 = C1	=	9.31E-3
B1	=	3.40E-2
Q	=	4.75E-2
D E	=	4.42E-4
W·X	=	7.94E-5
B·W	= C·X =	2.44E-5
D·X	= E·W =	1.90E-4
B·D	= E·C =	5.69E-5

TABLE 1. (contd)

10. Adjusted-Case Values for Affected Parameters (contd):

Grand Gulf:

H	=	2.05E-2
P	=	9.50E-2
R	=	4.91E-2
L	=	2.06E-2
LA2 = LB2	=	1.37E-2
LB1	=	1.24E-2
LC	=	2.08E-2
VGA1 = VGB1	=	1.43E-2
VGA2 = VGB2	=	2.28E-2
SA = SB	=	1.33E-2
SSA = SSB	=	1.99E-2
SSC	=	1.37E-2
V1 = V2	=	7.60E-3
V3	=	3.14E-3
SCVA = SCVB	=	3.04E-2

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Oconee:

$$T_2MLU - \begin{cases} \gamma(PWR\text{-}3) = 4.1E\text{-}7/py \\ \beta(PWR\text{-}5) = 6.0E\text{-}9/py \\ \varepsilon(PWR\text{-}7) = 4.1E\text{-}7/py \end{cases}$$

$$T_1MLU - \begin{cases} \gamma(PWR\text{-}3) = 8.6E\text{-}7/py \\ \beta(PWR\text{-}5) = 1.3E\text{-}8/py \\ \varepsilon(PWR\text{-}7) = 8.6E\text{-}7/py \end{cases}$$

$$T_2MQH - \begin{cases} \gamma(PWR\text{-}3) = 4.9E\text{-}6/py \\ \beta(PWR\text{-}5) = 7.1E\text{-}8/py \\ \varepsilon(PWR\text{-}7) = 4.9E\text{-}6/py \end{cases}$$

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

$$S_3H - \begin{cases} \gamma\,(PWR\text{-}3) = 5.3E\text{-}7/py \\ \beta\,(PWR\text{-}5) = 7.8E\text{-}9/py \\ \varepsilon\,(PWR\text{-}7) = 5.3E\text{-}7/py \end{cases}$$

$$S_1D - \begin{cases} \alpha\,(PWR\text{-}1) = 5.0E\text{-}8/py \\ \gamma\,(PWR\text{-}3) = 1.0E\text{-}6/py \\ \beta\,(PWR\text{-}5) = 3.7E\text{-}8/py \\ \varepsilon\,(PWR\text{-}7) = 4.0E\text{-}6/py \end{cases}$$

$$T_2MQFH - \begin{cases} \gamma\,(PWR\text{-}2) = 2.2E\text{-}6/py \\ \beta\,(PWR\text{-}4) = 3.3E\text{-}8/py \\ \varepsilon\,(PWR\text{-}6) = 2.2E\text{-}6/py \end{cases}$$

$$S_3FH - \begin{cases} \gamma\,(PWR\text{-}2) = 8.3E\text{-}8/py \\ \beta\,(PWR\text{-}4) = 1.2E\text{-}9/py \\ \varepsilon\,(PWR\text{-}6) = 8.3E\text{-}8/py \end{cases}$$

$$S_2FH - \begin{cases} \alpha\,(PWR\text{-}1) = 5.1E\text{-}10/py \\ \beta\,(PWR\text{-}4) = 3.7E\text{-}10/py \\ \varepsilon\,(PWR\text{-}6) = 4.1E\text{-}8/py \end{cases}$$

$$S_2D - \begin{cases} \alpha\,(PWR\text{-}1) = 6.3E\text{-}9/py \\ \gamma\,(PWR\text{-}3) = 1.3E\text{-}7/py \\ \beta\,(PWR\text{-}5) = 4.6E\text{-}9/py \\ \varepsilon\,(PWR\text{-}7) = 5.0E\text{-}7/py \end{cases}$$

$$S_3D - \begin{cases} \gamma\,(PWR\text{-}3) = 5.8E\text{-}7/py \\ \beta\,(PWR\text{-}5) = 8.5E\text{-}9/py \\ \varepsilon\,(PWR\text{-}7) = 5.8E\text{-}7/py \end{cases}$$

$$T_2MQD - \begin{cases} \gamma\,(PWR\text{-}3) = 6.8E\text{-}7/py \\ \beta\,(PWR\text{-}5) = 9.9E\text{-}9/py \\ \varepsilon\,(PWR\text{-}7) = 6.8E\text{-}7/py \end{cases}$$

Grand Gulf:

$$T_1PQI - \begin{cases} \alpha\,(BWR\text{-}1) = 1.3E\text{-}8/py \\ \delta\,(BWR\text{-}2) = 1.3E\text{-}6/py \end{cases}$$

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

$T_{23}PQI$ - $\begin{cases} \alpha\text{(BWR-1)} = 3.2E\text{-}8/py \\ \delta\text{(BWR-2)} = 3.2E\text{-}6/py \end{cases}$

T_1PQE - $\begin{cases} \alpha\text{(BWR-3)} = 8.2E\text{-}8/py \\ \delta\text{(BWR-4)} = 8.2E\text{-}8/py \end{cases}$

$T_{23}PQE$ - $\begin{cases} \gamma\text{(BWR-3)} = 2.4E\text{-}7/py \\ \delta\text{(BWR-4)} = 2.4E\text{-}7/py \end{cases}$

SI - $\begin{cases} \alpha\text{(BWR-1)} = 4.2E\text{-}8/py \\ \delta\text{(BWR-2)} = 4.2E\text{-}6/py \end{cases}$

T_1QW - δ(BWR-2) = 4.1E-6/py

$T_{23}QW$ - δ(BWR-2) = 9.7E-6/py

T_1QUV - $\begin{cases} \gamma\text{(BWR-3)} = 8.0E\text{-}7/py \\ \delta\text{(BWR-4)} = 8.0E\text{-}7/py \end{cases}$

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 5.7E-8/py
PWR-2 = 2.3E-6/py
PWR-3 = 9.1E-6/py
PWR-4 = 3.4E-8/py
PWR-5 = 1.6E-7/py
PWR-6 = 2.3E-6/py
PWR-7 = 1.3E-5/py

BWR-1 = 8.7E-8/py
BWR-2 = 2.3E-5/py
BWR-3 = 1.1E-6/py
BWR-4 = 1.1E-6/py

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

$\bar{F}^*_{PWR}$ = 2.65E-5/py $\bar{F}^*_{BWR}$ = 2.49E-5/py

TABLE 1. (contd)

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

W^*_{PWR} = 61.1 man-rem/py W^*_{BWR} = 170 man-rem/py

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

$\Delta\bar{F}_{PWR}$ = 2.0E-6/py $\Delta\bar{F}_{BWR}$ = 3.0E-6/py

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

ΔW_{PWR} = 5.0 man-rem/py ΔW_{BWR} = 15 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
3.1E+4	1.2E+7	0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

In-Situ Testing of Values (II.E.6)

2. Affected Plants (N):

		N
PWR:	planned	43
	operating	47
BWR:	planned	20
	operating	24
	Total	134

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

		$\bar{T}$(yr)	
PWR:	planned	30.0	28.8
	operating	27.7	
BWR:	planned	30.0	27.4
	operating	25.2	

TABLE 2. (contd)

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

$\Delta(\bar{F}D_R)_{PWR}$ = (2.0E-6/py)(19,900 man-rem) = 4.0E-2 man-rem/py

$\Delta(\bar{F}D_R)_{BWR}$ = (3.0E-6/py)(19,900 man-rem) = 6.0E-2 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
280	1.3E+4	0

6-8. Labor Hours in a Radiation Zone for Implementation:

It is assumed that no labor hours are required for implementation.

D = 0.

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

It is assumed that increased surveillance testing and maintenance of safety-related valves require an additional 8 man-wk/py of utility labor for a PWR and 6 man-wk/py for a BWR. The PWRs are assumed to have more safety-related valves than the BWRs. A utilization factor of 75% for actual work in the radiation zone translates into 6 man-wk/py for a PWR and 4.5 man-wk/py for a BWR.

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

Assuming a 0.10 R/hr radiation field:

D_o(PWR) = (6 man-wk/py)(40 man-hr/man-wk)(0.10 R/hr) = 24 man-rem/py

D_o(BWR) = (4.5 man-wk/py)(40 man-hr/man-wk)(0.10 R/hr) = 18 man-rem/py

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

$N\bar{T}D_o$ = 90(28.8 yr)(24 man-rem/py) + 44(27.4 yr)(18 man-rem/py) = 8.39E+4 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
8.4E+4	2.5E+5	2.8E+4

3.0 SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. The results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   In-Situ Testing of Valves (II.E.6)

2. Affected Plants (N):

		N
PWR:	planned	43
	operating	47
BWR:	planned	20
	operating	24
	Total	134

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

		$\bar{T}$(yr)	
PWR:	planned	30.0	28.8
	operating	27.7	
BWR:	planned	30.0	27.4
	operating	25.2	

Industry Costs (Steps 4 through 12)

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

   $\Delta(\bar{F}D_R)_{PWR}$ = (\$1.65E+9)(2.0E-5/py) = \$3.30E+3/py

   $\Delta(\bar{F}D_R)_{BWR}$ = (\$1.65E+9)(3.0E-6/py) = \$4.95E+3/py

TABLE 3. (contd)

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

Best Estimate	Upper Bound	Lower Bound
\$1.5E+7	\$1.1E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

PWR: Labor (engineering, clerks, etc.) = 10 man-wk/plant

BWR: Labor (engineering, clerks, etc.) = 8 man-wk/plant

This difference arises from the smaller number of safety-related valves in BWRs.

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

I_{PWR} = (10 man-wk/plant)(\$2270/man-wk) = \$2.27 E+4/plant

I_{BWR} = (8 man-wk/plant)(\$2270/man-wk) = \$1.82 E+4/plant

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

NI = 90(\$2.27E+4/plant) + 44(\$1.82E+4/plant) = \$2.84E+6

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

PWR: Labor (additional testing and maintenance) = 16 man-wk/py

BWR: Labor (additional testing and maintenance) = 12 man-wk/py

This difference arises from the smaller number of values in BWRs.

The 5% reduction in the failure probabilities of safety-related values (Step 10, Table 1) is expected to result in a decrease in scheduled outages requiring replacement power. In Issue C-11 (Assessment of Failure and Reliability of Pumps and Values), plant outage time is estimated at two months per year, with 5% of this time attributable to pump and valve malfunctions. Assuming that half of this can be attributed to valves alone, it is estimated that

(0.05)(2 mo/yr)(30 days/mo)/2 = 1.5 days/yr

of outage time per plant results from valve failures. Thus, SIR will presumably reduce this outage time by 5%, or (0.05)(1.5) = 0.075 day/yr at each plant. This estimate is the same for both PWRs and BWRs.

TABLE 3. (contd)

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_o):

I_o(PWR) = (16 man-wk/py)($2270/man-wk) + (-0.075 day/py) ($3.0E+5/day) = $1.38E+4/py

I_o(BWR) = (12 man-wk/py)($2270/man-wk) + (-0.075 day/py) ($3.0E+5/day) = $4740/py

(Negative signs indicate reductions.)

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

$N\bar{T}I_o$ = 90(28.8 yr)($1.38E+4/py) + 44(27.4 yr)($4740/py) = $4.15E+7

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$4.4E+7	$6.5E+7	$2.4E+7

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Generic issue development:

All PWRs = 50 man-wk
All BWRs = 25 man-wk
Total 75 man-wk

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

C_D = (75 man-wk)($2270/man-wk) = $1.70E+5

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

PWR and BWR implementation support will be approximately 1 man-wk/plant.

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

C = (1 man-wk/plant)($2270 man-wk) = $2270/plant

TABLE 3. (contd)

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

$$NC = 134(\$2270/\text{plant}) = \$3.04E+5$$

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

PWR: 1 man-wk/py

BWR: 0.5 man-wk/py

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

$$Co_{PWR} = (1 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$2270/\text{py}$$

$$Co_{BWR} = (0.5 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$1135/\text{py}$$

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

$$N\bar{T}C_o = 90(28.8 \text{ yr})(\$2270/\text{py}) + 44(27.4 \text{ yr})(\$1135/\text{py}) \approx \$7.25E+6$$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$7.7E+6	\$1.1E+7	\$4.1E+6

REFERENCES

Andrews, W. B. et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

Public Service Company of New Hampshire. 1981. Seabrook Station: Final Safety Analysis Report. Sections 3.9(B), 3.2 and 6.2.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: II.F.5, Instrumentation and Controls: Classification of Instrumentation, Control, and Electrical Equipment

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

The overall objective of this issue to provide a method for classifying non-IE instrumentation, control, and electrical systems with respect to their importance to safety, and to provide a design basis and standards for such systems and components. The proposed resolution assumes a program to upgrade important non-IE systems identified via improved reliability, redundancy, and environmental qualification.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	2.2E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	3.1E+2
SIR Operation/Maintenance =	0
Total of Above =	3.1E+2
Accident Avoidance =	1.3E+2

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	1.0E+2
SIR Operation/Maintenance =	-4.3E+2
Total of Above =	-3.3E+2
Accident Avoidance =	1.1E+1
NRC COSTS:	
SIR Development =	4.0E-2
SIR Implementation Support =	1.3E+1
SIR Operation/Maintenance Review =	0
Total of Above =	1.3E+1

INSTRUMENTATION AND CONTROLS:

CLASSIFICATION OF INSTRUMENTATION, CONTROL AND ELECTRICAL EQUIPMENT

ISSUE II.F.5

1.0 SAFETY ISSUE DESCRIPTION

The purpose of this issue is to prepare a standard that will provide a classification approach for determining the applicability of design criteria and design requirements for instrumentation, control, and electrical systems and equipment, based on their level of importance to safety. This work has been initiated with the issuance of IEEE-827, Trial Use Guide: A Method for Determining Requirements for Instrumentation, Control, and Electrical Systems Important to Safety (IEEE 1981). This guide is based to a large degree on the draft version of IAEA Safety Series No. 50-SG-D8, Safety-Related Instrumentation and Control Systems (1982).

The IEEE guide is directed at instrumentation and controls not covered by IEEE Standard 603 (IEEE 1980), the latter covering IE safety systems. As such, IEEE-827 does not cover the primary safety functions of a light water reactor. However, safety-related non-IE systems may indirectly impact plant risk.

The primary goal of the regulatory guide will be to identify systems not covered by IEEE-603 that perform functions important to plant safety, and to identify those systems where failure could lead to events more severe than design basis events. The primary systems of concern are power conversion, fire detection and prevention, security, and communications.

Issue II.F.5 will not by itself reduce plant risk by issuing a regulatory guide based on IEEE-827. It must be assumed that a program of utility conformance to the standards is implemented with a research program backing the choice of important safety criteria for reliability, redundancy, and environmental qualifications. This issue has a number of goals and requirements in common with other issues in Task II.F (Instrumentation and Controls), I.D (Control Room Design), No. 57 (Effects of Fire Protection System Actuation on Safety-Related Equipment Systems) and No. A-47 (Safety Implications of Control Systems). The delineation of scope is discussed further in Attachment 1.

PROPOSED RESOLUTION

This proposed safety issue resolution (SIR) is to produce a regulatory guide to classify non-IEEE-603 systems and equipment, and to implement utility upgrades in accordance with this guide.

AFFECTED PLANTS

This issue affects all PWRs and BWRs.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose are estimated in this section and summarized in Tables 1 and 2, respectively.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Instrumentation and Controls: Classification of Instrumentation, Control, and Electrical Equipment (II.F.5)

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

	N	$\bar{T}$ (yr)
PWR	90	28.8
BWR	44	27.4

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf 1 - representative BWR

4. Parameters Affected by SIR:

   A program to classify and upgrade non-IE instrumentation, controls, and electrical systems is assumed to improve balance of plant reliability, thus reducing transient frequencies. Relationships to other safety issues and assumptions are discussed in Attachment 1.

   PWR: T_2, T_3

   BWR: T_{23}

5. Base-Case Values for Affected Parameters:

   All parameters have the original values as given in Tables A.4 (PWR) and B.4 (BWR) (Andrews et al. 1983).

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies:

For the PWR, all accident sequences initiated by T_2 and T_3 are affected, i.e.,

T_2MLU - γ, β, ε

T_2MQH - γ, β, ε

T_2MQFH - γ, β, ε

T_2MLUO - γ, β, ε

T_2KMU - γ, β, ε

T_3MLUO - γ, β, ε

T_2MQD - γ, β, ε

For the BWR, all accident sequences initiated by T_{23} are affected, i.e.,

T_{23}PQI - α, δ

T_{23}PQE - α, δ

T_{23}QW - δ

T_{23}C - δ

All of the above have base-case frequencies equal to the original values in Appendices A and B of NUREG/CR-2800.

7. Affected Release Categories and Base-Case Frequencies:

PWR-2 = 2.51E-6/py	BWR-1 = 3.68E-8/py
PWR-3 = 1.55E-5/py	BWR-2 = 2.11E-5/py
PWR-4 = 3.66E-8/py	BWR-3 = 2.72E-7/py
PWR-5 = 2.26E-7/py	BWR-4 = 2.72E-7/py
PWR-6 = 2.51E-6/py	
PWR-7 = 1.55E-5/py	

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

$\bar{F}$(PWR) = 3.63E-5/py $\bar{F}$(BWR) = 2.17E-5/py

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

W(PWR) = 9.65E+1 man-rem/py W(BWR) = 1.51E+2 man-rem/py

10. Adjusted-Case Values for Affected Parameters:

Based on the discussions in Attachment 1, a 6% reduction in transients other than loss of offsite power is assumed for T_2 and T_3 transients for PWRs, and a 4% reduction for T_{23} transients for BWRs.

PWR: T_2 = 2.82/py   BWR: T_{23} = 6.72/py

T_3 = 3.76/py

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Sequence		Base-Case Frequency (1/py)
PWR:		
T_2MLU -	γ (PWR-3)	5.50E-7
	β (PWR-5)	8.03E-9
	ε (PWR-7)	5.50E-7
T_2MQH -	γ (PWR-3)	5.35E-6
	β (PWR-5)	7.81E-8
	ε (PWR-7)	5.35E-6
T_2MQFH -	γ (PWR-2)	2.36E-6
	β (PWR-4)	3.44E-8
	ε (PWR-6)	2.36E-6
T_2MLUO -	γ (PWR-3)	3.81E-6
	β (PWR-5)	5.56E-8
	ε (PWR-7)	3.81E-6
T_2KMU -	γ (PWR-3)	3.67E-6
	β (PWR-5)	5.35E-8
	ε (PWR-7)	3.67E-6
T_3MLUO -	γ (PWR-3)	4.98E-7
	β (PWR-5)	7.28E-9
	ε (PWR-7)	4.98E-7

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

Sequence		Base-Case Frequency (1/py)
PWR:		
T_2MQD -	γ (PWR-3)	7.12E-7
	β (PWR-5)	1.04E-8
	ϵ (PWR-7)	7.12E-7
BWR:		
$T_{23}PQI$ -	α (PWR-1)	3.53E-8
	δ (PWR-2)	3.53E-6
$T_{23}PQE$ -	γ (BWR-3)	2.61E-7
	δ (BWR-4)	2.61E-7
$T_{23}QW$ -	δ (BWR-2)	1.15E-5
$T_{23}C$ -	δ (BWR-2)	5.18E-6

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-2 = 2.36E-6/py   BWR-1 = 3.53E-8/py
PWR-3 = 1.46E-5/py   BWR-2 = 2.02E-5/py
PWR-4 = 3.44E-6/py   BWR-3 = 2.61E-7/py
PWR-5 = 2.13E-7/py   BWR-4 = 2.61E-7/py
PWR-6 = 2.36E-6/py
PWR-7 = 1.46E-5/py

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

$\bar{F}^*(PWR) = 3.42E-5/py$ $\bar{F}^*(BWR) = 2.08E-5/py$

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

$W^*(PWR) = 9.07E+1$ man-rem/py $W^*(BWR) = 1.45E+2$ man-rem/py

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

$\Delta\bar{F}(PWR) = 3.63E-5/py - 3.42E-5/py = 2.1E-6/py$

$\Delta\bar{F}(BWR) = 2.17E-5/py - 2.08E-5/py = 9.0E-7/py$

TABLE 1. (contd)

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

$\Delta W(PWR)$ = 9.65E+1 man-rem/py - 9.07E+1 man-rem/py = 5.8 man-rem/py

$\Delta W(BWR)$ = 1.51E+2 man-rem/py - 1.45E+2 man-rem/py = 6.0 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.2E+4	1.3E+7	0

ATTACHMENT 1

The instrumentation, control, and electrical systems and equipment (ICE/SE) covered by IEEE-827, 1981, (non-IE) cover a wide range of systems in the plant. Typical examples given include the ICE/SE associated with the power conversion system and the fire protection system, as well as security and communications. The primary example given by the past chairman of the IEEE-827 working committee (J.M. Gallagher, Jr., Westinghouse Electric Corp.) is the Safety Parameter Display System (SPDS) developed for accident monitoring. Issue II.F.5 interfaces with a number of other issues dealing with control and instrumentation, as shown below.

Issues Dealing with Instrumentation and Control

II.F.1	Additional Accident Monitoring Instrumentation
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling
II.F.3	Instruments for Monitoring Accident Conditions (Reg. Guide 1.97)
II.F.4	Study and Control of Protective Action Design Requirements
II.F.5	Classification of Instruments, Control and Electrical Equipment
I.D.3	Safety System Status Monitoring
I.D.4	Control Room Design Standards
II.D.5	Improved Control Room Instrumentation Research
57	Effects of Fire Protection System Actuation on Safety-Related Equipment

Issues II.F.1 through II.F.4 are in various stages of implementation, and hence are beyond the scope of this review. However the Safety Parameter Display System is covered specifically in Issue II.F.3. Other aspects of control room instrumentation are covered in Task II.D, making it unlikely that Issue II.F.5 would make a significant contribution to plant safety solely because of instrumentation. Rather, it is assumed that the primary benefit of II.F.5 will result from improvements in control and electrical equipment in the balance of plant.

The primary non-IE systems identified in IEEE-827 were associated with the power conversion system and fire protection. However, specific action items address these areas. Safety Issue A-47, "Safety Implications of Control Systems," addresses the potential for failure of non-safety-grade systems initiating transients or making recovery from transients more difficult. These systems include reactivity control, reactor coolant parameters, secondary system pressure and flow controls, etc. As outlined in the PNL consideration of No. A-47, a Task Action Plan has been prepared (Szukiewicz 1982), and research programs are underway in the Office of Nuclear Regulatory Research and the Office of Nuclear Reactor Regulation to provide technical information needed for issue resolution. These programs are currently scheduled to be completed at the end of FY83.

The above work is focusing on the contribution of control system failures to vessel overfill transients in BWRs and steam generator overfill transients in PWRs. It is expected that II.F.5 will be more expansive than A-47, including non-IE instrumentation and electrical systems and equipment along with control systems. As a result, it is assumed that a reduction in transients will be realized by implementation of this safety issue.

The remaining non-IE system of importance mentioned above is the fire detection and protection system. This is recognized in the IAEA safety guide, as well as in some plant safety analysis reports, as a non-IE system which is capable of interfacing with the function of an IE safety system (i.e., the standby gas treatment system in BWRs). This concern is specifically covered in Safety Issue No. 57.

It is assumed that primary issue benefit will center around improved instrumentation, control, and electrical systems and equipment, primarily associated with the power conversion system. An approach similar to that used in Issue A-47 is assumed, where a review of EPRI data on previous transients (McClymont 1982) is used to identify transient categories and frequencies of interest. These are given below.

Transients of Interest for Instrumentation, Control, and Electrical Systems

EPRI Category	Item	Frequency (Events/py)
PWR		
2	Uncontrolled rod withdrawal	.02
3	Control rod drive mechanism problems and/or rod drop	.65
6	Low pressurizer pressure	.03
8	High pressurizer pressure	.03
9	Inadvertent safety injection signal	.03
10	Containment pressure problem	.01
11	CVCS malfunction - boron dilution	.04
12	P/T/power imbalance - rod position error	.16

ATTACHMENT 1 (contd)

EPRI Category	Item	Frequency (Events/py)
PWR (contd)		
15	Loss or reduction in feedwater flow (one loop)	1.88
16	Loss or reduction in feedwater flow (all loops)	.15
19	Increase in feedwater flow (one loop)	.69
20	Increase in feedwater flow (all loops)	.01
22	Feedwater flow instability - miscellaneous mechanical causes	.21
33	Turbine trip, throttle, valve closure, electro-hydraulic control problems	1.38
36	Pressurizer spray failure	.04
		5.33
	Total PWR transients (excluding loss of offsite power)	9.67
BWR		
3	Turbine trip	1.05
4	Turbine trip with bypass valve failure	.01
9	Pressure regulator fails open	.17
10	Pressure regulator fails closed	.17
12	Turbine bypass fails open	.06
13	Turbine bypass or control valves cause increased pressure (closed)	.42
14	Recirculation control failure - increasing flow	.23
15	Recirculation control failure - decreasing flow	.10
20	Feedwater - increasing flow at power	.16
22	Loss of all feedwater flow	.13
23	Trip of one feedwater (or condensate) pump	.14
24	Feedwater - low flow	.52
27	Rod withdrawal at power	.02
29	Inadvertent insertion of rod(s)	.12
		3.30
	Total BWR transients (excluding loss of offsite power)	8.78

Based on the assumption that 50 percent of failures are control-related and could be eliminated through an upgrade program, Issue A-47 focused on the important categories for PWRs (15, 33) and BWRs (3, 13, 24). Issue II.F.5 encompasses all electrical and instrumentation aspects of the equipment, as well as control, so all categories will be considered. From the table, this indicates (5.33/9.67) = 0.55 of PWR transients and (3.30/8.78) = 0.38 of BWR transients are of interest (excluding loss of offsite power). As in Issue A-47, it is again assumed that 50 percent of such transients are attributable to instrumentation, control, and electrical systems failures. This is consistent with failure data given for pumps and valves, etc. in the Oconee risk equations (Appendix A, Andrews et al. 1983).

It is assumed here that a program to classify systems with respect to safety and implement equipment upgrades will result in a 20 percent reduction in such failures, or a reduction in transient frequency of (0.55)(0.5)(0.2) ~ 0.06, or 6 percent, for PWRs, and (0.38)(0.5)(0.2) ~ 0.04, or 4 percent, for BWRs, for transients other then loss of offsite power.[a] These reductions are applied to T_2 and T_3 in the Oconee risk equations for PWRs, and T_{23} in the Grand Gulf equations for BWRs.

(a) Some overlap results, with the credit given for risk reduction by the SIR for A-47, since all control system failures in the dominant PWR and BWR categories (i.e., 50 percent of the total failures in these categories) were assumed to be eliminated in the A-47 SIR.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Instrumentation and Controls: Classification of Instrumentation, Control and Electrical Equipment (II.F.5)

2. Affected Plants (N):

		N			N
PWRs:	Operating	47	BWRs:	Operating	24
	Planned	43		Planned	20
	Total	90			40

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

		$\bar{T}$ (yr)			$\bar{T}$ (yr)
PWRs:	Operating	27.7	BWRs:	Operating	5.2
	Planned	30.0		Planned	30.0
	All	28.8		All	27.4

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

PWR = (19,900 man-rem)(2.1E-6/py) = 4.2E-2 man-rem/py

BWR = (19,900 man-rem)(9.0E-7/py) = 1.8E-2 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.3E+2	1.4E+4	0

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

Extensive examination of equipment may be required. One man-year per plant to update instruments, controls, and electrical equipment in possible radiation zones will be assumed for backfit LWRs.

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

It is assumed that the equipment for non-IE systems will be outside containment. A dose rate of 2.5 mR/hr is assumed.

D = (0.0025 man-yr/plant)(1760 man-hr/man-yr) = 4.4 man-rem/plant

TABLE 2. (contd)

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

ND = 312 man-rem

9-11. Steps Related to Occupational Dose Increase for SIR Operation and Maintenance:

No significant change in operation and maintenance requirements is anticipated as a result of this SIR. Thus, $D_0 = 0$.

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
3.1E+2	9.3E+2	1.0E+2

3.0 SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. Results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Instrumentation and Controls: Classification of Instrumentation, Control, and Electrical Equipment (II.F.5)

2. Affected Plants (N):

		N			N
PWRs:	Operating	47	BWRs:	Operating	24
	Planned	43		Planned	20
	All	90			44

TABLE 3. (contd)

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

		$\bar{T}$ (yr)			$\bar{T}$ (yr)
PWRs:	Operating	27.7	BWRs:	Operating	25.2
	Planned	30.0		Planned	30.0
	All	28.8		All	27.4

Industry Costs (Steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, $\Delta(\overline{FA})$:

PWR = ($1.65E+9)(2.1E-6/py) = $3.5E+3/py

BWR = ($1.65E+9)(9.0E-7/py) = $1.5E+3/py

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

Best Estimate	Upper Bound	Lower Bound
$1.1E+7	$1.2E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

Estimates included directly in the next step.

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

Installation cost of the safety parameter display system (SPDS) in Yankee Rowe as per Issue II.F.3 (Reg. Guide 1.97) is estimated at $1E+6 (*Nuclear News*, 1982). The SPDS is considered a non-IE instrumentation system. It is assumed that classification and upgrading all remaining non-IE systems will represent a similar cost ($1E+6/plant), divided evenly between equipment costs and manpower costs for operating plants. On planned plants, only the additional equipment costs ($5E+5/plant) are assumed. Additional manpower for equipment acquisition or installation should not be required. Therefore,

I = $1E+6/operating plant

I = $5E+5/planned plant

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

$1.03E+8

TABLE 3. (contd)

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

The improved reliability of non-IE instrumentation, control, and electrical systems is assumed to reduce transients other than loss of offsite power from 7/py to 6.58/py for PWRs and from 7/py to 6.72/py for BWRs. Assuming one day of power generation lost per transient, this reduces unscheduled outages by 0.42 days/py for PWRs and by 0.28 days/py for BWRs.

No other unique operation and maintenance requirements are assumed.

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_O):

I_O(PWR) = (\$3.0E+5/day)(-0.42 days/py) = -\$1.26E+5/py

I_O(BWR) = (\$3.0E+5/day)(-0.28 days/py) = -\$8.4E+4/py

(Negative signs indicate cost savings.)

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

-\$4.28E+8

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
-\$3.3E+8	-\$1.0E+8	-\$5.5E+8

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

The IEEE-827 Trial Use Guide on this issue has been released. Resolution of public comments and publishing of the final version is assumed to require 0.4 man-yr. This compares to FY80 and FY81 estimates of 0.4 man-yr and 1.0 man-yr, respectively, from NUREG-0660 (NRC 1980).

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

C_D = (\$1.0E+5/man-yr)(0.4 man-yr) = \$4E+4

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

NRC staff time required for approval of equipment classification and implementation of equipment upgrades could be signficant. It is assumed that this will require 1 man-yr/plant.

TABLE 3. (contd)

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

$$C = (1 \text{ man-yr/plant})(\$1.0E+5/\text{man-yr}) = \$1.0E+5/\text{plant}$$

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

$$N\dot{C} = \$1.34E+7$$

18-20. Steps Related to NRC Cost for Review of SIR Operation and Maintenance (C_O):

No additional operation and maintenance above that currently required is anticipated. Thus, $C_O = 0$.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$1.3E+7	\$2.0E+7	\$6.7E+6

REFERENCES

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

IEEE. 1980. Standard Criteria for Safety Systems for Nuclear Power Generating Stations. IEEE-603, Institute of Electrical and Electronic Engineers, Inc., 345 East 47th Street, New York 10017.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: II.J.3.1/II.J.3.2, Organization and Staffing to Oversee Design and Construction, Issue Regulatory Guide

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This TMI action item seeks to improve the qualification of licensees for operating nuclear power plants by requiring greater oversight of design, construction, and modification activities.

AFFECTED PLANTS	BWR:	Operating = 0	Planned = 10
	PWR:	Operating = 0	Planned = 25

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 3.4E+3

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	-1.2E+4
Total of Above =	-1.2E+4
Accident-Avoidance =	21

COST RESULTS ($$10^6$)

INDUSTRY COSTS:

SIR Implementation =	64
SIR Operation/Maintenance =	4.0
Total of Above =	68
Accident-Avoidance =	1.8

NRC COSTS:

SIR Development =	0.15
SIR Implementation Support =	0.53
SIR Operation/Maintenance Review =	0.40
Total of Above =	1.1

ORGANIZATION AND STAFFING TO OVERSEE DESIGN AND CONSTRUCTION,

ISSUE REGULATORY GUIDE

ISSUE II.J.3.1/II.J.3.2

1.0 SAFETY ISSUE DESCRIPTION

This safety issue as described in NUREG/0660 (NRC 1980) calls for the Office of Nuclear Reactor Regulation (NRR) to develop criteria requiring license applicants and licensees to improve the oversight of design, construction, and modification activities. These criteria will be developed as an inherent part of those criteria planned under item I.B.1.1, Management for Operations. These criteria are to be set considering the results of studies by the Nuclear Safety Analysis Center (NSAC) and Institute for Nuclear Power Operations (INPO). The sequences and timing for development of the criteria are documented in item I.B.1.1., parts (1) through (5). A new Regulatory Guide will be prepared and issued by the Office of Standards Development (SD) that codifies the criteria relating to design and construction.

Under this item, licensees would submit a description of the organization, training, and staffing it proposes to meet the criteria. The technical resources needed by the utility to oversee the design and construction of the plant (including modifications to operating plants) need to be enumerated. Another consideration is the degree of management and technical control to be exercised by the utility during design and construction. The licensee would restructure its organization to assure that the decision-making process is integrated during design, construction, and modification phases and to assure that management is aware of and involved in these activities. The licensee would supplement its staff to provide adequate technical and management resources to oversee design, construction, and modifications.

To assess this safety issue, a number of engineers at the Pacific Northwest Laboratory (PNL) were consulted. These engineers have expertise in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas.

The engineers at PNL determined that the resolution of this safety issue was for the utility to place junior engineers into positions where they would be involved with design and construction from the beginning. The utility engineers would serve as a liaison or interface between the utility and the construction firm(s). They would also assist in designing control systems and write operating procedures. To be effective, they should have authority that has been carefully delineated. Improved management is the major benefit to be accrued from the implementation of this Safety Issue Resolution (SIR) when the utility engineers are placed in their supervisory roles after plant construction. The integration of design and construction with operation via these engineers/supervisors should lead to a reduction in risk from improved management.

It is recognized that some utilities have instituted similar programs to the one recommended. This would lead to quick approval for these utility programs. Significant effort and expense may be required to meet the criteria for those utilities who do not have such a program.

2.0 SAFETY ISSUE RISK AND DOSE

The analyses of public risk reduction and occupational dose are described in the following two sections. The latter term deals primarily with maintenance personnel who work in radiation zones. However, some reduction in routine occupational exposure can be expected for other operations personnel.

PUBLIC RISK REDUCTION

Improvement of operator and maintenance personnel performance is the major result of this SIR. For approximately 25 percent of the nuclear plants under construction after 1983, this issue will have no effect because it is already being implemented and will require no discernible improvement in operator or maintenance personnel performance. Other utilities will see a varying degree of improvement. Development of this SIR was assumed to require one year. For this reason, only plants under construction after 1983 were considered. Those affected would be 10 BWRs with a total of 37 years of construction left and 25 PWRs with a total of 90 years of construction remaining.

The effect on human error due to an improvement in management's knowledge of the power plant was arrived at in the following way. Increased management involvement during design and construction increases management's knowledge of the plant by about 50 percent. Management's effectiveness is composed of such factors as knowledge, aptitude, background, motivation, social affinities, etc. It was assumed that knowledge was approximately 50 percent responsible for effective management. Management ineffectiveness was found to cause about 30 percent of human errors in PWRs and BWRs (Potash 1980). A reduction of 7.5 percent in human errors (50% x 50% x 30%) was assumed. Although these errors are usually maintenance related, some are operator induced. This 7.5 percent is an average over both error types.

It was assumed that after five years the management's knowledge level would be the same as in existing plants after the same length of time. In other words, the operation of this SIR results in risk reduction for the first five years of operation, on average. After five years, the risk level will be the same as now exists in similar nuclear power plants. The implementation of this SIR speeds up management's learning curve. The learning curve is shifted for five years because of management's involvement during construction and design phases. Major modifications that occur in the first five years will also involve the same management personnel who were involved in the design and construction phases.

The industry implementation phase of the SIR takes place during the average remaining construction period of the plant. For the 10 BWRs being examined this average time remaining (after 1983) is 3.7 years and for the 25 PWRs it is 3.6 years, as stated by current schedules obtained from Nuclear News, August 1982.

Table 1 is the work sheet for public risk reduction.

OCCUPATIONAL DOSE

The PNL engineers felt that the major effect is expected to be the reduction of dose to maintenance personnel. The exposure that operators receive in routine duties is also expected to be reduced somewhat. Increased management knowledge of plant layout and operation in the first five years of plant operation is assumed to decrease maintenance personnel exposure in radiation zones. The overall effect may range from a reduction of 5 to 30 percent in the total estimated dose. An average value of 17.5 percent is used in the calculations which follow. It is estimated that 300 to 500 man-rem of occupational exposure occur annually at a typical facility. If we assume 400 man-rem as a best estimate, the 17.5 percent reduction results in an occupational dose reduction of 70 man-rem per plant year for the first five years of operation.

For this issue, there is no implementation dose since it applies only to planned plants. The occupational dose analysis is summarized in Table 2.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Organization and Staffing to Oversee Design and Construction, Issue Regulatory Guide (II.J.3.1/II.J.3.2)

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

   75% of reactors under construction after 1983 are assumed to be affected:

   PWR (N = 25, $\bar{T}$ = 5 yr), BWR (N = 10, $\bar{T}$ = 5 yr)

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf 1 - representative BWR

TABLE 1. (contd)

3. Plants Selected for Analysis (contd):

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

4. Parameters Affected by SIR:

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

5. Base-Case Values for Affected Parameters:

Original values from Appendix A are assumed (Andrews et al. 1983).

6. Affected Accident Sequences and Base-Case Frequencies:

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

7. Affected Release Categories and Base-Case Frequencies:

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

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

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

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

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

10. Adjusted-Case Values for Affected Parameters:

B = C	=	0.0031
D = E	=	0.022
CH1	= CH2 = CH3 = CH4 =	0.0048
CONST1	=	1.7E-4

(a) See Attachment 1.

TABLE 1. (contd)

10. Adjusted-Case, Values for Affected Parameters (contd):

CONST2	=		5.4E-4
A1 = C1	=		0.0096
B1	=		0.035
(B_3)	=		5.0E-4
K	=		2.4E-5
G1	=		0.013
HHMAN	= HPMAN1 =		0.09
HPMAN	=		0.0139
LPISCM	=		0.0028
HPRSCM	= WXCM =		0.0028
RCSRBCM	=		3.0E-5
D•E	=		4.5E-4
W•X	=		8.5E-5
B•W	= C•X =		2.5E-5
D•X	= E•W =		2.0E-4
B•D	= E•C =		5.7E-5

11. Affected Accident Sequences and Adjusted-Case Frequencies:

$$T_2MLU - \begin{cases} \gamma \text{ (PWR-3)} = 4.4E-7/py \\ \beta \text{ (PWR-5)} = 6.5E-9/py \\ \varepsilon \text{ (PWR-7)} = 4.4E-7/py \end{cases}$$

$$T_1MLU - \begin{cases} \gamma \text{ (PWR-3)} = 7.9E-7/py \\ \beta \text{ (PWR-5)} = 1.1E-8/py \\ \varepsilon \text{ (PWR-7)} = 7.9E-7/py \end{cases}$$

$$T_1(B_3)MLU - \begin{cases} \gamma \text{ (PWR-3)} = 9.6E-7/py \\ \beta \text{ (PWR-5)} = 1.4E-8/py \\ \varepsilon \text{ (PWR-7)} = 9.6E-7/py \end{cases}$$

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

T_2MQH -	γ (PWR-3)	= 5.3E-6/py
	β (PWR-5)	= 7.7E-8/py
	ε (PWR-7)	= 5.3E-6/py
S_3H -	γ (PWR-3)	= 4.7E-6/py
	β (PWR-5)	= 6.8E-8/py
	ε (PWR-7)	= 4.7E-6/py
S_1D -	α (PWR-1)	= 6.4E-8/py
	γ (PWR-3)	= 1.3E-6/py
	β (PWR-5)	= 4.6E-8/py
	ε (PWR-7)	= 5.1E-6/py
T_2MQFH -	γ (PWR-2)	= 2.3E-6/py
	β (PWR-4)	= 3.4E-8/py
	ε (PWR-6)	= 2.3E-6/py
S_3FH -	γ (PWR-2)	= 2.0E-6/py
	β (PWR-4)	= 2.9E-8/py
	ε (PWR-6)	= 2.0E-6/py
S_2FH -	α (PWR-1)	= 1.2E-8/py
	β (PWR-4)	= 8.6E-9/py
	ε (PWR-6)	= 9.4E-7/py
T_2MLUO -	γ (PWR-3)	= 3.8E-6/py
	β (PWR-5)	= 5.5E-8/py
	ε (PWR-7)	= 3.8E-6/py
T_2KMU -	γ (PWR-3)	= 3.2E-6/py
	β (PWR-5)	= 4.7E-8/py
	ε (PWR-7)	= 3.2E-6/py

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies (contd):

$$S_2D - \begin{cases} \alpha\ (PWR\text{-}1) & = 1.9E\text{-}8/py \\ \gamma\ (PWR\text{-}3) & = 3.7E\text{-}7/py \\ \beta\ (PWR\text{-}5) & = 1.4E\text{-}8/py \\ \varepsilon\ (PWR\text{-}7) & = 1.5E\text{-}6/py \end{cases}$$

$$S_3D - \begin{cases} \gamma\ (PWR\text{-}3) & = 6.9E\text{-}7/py \\ \beta\ (PWR\text{-}5) & = 1.0E\text{-}8/py \\ \varepsilon\ (PWR\text{-}7) & = 6.9E\text{-}7/py \end{cases}$$

$$T_1MLUO - \begin{cases} \gamma\ (PWR\text{-}3) & = 2.5E\text{-}6/py \\ \beta\ (PWR\text{-}5) & = 3.7E\text{-}8/py \\ \varepsilon\ (PWR\text{-}7) & = 2.5E\text{-}6/py \end{cases}$$

$$T_3MLUO - \begin{cases} \gamma\ (PWR\text{-}3) & = 5.0E\text{-}7/py \\ \beta\ (PWR\text{-}5) & = 7.3E\text{-}9/py \\ \varepsilon\ (PWR\text{-}7) & = 5.0E\text{-}7/py \end{cases}$$

$$T_2MQD - \begin{cases} \gamma\ (PWR\text{-}3) & = 7.4E\text{-}7/py \\ \beta\ (PWR\text{-}5) & = 1.1E\text{-}8/py \\ \varepsilon\ (PWR\text{-}7) & = 7.4E\text{-}7/py \end{cases}$$

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

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 1.0E-7/py
PWR-2 = 5.6E-6/py
PWR-3 = 2.6E-5/py
PWR-4 = 9.0E-8/py
PWR-5 = 4.2E-7/py
PWR-6 = 6.8E-6/py
PWR-7 = 3.2E-5/py

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

TABLE 1. (contd)

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

$\bar{F}^* = 7.1E-5/py$

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

$W^*_{PWR} = 170$ man-rem/py

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

$(\Delta\bar{F})_{PWR} = 7.0E-6/py$ $(\Delta\bar{F})_{BWR} = 3.2E-6/py$[a]

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

$(\Delta W)_{PWR} = 18$ man-rem/py $(\Delta W)_{BWR} = 22$ man-rem/py[a]

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
3.4E+3	1.0E+6	0

<u>ATTACHMENT 1</u>

The RSSMAP studies for Oconee 3 and Grand Gulf 1 give total core-melt frequencies ($\bar{F}_o$) of 8.2E-5/ry and 3.7E-5/ry, respectively, for these plants. Using the original release category frequencies and the public dose factors (Appendix D of Andrews et al. 1983) one obtains total public risks (W_o) of 207 man-rem/ry and 250 man-rem/ry, respectively, for Oconee and Grand Gulf. For the purposes of scaling the base-case, affected core-melt frequency ($\bar{F}$) and public risk (W), and the reductions in the core-melt frequency ($\Delta\bar{F}$) and public risk (ΔW) from Oconee to Grand Gulf, the following are assumed:

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

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

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

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

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

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

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

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Organization and Staffing to Oversee Design and Construction, Issue Regulatory Guide (II.J.3.1/II.J.3.2)

2. Affected Plants (N):

   75% of plants under construction after 1983--10 BWRs and 25 PWRs.

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

   Only the first 5 years after startup will be affected by this issue.

   ($\bar{T}$ = 5 yr for both PWRs and BWRs)

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

   PWR = (19,9D0 man-rem)(7.0E-6/py = 1.4E-1 man-rem/py

   BWR = (19,900 man-rem)(3.2E-6/py = 6.4E-2 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
21	1.4E+3	0

6-8. Steps Related to Occupational Dose from Implementation of SIR:

   These steps are not applicable since the issue resolution is for planned plants only.

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

   Estimate not needed for subsequent steps.

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

    -70 man-rem/py for both BWRs and PWRs

    (negative sign indicates reduction)

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

    -1.2E+4 man-rem (reduction)

TABLE 2. (contd)

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
-1.2E+4 (reduction)	4.0E+3	3.6E+4

3.0 SAFETY ISSUE COSTS

The PNL engineers also estimated the costs associated with the resolution of this safety issue. After reviewing a typical nuclear plant organizational chart (Allenspach and Crocker 1980), it was decided that 10 management positions should be filled with personnel who have gone through the design and construction liaison program. It was estimated that the 10 managers/engineers would spend approximately one-half of the time during the construction phase in this program. Thus, to implement this program, five persons per plant are needed during the remaining 3.7 years of BWR construction and 3.6 years of PWR construction.

The cost to industry for operation/maintenance of the SIR is much less than for the design/construction liaison phase because only during major modifications would time be spent in this program. As mentioned previously, industry costs of operation/maintenance are estimated to be incurred only during the first five years of plant operation; it was estimated that one week per year for each of the 10 full-time industry personnel would be required for modification interfacing during this period of time.

The NRC costs of development for resolution of the safety issue and for issuance of the regulatory guide were taken from the TMI Action Plan, NUREG-0660 (NRC 1980). The plan called for 0.5 man-years of NRC effort for resolution development, which is equivalent to 22 man-weeks, and 0.9 man-years, which is equivalent to 40 man-weeks, and $5000 for issuance of the regulatory guide. The NRC labor for review of industry implementation of the SIR is estimated to be 4 man-weeks during the first year of construction and 1 man-week/year for the remaining construction period. Using a 3.7-year and a 3.6-year average remaining construction period for BWRs and PWRs, respectively, the implementation review labor estimates total 6.7 man-weeks/plant (BWR) and 6.6 man-weeks/plant (PWR). For both PWRs and BWRs, it is assumed that 1 person-weeks/plant-year of NRC labor is required for review over the five years of SIR operation.

Table 3 summarizes the analysis for industry and NRC costs.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Organization and Staffing to Oversee Design and Construction, Issue Regulatory Guide (II.J.3.1/II.J.3.2)

2. Affected Plants (N):

   75% of plants under construction after 1983--10 BWRs and 25 PWRs.

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

   Only the first 5 years after startup. ($\bar{T}$ = 5 yr, for both PWRs and BWRs.)

Industry Costs (Steps 4 through 12)

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

   PWR = ($1.65E+9(7.0E-6/py) = $1.2E+4/py

   BWR = ($1.65E+9)(3.2E-6/py = $5.3E+3/py

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

Best Estimate	Upper Bound	Lower Bound
$7.8E+6	$1.1E+8	0

6. Per-Plant Industry Resources for SIR Implementation:

   Labor: PWR = (10 men/plant)(1/2 time)(44 wk/yr)(3.6 yr)
   = 792 man-wk/plant

   BWR = (10 men/plant)(1/2 time)(44 wk/yr)(3.7 yr)
   = 814 man-wk/plant

   No equipment nonreplacement power is required to implement this resolution.

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

   PWR = (792 man-wk/plant)($2270/man-wk) = $1.80E+6/plant

   BWR = (814 man-wk/plant)($2270/man-wk) = $1.85E+6/plant

TABLE 3. (contd)

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

$6.35E+7

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

(10 men/plant)(1 wk/yr) = 10 man-wk/py

(Same for both PWRs and BWRs.)

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

(10 man-wk/py)($2270/man-wk) = $2.27E+4/py

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

$4.0E+6

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$6.8E+7	$9.9E+7	$3.6E+7

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

It is assumed that 0.5 man-yr are required to develop this SIR and 0.9 man-yr and $5000 to issue the regulatory guide.

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

(1.4 man-yr)($1.0E+5/man-yr) + $5.0E+3 = $1.45E+5

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

PWR = 6.6 man-wk/plant

BWR = 6.7 man-wk/plant

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

PWR = (6.6 man-wk/plant)($2270/man-wk) = $1.50E+4/plant

BWR = (6.7 man-wk/plant)($2270/man-wk) = $1.52E+4/plant

TABLE 3. (contd)

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

$5.27E+5

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

1 man-wk/plant-yr for BWRs and PWRs.

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

(1 man-wk/py)($2270/man-wk) = $2.27E+3/py

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

$3.97E+5

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$1.1E+6	$1.4E+6	$7.3E+5

REFERENCES

Allenspach, F. W. and L. P. Crocker. 1980. Guidelines for Utility Management Structure and Technical Resources. NUREG-0731, U.S. Nuclear Regulatory Commission, Washington, D.C.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: II.J.4.1, Revise Deficiency Reporting Requirements

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

The objective of this safety issue is to clarify deficiency reporting requirements and to obtain uniform reporting, including earlier identification and correction of problems. The proposed resolution is to revise event-reporting requirements of 10 CFR 50.55(e) and 10 CFR 21.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 850

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	5.1

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:

SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	0.42

NRC COSTS:

SIR Development =	0.20
SIR Implementation Support =	0.71
SIR Operation/Maintenance Review =	0
Total of Above =	0.91

REVISE DEFICIENCY REPORTING REQUIREMENTS

ISSUE II.J.4.1

1.0 SAFETY ISSUE DESCRIPTION

The objective of this safety issue resolution (SIR) is to clarify deficiency reporting requirements and to obtain uniform reporting, including earlier identification and correction of problems. Clarification of deficiency reporting requirements should provide increased information on component failures that affect safety so that more prompt and effective corrective action can be taken. This information will also be used as input to an augmented role of the NRC's vendor and construction inspection programs.

NRC will improve, as necessary, the event-reporting requirements [10 CFR Part 50.55(e) for holders of construction permits and Part 21] to assure that all reportable items are reported promptly and that information submitted is complete. Improvements will be implemented by rule changes as appropriate and coordinated with those made under Task I.E.6, "Analysis and Dissemination of Operating Experience--Reporting Requirements." The clarified reporting requirements will provide for more prompt and effective action related to safety events.

This issue affects all 134 PWRs and BWRs, both operating and planned.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose are estimated in this section. Results are summarized in Tables 1 and 2, respectively.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Revise Deficiency Reporting Requirements (II.J.4.1)

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

   All plants

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

TABLE 1. (contd)

3. Plants Selected for Analysis:

Oconee 3 - representative PWR

Grand Gulf 1 - representative BWR

4-7. Steps Related to Affected Parameters, Accident Sequences, Release Categories, and Their Base-Case Values:

Estimates of the reduction in core-melt frequency and public risk due to issue resolution are calculated directly from the base-case values. Thus, these steps (and Steps 10-14) are omitted.

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

PWR = 8.2E-5/py (original Oconee value)

BWR = 3.7E-5/py (original Grand Gulf value)

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

PWR = 2.1E+2 man-rem/py (original Oconee value)

BWR = 2.5E+2 man-rem/py (original Grand Gulf value)

10-14. Steps Related to Adjusted-Case Values of Affected Parameters, Accident Sequences, Release Categories, Core-Melt Frequency and Public Risk:

These steps are omitted (see explanation, Steps 4-7).

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

This SIR can impact safety through the feedback of information gathered under improved deficiency reporting requirements. However, such an effect would be indirect at best, with only a minimal reduction in core-melt frequency. Thus, resolution of this issue is assumed to decrease the base-case, core-melt frequency by only 0.1%. This reduction would result from improved reliability due to corrected deficiencies that were promptly recognized.

PWR: (8.2E-5/py)(0.001) = 8.2E-8/py

BWR: (3.73-5/py)(0.001) = 3.7E-8/py

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

As for the core-melt frequency, the public risk will likewise be reduced by 0.17.

PWR: (2.1E+2 man-rem/py)(0.001) = 0.21 man-rem/py

BWR: (2.5E+2 man-rem/py)(0.001) = 0.25 man-rem/py

TABLE 1. (contd)

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
8.5E+2	2.5E+7	0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Revise Deficiency Reporting Requirements (II.J.4.1)

2. Affected Plants (N):

All plants

	N
PWRs	90
BWRs	44
All	134

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

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

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

PWR: (19,900 man-rem)(8.2E-8/py) = 1.6E-3 man-rem/py

BWR: (19.900 man-rem)(3.7E-8/py) = 7.4E-4 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
5.1	3.1E+4	0

TABLE 2. (contd)

6-12. Steps Related to Occupational Dose Increase from SIR Implementation and Operation/Maintenance, and Total Dose Increase:

The SIR is assumed to involve only procedural changes that would not result in increased occupational dose. Thus, $D = D_o = G = 0$.

3.0 SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. Results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Revise Deficiency Reporting Requirements (II.J.4.1)

2. Affected Plants (N):

All plants

		N			N
PWRs:	Operating	47	BWRs:	Operating	24
	Planned	43		Planned	20
		90			44

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

		$\bar{T}$ (yr)			$\bar{T}$ (yr)
PWRs:	Operating	27.7	BWRs:	Operating	25.2
	Planned	30		Planned	30
	All	28.8		All	27.4

Industry Costs (Steps 4 through 12)

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

PWR: (\$1.65E+9)(8.2E-8/py) = \$1.4E+2/py

BWR: (\$1.65E+9)(3.7E-8/py) = \$6.1E+1/py

TABLE 3. (contd)

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

Best Estimate	Upper Bound	Lower Bound
\$4.2E+5	\$2.5E+9	0

6-12. Steps Related to Industry Costs for SIR Implementation and Operation/Maintenance, and Total Industry Cost:

The industry cost of this SIR is believed to be near zero. Savings may be possible due to reduced redundancy in reporting. Cost increases may occur, due to time constraints and more detailed requirements. For this analysis, a cost of zero was used; i.e., $I = I_0 = S_I = 0$.

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

It is assumed that two man-years are required to finalize all regulation revisions.

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

C_D = (2 man-yr)(\$1.0E+5/man-yr) = \$2.0E+5

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

Estimates are included directly in next step.

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

For the purposes of this study, it is assumed that \$1.0E+4/operating plant is required to implement the mechanics of the revised reporting requirements.

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

NC = (71 operating plants)(\$1.0E+4/plant) = \$7.1E+5

18-20. Steps Related to NRC Cost for Review of SIR Operation and Maintenance:

For the purposes of this study, it is assumed that no additional NRC cost is incurred related to the revised deficiency reporting requirements ($C_0 = 0$). In fact, it is possible that NRC costs could decrease as a result of the elimination of multiple reporting requirements for the same event.

TABLE 3. (contd)

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$9.1E+5	\$1.3E+6	\$5.4E+5

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: III.A.1.3, Maintain Supplies of Thyroid Blocking Agent (Potassium Iodide)

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Following the accident at TMI, there was a resurgence of interest in the use of potassium iodide (KI) as an emergency protective measure. The proposed resolution to this issue is to maintain supplies of KI for onsite individuals.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 0

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	2100

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:

SIR Implementation =	0.32
SIR Operation =	2.9
Total of Above =	3.2
Accident Avoidance =	0

NRC COSTS:

SIR Development =	0
SIR Implementation Support =	0
SIR Operation/Maintenance Review =	1.7
Total of Above =	1.7

MAINTAIN SUPPLIES OF THYROID BLOCKING AGENT (POTASSIUM IODIDE)

ISSUE III.A.1.3.

1.0 SAFETY ISSUE DESCRIPTION

Following the accident at Three Mile Island, there was a resurgence of interest in the use of thyroid blocking as an emergency protective measure for nuclear reactor accidents. Recent studies since then have examined the following aspects:

1) the effectiveness of potasium iodide (KI) as a thyroid blocking agent in potential reactor accident situations

2) the projected radiation dose to the thyroid gland for which a benefit/risk decision would be favorable

3) the area within which KI should be distributed

4) the relative effectiveness of KI compared to other protective measures

5) whether stockpiling and distribution of KI is feasible and cost effective.

These issues are controversial because many decisions concerning the use of KI as a thyroid blocking agent in a radiation emergency are judgmental and based on analyses of a limited quantity of available information.

It is possible that a nuclear power reactor accident could release large quantities of radionuclides into the environment, including isotopes of radioiodine. Should a release of radioactive iodine occur, KI could help prevent radiation injury to the thyroid gland by saturating the gland with non-radioiodine. This blocks the thyroid from taking up radioiodine and can help prevent the development of radioiodine-induced nodules or cancer. KI at recommended doses could block about 90 percent of radioiodine absorption if the first dose is taken shortly before, or at, the time of exposure to radioiodine (FDA 1982). The Food and Drug Administration (FDA) has concluded that risks from short-term use of KI are outweighed by risks of radioiodine-induced thyroid nodules or cancer for persons who are likely to receive a projected radiation dose of 25 rem or greater to the thyroid gland (FDA 1982).

The KI safety issue involves deciding whether or not stockpiling and distribution of KI are feasible and cost effective. That feasibility and cost-effectiveness may depend upon each specific plant. Populations within a 10-mile-radius emergency planning zone (EPZ) around operating plants range from about 25,000 people at Indian Point to 2,000 in other places (ACRS 1982). In-place sheltering and KI distribution may be a more effective protective measure than trying to evacuate large populations. In a much less populated EPZ, completion of evacuation might be possible in a few hours, and the use of KI would be unnecessary as an ancillary protective action. However, distribution of KI to 25,000 people would be much more costly and complicated than distribution to 2,000 people. A KI distribution plan would include costs for stockpiling, distribution, and replacement. Costs may be included for informing the public of the nature of radiation hazard, the potential benefits and adverse effects of KI, and its use. Working costs would be needed for planning, monitoring, and execution of all distribution phases. If predistribution is not used, plans would be needed for rapid distribution of KI to potentially affected population groups in order to optimize KI's effectiveness. There will be little cost effectiveness if KI cannot be distributed in time to prevent thyroidal blocking of radioiodines.

Based upon a review of the documents on this subject and the present NRC position on KI, the proposed safety issue resolution (SIR) is as follows: Maintain supplies of the thyroid blocking agent, potassium iodide, as a protective measure only for energency workers and other individuals onsite during an emergency. Onsite individuals are plant staff, visitors, and others assisting in accident mitigation.

A large stockpile of KI would not be required for onsite workers, and it could be easily controlled. If necessary, KI cound be distributed at the time of an accident. The individuals involved are most likely to be exposed to an airborne release. Medical histories of onsite persons could be readily established, and people allergic to KI could be limited to areas of no plume exposure.

2.0 SAFETY ISSUE RISK AND DOSE

No public risk reduction will result from the proposed SIR. However, there will be a reduction in occupational dose due to accident avoidance as a result of decreasing the post-accident exposure to onsite personnel. This reduction is quantified in Attachment 1 to Table 1, the Occupational Dose Work Sheet.

TABLE 1. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Maintain Supplies of Thyroid Blocking Agent (Potassium Iodide) (III.A.1.3)

2. Affected Plants (N):

		N			N
PWRs:	Operating	47	BWRs:	Operating	24
	Planned	43		Planned	20
	Total	90			40

TABLE 1. (contd)

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

		$\bar{T}$ (yr)			$\bar{T}$ (yr)
PWRs:	Operating	27.7	BWRs:	Operating	25.2
	Planned	30.0		Planned	30.0
	All	28.8		All	27.4

4. Per-Plant Occupational Dose Reduction due to Accident Avoidance, $\Delta(\bar{FD}_R)$:

Estimated directly in next step

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2100[a]	not estimated	not estimated

6-12. Steps Related to Occupational Dose Increase from SIR Implementation and Operation/Maintenance, and Total Dose Increase:

No increase in occupational dose is foreseen for SIR implementation, operation, or maintenance. Thus, $D = D_0 = G = 0$.

(a) See Attachment 1.

ATTACHMENT 1

In the Guidelines (Andrews et al. 1983) used in conjunction with this safety issue assessment, all exposure pathways except injestion were included for core-melt sequences in determining the whole-body dose consequence factors (man-rem) for core-melt release categories (PWR 1-7 and BWR 1-4). For people exposed to a radioactive plume, the use of KI would result in a small reduction in the whole-body doses used in this assessment but can have a significant reduction in the dose received by the thyroid. The risk reduction used in this assessment is in the form of man-rem. A more appropriate way to assess the risk reduction of KI is in health effects, e.g., thyroid nodules prevented. However, for consistency in comparison with other safety issue assessments, a conversion to whole-body dose was performed.

To help determine risk reduction, data from NUREG/CR-1433 (Aldrich et al. 1980) are used. Table 5 of that document lists fractional components of thyroid dose to an exposed individual. These consist of doses from ground and cloud exposure and inhaled radioiodines and non-radioiodines. It is assumed in this analysis that 88 percent of the dose to the thyroid in release categories PWR 1-5 and BWR 1-3 is from inhaled radioiodines. A fraction of 81 percent was used for PWR 6-7 and BWR 4 categories. Using these assumptions and data from Table 3 (one mile downwind) of NUREG/CR-1433, total dose to an individual thyroid for PWR 1-5 and BWR 1-3 categories would be 11,400 man-rem. Doses from PWR 6-7 and BWR 4 would be 20 man-rem.

An onsite population of 140 persons is assumed. This results in plant population thyroid doses of 1.6E+6 man-rem and 2800 man-rem for the two accident types.

The core-melt frequencies for Oconee and Grand Gulf as originally assessed (see Appendices A and B of Andrews et al. 1983) were used to calculate the base-case expected thyroid dose to onsite staff. Results are as follows:

Release Category	Thyroid Dose (man-rem/py)
PWR 1	0.18
PWR 2	16.0
PWR 3	46.4
PWR 4	0.16
PWR 5	0.74
PWR 6	0.02
PWR 7	0.10
TOTAL	63.60

ATTACHMENT 1. (contd)

Release Category	Thyroid Dose (man-rem/py)
BWR 1	0.18
BWR 2	54.4
BWR 3	2.24
BWR 4	0.004
TOTAL	56.82

Total industry base-case thyroid dose is calculated by multiplying the above doses by the remaining number of plant-years. The result is 2.3E+5 man-rem to the thyroid.

The thyroid blocking agent, KI, is assumed to be 90 percent effective in reducing this occupational dose. The adjusted-case thyroid dose becomes 2.3E+4 man-rem. Total dose reduction to the thyroid is 2.1E+5 man-rem. This is believed to be an upper limit to thyroid dose reduction resulting from alternative protective actions available to onsite personnel. These include shelter, respiratory protection devices, and evacuation.

Two assumptions were made to convert this result to a basis comparable to other safety issues. First, health effects from thyroid dose are 95 percent curable with no long-term effects. Second, whole-body dose is given five times the weighting of thyroid dose in protective action guides (NRC 1980). Thus, thyroid dose was reduced by a factor of (20)(5) = 100 to give an equivalent whole-body dose comparable to other safety issue analyses. This results in a total occupational dose reduction of 2100 man-rem due to use of KI blocking agent by onsite personnel.

3.0 SAFETY ISSUE COSTS

Until recently, the Federal Emergency Management Agency (FEMA) was considering the possibility of buying a national stockpile of KI. However, FEMA has decided no to maintain a national stockpile. It is now the responsibility of the Health Department of each state to initiate a KI program if they wish to do so (Kremm 1982). With no financial support from Federal agencies, costs of a KI program will be incurred by nuclear power plant utilities and/or the state. For purposes of this analysis, it is assumed that each plant will finance its own stockpile.

Costs for this SIR are assumed to include purchasing and shipment of a KI stockpile based on a three-year shelf life. These costs are listed as operation and maintenance expenditures. Monitoring and administrative costs, which may include storage, KI distribution at the time needed, medical screening, and inclusion of KI distribution procedures into plant emergency plans, are assumed to require one man-week of effort every three years. The only NRC labor cost assumed in this analysis is for review of SIR operation and maintenance at one man-day per plant-year.

The number of onsite individuals per plant was partially determined from a report by Oak Ridge Associated University (ORAU 1982). As of March 1981, there were 35,853 onsite positions available in the U.S. for both operating plants and plants under construction. Out of this number, there were 24,600 onsite utility positions. Dividing 24,600 by the 134 plants used in this assessment, which includes both planned and operating plants, gives an average of 184 utility positions per plant. Seventy of these positions were assumed for swing and graveyard shifts, and were subtracted from 184 to yield an average maximum number of 114 utility positions available at an operating plant at any one time (day shift).

In addition to the onsite utility personnel, there are 5 onsite NRC staff members to operate the Technical Service Center (TSC) during an emergency (NRC 1981) and 21 assumed visitors, vendors, etc. Thus, the total number of onsite individuals per plant is assumed in this analysis to be 140. Cost details are shown in Table 2.

TABLE 2. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Maintain Supplies of Thyroid Blocking Agent (Potassium Iodide) (III.A.1.3).

2. Affected Plants (N):

   All 134 plants (90 PWRs and 44 BWRs)

TABLE 2. (contd)

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

	$\bar{T}$ (yr)
90 PWRs	28.8
44 BWRs	27.4
All LWRs	28.3

Industry Costs (Steps 4 through 12)

4-5. Steps Related to Industry Cost Savings Due to Accident Avoidance:

Since there is no change in core-melt frequency as a result of SIR, no cost will be saved as a result of accident avoidance. $\Delta H = 0$.

6. Per-Plant Industry Resources for SIR Implementation:

Labor = 1 man/plant (same for PWRs and BWRs)

Equipment = KI stockpile at 75¢/person/plant

Additional down-time = None

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

$$I = (\$0.75/\text{person})(140\ \text{persons/plant}) + (1\ \text{man-wk/plant})(\$2770/\text{man/wk})$$
$$= \$2375/\text{plant}$$

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

$$NI = (134\ \text{plants})(\$2375/\text{plant}) = \$3.18E+5$$

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

1 man-wk/plant every 3 years, or 0.33 man-wk/py

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

$$I_0 = (0.33\ \text{man-wk/py})(\$2270/\text{man-wk}) = \$7.57E+2/\text{py}$$

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

$$N\bar{T}I_0 = (134\ \text{plants})(28.3\ \text{yr})(\$7.57E+2/\text{py}) = \$2.87E+6$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$3.2E+6	\$4.6E+6	\$1.7E+6

TABLE 2. (contd)

NRC Costs (Steps 13 through 21)

13-17. Steps Related to NRC Costs for SIR Development and Implementation Support:

No NRC effort is foreseen in connection with SIR development or implementation support; thus, $C_D = C = 0$.

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

1 man-day/py

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

C_O = (1 man-day/py)($2770/man-wk)(1 man-wk/5 man-day) = $454/py

20. Total NRC Cost for Review of SIR Operation and Maintenance ($\bar{NTC}_O$):

$\bar{NTC}_O$ = (134 plants)(28.3 yr)($454/py) = $1.72E+6

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$1.7E+6	$2.6E+6	$8.6E+5

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: III.A.3.4, Nuclear Data Link

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

The NRC plans to respond to a nuclear plant accident via data sent from the plant to the NRC operations center. A voice link cannot convey data as rapidly or reliably as needed. A system is proposed to send the appropriate data to determine plant status and assess potential public health impact in a more efficient and reliable manner.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	1.7E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	D
SIR Operation/Maintenance	0
Total of Above =	0
Accident Avoidance =	100

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	1.5
SIR Operation/Maintenance =	1.7
Total of Above =	3.1
Accident Avoidance =	8.3
NRC COSTS:	
SIR Development =	0.70
SIR Implementation Support =	8.8
SIR Operation/Maintenance Review =	1.7
Total of Above =	11

NUCLEAR DATA LINK

ISSUE III.A.3.4

1.0 SAFETY ISSUE DESCRIPTION

"Nuclear data link" (NDL) is the term given to a system that will remotely assess and transmit data from nuclear power plants to the Nuclear Regulatory Commission's Operations Center (NRCOC). The system was proposed by the NRC staff to augment and improve its incident response capabilities and the response capabilities of the plant licensee. The data will allow the NRC to analyze and evaluate the plant situation in emergency conditions and to develop or evaluate proposed accident-mitigating actions. The NDL is perceived as a major element in the task of upgrading incident response capabilities at nuclear power plant sites, NRC headquarters, NRC regional offices, and other federal, state, and local government agencies.

Over the past years, the NRC has made other improvements to the incident response capabilities which it inherited from the Atomic Energy Commission, the most important being the NRCOC, located in Bethesda, Maryland. However, major limitations in NRCOC capabilities were discovered during the Three Mile Island incident. NRC efforts to obtain data were hampered by busy telephone circuits and a limited and inaccurate transfer of technical data using voice telephone communications. It was recognized that if response agencies were to function as conceived, there was need for improved communications. The initial NRC response to this need was to install dedicated telephone lines between the NRCOC and all operating nuclear power plants. While dedicated phone lines improved the NRC's ability to contact plant personnel by eliminating busy phone circuits, it has become evident that voice communication alone may not be adequate.

The NRC document, Report to Congress on the Acquisition of Reactor Data for the NRC Operations Center, NUREG-0730, indicated a problem with manpower. If readings of several parameters are to be recorded on a regular basis and reported to the NRC, licensees may need a staff of several persons per shift dedicated to that task 24 hours a day. During an incident, assistance of persons with the required plant familiarity is at a premium.

Furthermore, some of the incident-response measures instituted by the NRC are for the specific purpose of reducing crowded conditions in the control room and other critical areas of the plant. Adding a manual means of acquiring plant data may be counterproductive to that purpose. Additionally, the urgent, stressful environment surrounding an incident increases the likelihood of human errors in transcribing parameter values and in possibly reading the wrong parameter.

Finally, the use of a voice-based system would require a further reduction in the number of parameters read and the frequency of their reading because one dedicated phone line would not be sufficient to verbally transfer the volume of data required. Therefore, it has been concluded that a voice-based system may not be adequate to communicate the required data to the NRC Staff.

PROPOSED RESOLUTION

The resolution to this safety issue is to complete the definition of requirements for the NDL and then install and operate it in power plants. Current estimates indicate that parameter data from about 100 (between 50 and 150) sensors at each plant would be adequate to determine plant status and assess potential impacts to public health and safety. These 100 parameters include reactor temperatures and pressures, radiological parameters in-plant and offsite, and meteorological data.

Preliminary work on this safety issue resolution (SIR) has been accomplished. Sandia National Laboratories (SNL) has been contracted as system integrator for developing the concept for data acquisition from licensed facilities and for upgrading the NRCOC.

The Sandia program defined a scope for an NRC NDL. This work was coordinated with the criteria being developed by the Office of Nuclear Reactor Regulation (NRR) for licensee data links in the Technical Support Center (TSC) and the Emergency Operations Facility (EOF) and with various groups in the industry. NRC links with nuclear facilities, methods of transmission, and the display and arrangement of the improved NRCOC were studied. In this initial development, consideration was given to a series of alternate data inputs (i.e., 20-100-500 parameters monitored) and associated problems and implications of availability as follows:

1) From plant computer, is hard-wiring to monitor/sensor necessary; is signal in analog or digital form; what form should output signal be in?

2) What standardization criteria must be developed for interfacing and tie-in with the licensee data links for the TSC and EOF and the industry-operated data centers recommended by the NRC Special Inquiry Group?

The results of SNL's reports played a significant role in the NRC's decision in late April 1981 to install several prototype data links. Following evaluation of the experience with the prototype displays, the Commission will decide how far to go in the development of the automated NDL concept.

Currently, there is a "hold" on the decision to install a prototype NDL. The purpose of this delay is to allow discussion on the ramifications of implementing an NDL, regarding the degree of NRC involvement with a licensee during an emergency situation. Although it is currently funded, those discussions will undoubtedly have to take place before the NDL prototype program is initiated.

2.0 SAFETY ISSUE RISK AND DOSE

For the purposes of this analysis, the safety issue risk and dose are measured in terms of the public risk reduction and the occupational dose. The estimated benefit associated with installing and operating an NDL between nuclear power plants and the NRCOC is described below.

One of the main benefits of this SIR is indirectly related to a reduction in risk. Implementing an NDL system would allow the NRC to perform a simultaneous independent assessment of a nuclear power plant emergency situation. Assurance (a double check) could be given to the licensee's actions, and/or recommendations of protective actions could be made that would mitigate accidents and reduce or prevent radiation exposure. An NDL would give the NRC the opportunity to provide additional expertise during an accident and act as an independent advisor to the licensee and/or involved local authorities.

Circumstances can also be envisioned where implementing an NDL might result in direct risk reduction. An example would be the occurrence of a progressive accident (not a single event, but an evolving degradation of safety systems) while the NRCOC is functional. An operator (licensee) error could occur that would eventually result in radiation exposure to the public. With an assessment of the data received through the NDL, the NRCOC personnel may be able to detect the error in time to recommend protective actions.

Factors contributing to the avoidance of core-melt accidents using the NDL include the likelihoods that it is operating during an accident (assumed 50%), that an operator error occurs (assumed 8%) and that the NRCOC is able to detect the error and recommend corrective action (assumed 50%). This sequence of events was estimated to reduce core-melt accident frequencies by 2%. This figure is believed to be an upper bound, given the reduced level of information available to the NRCOC and the potential for making incorrect recommendations.

The public risk reduction and occupational dose results are summarized in Tables 1 and 2, respectively.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Nuclear Data Link (III.A.3.4)

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

   All plants

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

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf 1 - representative BWR

TABLE 1. (contd)

4-6. Steps Related to Affected Parameters, Accident Sequences, and Their Base-Case Values:

The release category frequencies are assumed to be directly affected. Thus these steps are omitted from the analysis.

7. Affected Release Categories and Base-Case Frequencies:

All release categories are presumed to be affected. Base-case frequencies are taken as original values (Andrews et al. 1983).

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

$\bar{F}_{PWR}$ = 8.15E-5/py $\bar{F}_{BWR}$ = 3.67E-5/py

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

W_{PWR} = 207 man-rem/py W_{BWR} = 250 man-rem/py

10-
11. Steps Related to Adjusted-Case Values of Affected Parameters and Accident Sequences:

Since release category frequencies are assumed to be directly affected, these steps are omitted.

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 1.08E-7/py	BWR-1 = 1.08E-7/py
PWR-2 = 9.80E-6/py	BWR-2 = 3.33E-5/py
PWR-3 = 2.84E-5/py	BWR-3 = 1.37E-6/py
PWR-4 = 9.51E-8/py	BWR-4 = 1.57E-6/py
PWR-5 = 4.51E-7/py	
PWR-6 = 7.15E-6/py	
PWR-7 = 3.43E-5/py	

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

$\bar{F}^*_{PWR}$ = 7.99E-5/py $\bar{F}_{BWR}$ = 3.60E-5/py

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

W_{PWR} = 203 man-rem/py W_{BWR} = 245 man-rem/py

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

$\Delta\bar{F}_{PWR}$ = 1.6E-6/py $\Delta\bar{F}_{BWR}$ = 7.3E-7/py

TABLE 1. (contd)

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

ΔW_{PWR} = 4.1 man-rem/py ΔW_{BWR} = 5.0 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.7E+4	2.5E+7	0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Nuclear Data Link (III.A.3.4)

2. Affected Plants (N):

All plants (90 PWRs and 44 BWRs)

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

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

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

$$(\Delta \bar{F}_R)_{PWR} = (19{,}860 \text{ man-rem})(1.6E\text{-}6/py)$$
$$= .0318 \text{ man-rem/py}$$
$$(\Delta \bar{FD}_R)_{BWR} = (19{,}860 \text{ man-rem})(7.3E\text{-}7/py)$$
$$= .0145 \text{ man-rem/py}$$

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
100	3.1E+4	0

6-12. Steps Related to Occupational Dose Increase for SIR Implementation and Operation/Maintenance, and Total Dose Increase:

No change in occupational dose is assumed to result from the SIR for III.A.3.4. Therefore, $D = D_0 = G = 0$.

3.0 SAFETY ISSUE COSTS

Much of the cost for the NDL SIR will be incurred by the NRC. Currently, $700K is budgeted to sponsor the NDL prototype program. Completion of this program should enable the NDL scope and design to be finalized.

Industry costs will include labor for standardizing the NDL data format between the plant and the NRCOC and for operation and maintenance of the NDL portion of the plant. Equipment, labor, operation and maintenance costs are based upon the assumption that the data acquisition system (DAS), in conjunction with the safety parameter display system (SPDS, which is now required at all plants), is installed and operating before NDL implementation. Furthermore, it is assumed that the DAS will contain the parameter data that will be required by the NDL. This will minimize connections to plant sensors and related equipment and labor costs. The DAS taps the lines that send plant sensor data to the control room (such as temperature and pressure). The DAS then sends this data in an appropriate form to be displayed on the SPDS, which is located in the control room.

It is assumed that the NDL will not require additional hardware equipment at the NRCOC and that the NRCOC will have been improved before NDL implementation. However, software will be needed to transform plant data to the appropriate form (e.g., consistent units) used at the NRCDC. An assumption of one percent per plant-year of NDL equipment costs is used for both industry and NRC-related operation and maintenance costs. Cost details are shown in Table 3. All 134 PWRs and BWRs will experience a cost savings due to accident avoidance. However, several reactors at the same sites share a common DAS and SPDS. Thus, additional implementation, maintenance, and operational costs for each of these reactors will not be needed. There are only 31 BWRs (16 operating and 15 planned) and 60 PWRs (34 operating and 26 planned) that will incur NDL costs.

TABLE 2. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Nuclear Data Link (III.A.3.4)

2. Affected Plants (N):

   All 134 plants (90 PWRs and 44 BWRs) will experience a cost savings due to accident avoidance. With respect to all other costs estimated for this SIR, only the following numbers of plants will be affected:

		N
PWRs:	Operating	34
	Planned	26
BWRs:	Operating	16
	Planned	15
		91

TABLE 2. (contd)

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

		$\bar{T}$ (yr)	
PWRs:	All 90[a]	28.8	
	34 Operating	27.7	avg = 28.7
	26 Planned	30	
PWRs:	All 44[a]	27.4	
	16 Operating	25.2	avg = 27.5
	15 Planned	30	

Industry Costs (Steps 4 through 12)

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

$$(\Delta \bar{F}A)_{PWR} = (\$1.65E+9)(1.6E-6/py) = \$2640/py$$

$$(\Delta \bar{F})_{BWR} = (\$1.65E+9)(7.3E-7/py) = \$1200/py$$

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

Best Estimate	Upper Bound	Lower Bound
\$8.3E+6	\$2.5E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

Labor = 7 man-wk/plant (same for PWRs and BWRs)

Equipment = none (equipment to be provided by NRC--see Steps 15 & 16)

Additional down-time = none

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

I = (7 man-wk/plant)(\$2270/man-wk) = \$1.59E+4/plant

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

NI =(91)(\$1.59E+4) = \$1.45E+6

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

Cost is estimated directly in the next step.

(a) Affected numbers of plants experiencing accident-avoidance cost savings.

TABLE 2. (contd)

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_O):

Operation/maintenance cost is assumed to be 1% of equipment cost. (See Step 16.)

$$I_O = (.01)(\$6.5E+4) = \$650/py$$

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

$$N\bar{T}I_O = [(60)(28.7) + (31)(27.5)](\$650)$$
$$= \$1.67E+6$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$3.1E+6	$4.2E+6	$2.0E+6

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Cost is estimated directly in the next step.

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

$$C_O = \$7.0E+5$$

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

Labor = 14 man-wk/plant

Equipment cost (both in-plant and at NRCOC) is estimated directly in the next step.

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

Labor = (14 man-wk/plant)($2270/man-wk) = $3.18E+4/plant

Equipment = $6.50E+4/plant

C = $9.68E+4/plant

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

$$NC = (91)(\$9.68E+4) = \$8.81E+6$$

TABLE 2. (contd)

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

Cost is estimated directly in the next step.

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

Operation/maintenance cost is assumed to be 1% of equipment cost. (See Step 17.)

$C_O = (.01)(\$6.5E+4) = \$650/py$

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

$N\bar{T}C_O = [(60)(28.7) + (31)(27.5)](\$650)$

21 Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$1.1E+7	\$1.6E+7	\$6.7E+6

REFERENCES

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: III.D.1.4, Radwaste System Design Features to Aid in Accident Recovery and Decontamination

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This safety issue requests a study to investigate what features might be required on all reactor plants to aid in accident recovery and decontamination. The resolution is assumed to be the addition of a number of recommended features, resulting from the study, to all plants.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	NA
OCCUPATIONAL DOSES:	
SIR Implementation =	1600
SIR Operation/Maintenance =	280
Total of Above =	1900
Accident Avoidance =	510

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation	380
SIR Operation/Maintenance =	8.6
Total of Above	380
Accident Avoidance =	12
NRC COSTS:	
SIR Development =	0.057
SIR Implementation Support =	1.2
SIR Operation/Maintenance Review =	1.7
Total of Above =	3.0

RADWASTE SYSTEM DESIGN FEATURES TO AID IN ACCIDENT RECOVERY AND DECONTAMINATION

ISSUE III.D.1.4

1.0 SAFETY ISSUE DESCRIPTION

This safety issue requests a study to investigate what features might be required on all reactor plants to aid in accident recovery and decontamination (NRC 1980). To develop the information for prioritization, the following features were assumed to be recommended from the study:

- piping and connections installed for attaching a portable water demineralization system
- additional spray nozzles in containment directed for wash down of major surfaces
- addition of shielding on stairways inside containment.

It was assumed that these features would be added to all PWRs and BWRs, both operating and under construction.

2.0 SAFETY RISK AND DOSE

PUBLIC RISK REDUCTION

This safety issue resolution (SIR) would have no effect on reducing public risk, as it is only concerned with post-accident cleanup and refurbishment. The features added would not reduce the core-melt frequency or public dose. No Public Risk Reduction Work Sheet is given.

OCCUPATIONAL DOSE

For the occupational dose due to accident avoidance, it is assumed that a 10 percent reduction in occupational dose received during cleanup, repair, and refurbishment could result from implementation of the features described. Analysis results are given in Table 1.

TABLE 1. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Radwaste System Design Features to Aid in Accident Recovery and Decontamination (III.D.1.4)

TABLE 1. (contd)

2. Affected Plants (N):

		N	
PWR:	planned	43	90
	operating	47	
BWRs:	planned	20	44
	operating	24	
Total		134	

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

		$\bar{T}$(yr)	
PWRs:	43 planned	30.0	avg = 28.8
	47 operating	27.7	
BWRs:	20 planned	30.0	avg = 27.4
	24 operating	25.2	
All	134 plants	28.3	

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

Issue resolution is assumed to reduce the occupational radiation dose from cleanup, repair, and refurbishment 10% from 19,900 man-rem to 17,900 man-rem. The original core-melt frequencies for Oconee 3 and Grand Gulf 1 are taken to be representative of a PWR and BWR, respectively.

$\Delta\ (\bar{FD}_R)_{PWR}$ = (8.2E-5/py)[(19,900 - 17,900) man-rem] = 0.163 man-rem/py

$\Delta\ (\bar{FD}_R)_{BWR}$ = (3.7E-5/py)[(19,900 - 17,900) man-rem] = 0.0736 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
510	3.1E+4	0

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

It is assumed that installation of the features will require a total of 300 man-wk/plant of utility time in radiation zones while the reactor is in a shutdown condition. Assuming a 75% utilization factor of manpower in the radiation zones:

(0.75)(300 man-wk/plant)(40 man-hr/man-wk) = 9000 man-hr/plant

(Same for operating PWRs and BWRs)

TABLE 1. (contd)

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

It is assumed that an average field of 2.5 mR/hr exists during shutdown where installation of the features is required.

0 = (9000 man-hr/plant)(0.0025 R/hr) = 23 man-rem/plant

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

ND = (71)(23 man-rem/plant) = 1.63E+3 man-rem

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

It is assumed that one man-wk/yr of utility labor would be required for examination and preventive maintenance of the installed features. Assuming a 75% utilization factor for work in the radiation fields:

(0.75)(1 man-wk/plant-yr)(40 man-hr/man-wk) = 30 man-hr/py

(Same for PWRs and BWRs)

10. Per-Plant Occupational Dose Increase for SIR Operation and Maintenance (0_0):

Assuming an average radiation field of 2.5 mR/hr:

D_0 = (30 man-hr/py)(0.0025 R/hr) = 0.075 man-rem/py

11. Total Occupational Dose Increase for SIR Operation and Maintenance ($N\bar{T}D_0$):

Total $N\bar{T}D_0$ = (134)(28.3 yr)(0.075 man-rem/py) = 2.84E+2 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.9E+3	5.7E+3	6.4E+2

3.0 SAFETY ISSUE COSTS

For the industry cost savings due to accident avoidance, it was assumed that the cost of cleanup, repair and refurbishment after an accident would be reduced by 10 percent because of the added features. Analysis results for both industry and NRC costs are given in Table 2.

TABLE 2. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

    Radwaste System Design Features to Aid in Accident Recovery and Decontamination (III.D.1.4).

2. Affected Plants (N):

		N	
PWRs:	planned	43	90
	operating	47	
BWRs:	planned	20	44
	operating	24	
Total		134	

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

		$\bar{T}$(yr)	
PWRs:	43 planned	30.0	avg = 28.8
	47 operating	27.7	
BWRs:	20 planned	30.0	avg = 27.4
	24 operating	25.2	
All	134 plants	28.3	

Industry Costs (Steps 4 through 12)

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

    Resolution of this issue does not involve a change in core-melt frequency, but it is assumed that the cost of cleanup, repair and refurbishment would be reduced by 10% due to the radwaste safety features added. Adjusting the values given in Appendix E (Andrews et al. 1983) for cleanup, repair and refurbishment yields

    A* = \$335E+6 (Clean-up) + \$95E+6 (Repair/Refurbishment) + \$1172E+6

    (Replacement power) = \$1602E+6

    ΔA = A - A* = \$1650E+6 - \$1602E+6 = \$48E+6

    Again, the original core-melt frequencies for Oconee 3 and Grand Gulf 1 are taken to be representative of a PWR and BWR, respectively:

$$(\bar{F}\Delta A)_{PWR} = (\$48E+6)(8.2E-5/py) = \$3.94E+3/py$$

$$(\bar{F}\Delta A)_{BWR} = (\$48E+6)(3.7E-5/py) = \$1.78E+3/py$$

TABLE 2. (contd)

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

Best Estimate	Upper Bound	Lower Bound
$1.2E+7	$2.5E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

Development of estimates for labor and materials on this issue was based on providing connections for portable demineralization, spray nozzles for wash down and shielding for stairways inside containment. Assumed resources are as follows:

Labor:	Labor, man-wk: Operating Plants	Labor, man-wk: Planned Plants
Piping connections		
Engineering	75	60
Crafts, etc.	300	240
Spray Nozzles		
Engineering	75	60
Crafts, etc.	300	240
Additional Shielding		
Engineering	25	20
Crafts, etc.	100	80
Total	875	700

Equipment:

- Piping and connections for portable demineralization $5E+5/plant
- Additional spray nozzles for wash down $4E+5/plant
- Additional shielding on stairways inside containment. $1E+5/plant

Total $1E+6/plant

TABLE 2. (contd)

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

Planned plants:

Labor = (700 man-wk/plant)($2270/man-wk) =	$1.58E+6/plant
Equipment/materials =	$1.00E+6/plant
	$2.58E+6/plant

Operating plants:

Labor = (875 man-wk/plant)($2270/man-wk) =	$1.99E+6/plant
Equipment/materials =	$1.00E+6/plant
	$2.99E+6/plant

(Same for PWRs and BWRs)

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

NI = (63)($2,58E+6/plant) + (71)($2.99E+6/plant) = $3.75E+8

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

It is assumed that 1 man-wk/py is needed to maintain any new equipment. This estimate applies to both PWRs and BWRs.

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_o):

I_o = (1 man-wk/py)($2270/man-wk) = $2.27E+3/py

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

$N\bar{T}I_o$ = (134)(28.3 yr)($2270/py) = $8.6E+6

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$3.8E+8	$5.7E+8	$2.0E+8

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Generic issue development = 25 man-wk

TABLE 2. (contd)

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

C_D = (25 man-wk)($2270/man-wk) = $5.68E+4

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

4 man-wk/plant (to review implementation problems on a per-plant basis) (Same for PWRs and BWRs)

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

C = (4 man-wk/plant)($2270/man-wk) = $9.08E+3/plant

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

NC = (134)($9080/plant) = $1.22E+6

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

It is assumed that 0.2 man-wk/py is required for review of maintenance and installation of new equipment. This estimate applies to both PWRs and BWRs.

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

C_o = (0.2 man-wk/py)($2270/man-wk) = $4.54E+2/py

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

$N\bar{T}C_o$ = (134)(28.3 yr)(4.54E+2/py) = $1.72E+6

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$3.0E+6	$4.1E+6	$1.9E+6

REFERENCES

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

U.S. NRC 1980. NRC Action Plan Developed as a Result of the TMI-2 Accident. NUREG-0660, U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: III.D.2.1, Radiological Monitoring of Effluents

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Present post-accident effluent monitoring relies on projections from a high-range noble gas monitor with sampling and laboratory analysis for radioiodines and particulates. The three-hour delay associated with sampling and analysis could be eliminated by the provision of improved effluent monitors incorporating continuous real-time readouts and automatic sample change capability. The result would be substantially better knowledge of actual releases allowing more timely and appropriate recommendation of public protective actions.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	8500
OCCUPATIONAL DOSES:	
SIR Implementation =	36
SIR Operation/Maintenance =	0
Total of Above =	36
Accident Avoidance =	0

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	87
SIR Operation/Maintenance =	17
Total of Above =	100
Accident Avoidance =	0
NRC COSTS:	
SIR Development =	0.35
SIR Implementation Support =	0.091
SIR Operation/Maintenance Review =	0
Total of Above =	0.44

RADIOLOGICAL MONITORING OF EFFLUENTS

ISSUE III.D.2.1

1.0 SAFETY ISSUE DESCRIPTION

This safety issue is part of the TMI) Action Plan, Task III.D.2.1, Public Radiation Protection Improvement, NUREG-0660 (NRC 1980c). Item One of this task requires development and implementation of acceptance criteria for monitors used to evaluate effluent releases under accident and post-accident conditions. Criteria would be developed for pathways to be monitored (stack, plant vent, steam dump vents) as well as for monitoring instrumentation. To meet the new criteria, licensees would have to develop, procure and install monitoring systems which are currently beyond state-of-the-art. This is seen to encompass the requirements in NUREG-0578, Recommendation 2.1.8-b (NRC 1979), and Appendix 2 to NUREG-0654 (NRC 1980b).

The envisioned monitoring system would provide automatic on-line analysis of airborne effluents, including isotopic analyses of particulate, radioiodine and gas samples. To prevent saturation of detectors, an automatic sample cartridge changeout feature would be included. The system would include microprocessor control and real-time readouts, and would be located in a low post-accident background area. The sampling system would be designed to provide a representative sample under anticipated accident release conditions.

A PWR steam dump sampling and monitoring system would be provided for PWR safety relief and vent values. Such a system might consist of a noble gas monitor and a radioiodine sampling and monitoring system. The features of such a system would be similar to the above described airborne effluent monitor, with two notable differences: The system would be required to function in a very high-humidity (steam-air mixture) environment, and operation would only be required during actual steam venting. Because such venting is usually of a short-term or intermittent duration, the monitoring system activation could be keyed to the opening of the vents.

Liquid effluents are not envisioned as posing a major release pathway because licensees typically have installed or are installing adequate storage capacity to prevent discharges. Consequently, present liquid effluent monitoring systems are considered adequate.

2.0 SAFETY ISSUE RISK AND DOSE

The estimated public risk reduction and occupational dose associated with improved radiological monitoring of airborne effluents are described in the following two sections, with results quantified in Tables 1 and 2.

PUBLIC RISK REDUCTION

The magnitude of public risk reduction attributable to improved radiological monitoring of airborne effluents is not certain, but it is estimated to range from zero to one percent (1%). An estimate of one percent is used in this analysis based on the following logic.

Present radiological monitoring requirements, as contained in NUREG-0737 (NRC 1980a), require real-time noble gas monitoring with sampling and laboratory analyses capabilities for radioiodines and particulates. Design basis conditions defined in NUREG-0737 (100 μCi/cc radioiodines and particulates, 30-minute sample time) indicate that sample collection devices would pose special handlng and analysis problems due to very high radioactivity buildup. Consequently, licensees have typically provided alternate sample collection and analysis procedures. Execution of those procedures is estimated to require between two and three hours. During this time, radioiodine and particulate releases would be estimated based on computer-modeled interpretation of noble gas monitor readings, or on previous post-accident containment atmosphere analysis results if such results were available. Public protective action recommendations would be made based on modeled estimates rather than actual effluent data. It is assumed that these recommendations would err on the conservative side (e.g., evacuation when it is not really required) due to the conservatism built into the modeled source terms for radioiodine and particulate releases.

Requiring licensees to have more sophisticated airborne effluent monitors would reduce the time required for obtaining actual radioiodine and particulate release data to 15 minutes, and essentially eliminate reliance on conservative theoretical release models extrapolated from noble gas monitor readings. As projected by this safety issue resolution (SIR), real-time isotopic monitoring would save nearly two hours in arriving at realistic protective action recommendations based on actual releases.

Under these circumstances, the public risk reduction would be directly attributed to the decrease in public radiation exposure which results from a more rapid assessment of the radioactive releases (about a two-hour saving in analysis time). There may also be a public risk reduction due to non-evacuation. This could result from better knowledge of the isotopic releases, eliminating the need for evacuation (presumed to exist if release knowledge is based only on noble gas monitor data). Non-evacuation results in fewer evacuation-related risks (e.g., traffic accidents), the avoidance of which may outweigh the radiation exposure received. However for this analysis, it is assumed that the public risk reduction results primarily from the first effect (decrease in exposure due to more rapid assessment).

While protective actions can be recommended based on effluent releases in progress, the probability for a core-melt scenario is such that actions would be recommended based on anticipated releases prior to the actual releases themselves. Under that assumption, monitoring effluent release would have little or no impact on public risk, and would be mainly for confirmation and quantification. This SIR would not impact core-melt accident frequency.

OCCUPATIONAL DOSE

It is anticipated that improvement of radiological monitoring of airborne effluents would have no sigificant impact on occupational dose. The dose required to install equipment would probably not exceed 0.5 man-rem, which is negligible compared to the typical 800 man-rem/yr required to operate a plant. Minor man-rem savings might occur under accident conditions due to better direction of field survey teams; however, such savings would be negligible compared to the 19,900 man-rem total associated with response and clean-up following an accident.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Radiological Monitoring of Effluents (III.D.2.1)

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

All 134 plants

	N	$\bar{T}$ (yr)
PWR	90	28.8
BWR	44	27.4

3. Plants Selected for Analysis:

Oconee 3 - representative PWR

Grand Gulf 1 - representative BWR

4. Parameters Affected by SIR:

The dose factors R for PWR release categories 1-7 and BWR release categories 1-4 are assumed to be affected by the SIR (Appendix D of Andrews et al. 1983).

5. Base-Case Values for Affected Parameters:

Original values are used from Appendix D (Andrews et al. 1983).

6-7. Steps Related to Affected Accident Sequences, Release Categories and Their Base-Case Frequencies:

These are not affected. Original frequencies from Appendices A & B (Andrews et al. 1983) are used for calculations in Steps 8 and 9.

TABLE 1. (contd)

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

$\bar{F}_{(PWR)}$ = 8.2E-5/py $\bar{F}_{(BWR)}$ = 3.7E-5/py

9. Base-Case, Affected Public Risk (W):

W_{PWR} = 207 man-rem/py W_{BWR} = 250 man-rem/py

10. Adjusted-Case Values for Affected Parameters:

The dose factors R are assumed to decrease by 1% due to the SIR. Their adjusted-case values become

	man-rem		man-rem
R(PWR-1) =	5.3E+6	R(BWR-1) =	5.3E+6
R(PWR-2) =	4.8E+6	R(BWR-2) =	7.0E+6
R(PWR-3) =	5.3E+6	R(BWR-3) =	5.0E+6
R(PWR-4) =	2.7E+6	R(BWR-4) =	6.0E+5
R(PWR-5) =	9.9E+5		
R(PWR-6) =	1.5E+5		
R(PWR-7) =	2.3E+3		

11-12. Steps Related to Adjusted-Case Frequencies of Affected Accident Sequences and Release Categories:

These are not affected. Original frequencies from Appendices A & B (Andrews et al. 1983) are used for calculations in Steps 13 and 14.

13. Adjusted-Case, Affected Core-Melt Frequency ($\bar{F}^*$):

No change from base case.

$\bar{F}^*_{PWR}$ = 8.2E-5/py $\bar{F}^*_{BWR}$ = 3.7E-5/py

14. Adjusted-Case, Affected Public Risk (W^*):

W^*_{PWR} = 205 man-rem/py W^*_{BWR} = 248 man-rem/py

15. Reduction in Core-Melt Frequency ($\Delta\bar{F}$):

None

TABLE 1. (contd)

16. Per-Plant Reduction in Public Risk (ΔW):

ΔW_{PWR} = 2.1 man-rem/py ΔW_{BWR} = 2.5 man-rem/py

17. Total Public Risk Reduction, $(\Delta W)_{Total}$:

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
8500	2.4E+7	0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Radiological Monitoring of Effluents (III.0.2.1)

2. Affected Plants (N):

All 134 plants

		N
PWR:	Operating	47
	Planned	43
BWR:	Operating	24
	Planned	20

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

		$\bar{T}$ (yr)
PWRs:	47 Operating	27.7
	43 Planned	30.0
BWRs:	24 Operating	25.2
	20 Planned	30.0

4-5. Steps Related to Occupational Dose Reduction Due to Accident Avoidance:

Since there is no change in core-melt frequency, there is no reduction in occupational dose from accident avoidance.

TABLE 2. (contd)

6. Per-Plant Utility Labor in Radiation Zones for SIR Implementation:

Dose is estimated directly in next step.

7. Per-Plant Occupational Dose Increase for SIR Implementation (D):

D = 0.5 man-rem/operating plant.

8. Total Occupational Dose Increase for SIR Implementation (ND):

ND = 71(0.5) = 36 man-rem

9-11. Steps Related to Occupational Dose Increase for SIR Operation and Maintenance:

No additional work in radiation zones beyond current levels is foreseen for SIR operation and maintenance. Thus, $D_o = 0$.

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
36	110	12

3.0 SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. Industry costs include both implementation and operation components. Results are summarized in Table 3.

The industry cost for equipment development, installation, support facilities, and construction is estimated at $600,000 per plant. Development of procedures, software, and calibration for the equipment is estimated to require 16 man-wk of effort, with an additional 4 man-wk of effort for the initial training of all licensee operators and health physics personnel. The recurring industry operation and maintenance costs are estimated at 2 man-wk/py for retraining, 1 man-wk/py for calibration, and reduction of 1 man-wk/py (reduced laboratory analyses due to fully automated system) for a net increase of 2 man-wk/py. Material requirements are estimated at $2,000/py beyond present systems, including sample cartridges and spare parts. There is no accident-avoidance cost term for the SIR because improved radiological effluent monitoring systems would impact neither accident frequency nor costs of cleanup and refurbishing.

The NRC cost is assumed to be limited to development and implementation support costs. Since it is assumed that the new radiological monitoring systems would require no periodic inspection effort beyond that required for current systems, no additional NRC operation cost is envisioned.

The NRC development costs include 1.5 man-yr and $200,000 for research, criteria development, and engineering development. These estimates are based on information contained in the TMI Action Plan Task III.D.2.1 description (NUREG-0660, NRC 1980c). These resource requirements represent a lump sum total which could not meaningfully be broken down on a per-plant basis. NRC administrative and technical effort associated with the review and approval of licensee submittals is estimated at 0.3 man-wk/plant.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Radiological Monitoring of Effluents (III.D.2.1)

2. Affected Plants (N):

   All 134 plants (90 PWRs and 44 BWRs).

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

	$\bar{T}$ (yr)
90 PWR	28.8
44 BWR	27.4
All LWR	28.3

Industry Costs (Steps 4 through 12)

4-5. Steps Related to Industry Cost Savings Due to Accident Avoidance:

   Since there is no change in core-melt frequency, there is no industry cost savings from accident avoidance.

6. Per-Plant Industry Resources for SIR Implementation:

   Labor = 16 man-wk/plant (develop procedures, software, calibrate equipment) + 4 man-wk/plant (initial training) = 20 man-wk/plant

   Equipment (development, installation, support facilities, construction) - cost is estimated directly in next step.

TABLE 3. (contd)

7. Per-Plant Industry Cost for SIR Implementation (I):

Labor = (20 man-wk/plant)($2270/man-wk) =	$4.54E+4/plant
Equipment =	$6.00E+5/plant
I =	$6.45E+5/plant

8. Total Industry Cost for SIR Implementation (NI):

NI = 134($6.45E+5) = $8.65E+7

9. Per-Plant Industry Labor for SIR Operation and Maintenance:

2 man-wk/py (net increase - See text Section 3.0.)

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_o):

I_o = (2 man-wk/py)($2270/man-wk) = $4540/py

11. Total Industry Cost for SIR Operation and Maintenance ($N\bar{T}I_o$):

$N\bar{T}I_o$ = 134(28.3)($4540) = $1.72E+7

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$1.0E+8	$1.5E+8	$6.0E+7

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Labor = 1.5 man-yr

Additional development cost is estimated directly in next step.

14. Total NRC Cost for SIR Development (C_D):

C_D = (1.5 man-yr)($1.0E+5/man-yr) + $2.0E+5 = $3.5E+5

15. Per-Plant NRC Labor for Support of SIR Implementation:

0.3 man-wk/plant (See text Section 3.0.)

TABLE 3. (contd)

16. Per-Plant NRC Cost for Support of SIR Implementation (C):

$$C = (0.3 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$680/\text{plant}$$

17. Total NRC Cost for Support of SIR Implementation (NC):

$$NC = 134(\$680) = \$9.1E+4$$

18-20. Steps Related to NRC Cost for Review of SIR Operation and Maintenance:

No additional effort beyond current levels is foreseen for NRC review of SIR operation and maintenance. Thus, $C_o = 0$.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$4.4E+5	\$6.2E+5	\$2.6E+5

REFERENCES

Andrews, W. B., et. al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development, NUREG/CR-2800, PNL-4297. Pacific Northwest Laboratory, Richland, Washington.

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U.S. NRC. 1980b. Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants. NUREG-0654, U.S. Nuclear Regulatory Commission, Washington, D.C.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: III.D.2.2, Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Improvements in the understanding of radioiodine partitioning in nuclear power reactors and of the environmental behavior of radioiodine, carbon-14, and tritium following an accident and during normal operation can be made through further research in these areas. The primary result would be improved calculational methods concerning accident source terms, releases, and offsite public doses for radioiodine, carbon-14, and tritium. Results of the research will be used to revise the Standard Review Plan and Regulatory Guides.

AFFECTED PLANTS

	BWR:	Operating = 24	Planned = 20
	PWR:	Operating = 47	Planned = 43

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	0[(a)]
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	0

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	-48
SIR Operation/Maintenance =	-340
Total of Above =	-390
Accident Avoidance =	0
NRC COSTS:	
SIR Development =	7.5
SIR Implementation Support =	0.61
SIR Operation/Maintenance Review =	0
Total of Above =	8.1

(a) While the "true" risk reduction is believed to be zero, "perceived" risk could be affected. A perceived reduction of 7.7E+4 man-rem is estimated.

RADIOIODINE, CARBON-14, AND TRITIUM PATHWAY DOSE ANALYSIS ISSUE III.D.2.2

1.0 SAFETY ISSUE DESCRIPTION

Improvements in the understanding of radioiodine partitioning in nuclear power reactors and of the environmental behavior of radioiodine, carbon-14, and tritium following an accident and during normal operation can be made by further research.

Iodine isotopes are considered to be major contributors to the occupational and public dose during a loss-of-coolant accident (LOCA), along with noble gases and fission products. Recent study in these areas is documented in NUREG-0772 (NRC 1981). There are three major conclusions of that study: 1) uncertainties in predicting atmospheric release source terms are very large (at least a factor of ten); 2) source terms for certain accident sequences may have been overestimated in past studies, e.g., WASH-1400 (NRC 1975); and 3) cesium iodide should be the predominant chemical form of iodine under severe accident conditions.

These conclusions indicate that methodology and assumptions currently used for evaluating radioiodine releases may result in unrealistic estimates (e.g., Regulatory Guides 1.3 and 1.4). Also indicated is that more research in aerosol behavior and fission product chemistry is needed in order to improve and support calculational methodologies concerned with radioiodine partitioning, fission product behavior, and others.

It is assumed that further study will improve understanding of this safety issue and result in more realistic assumptions and methods for evaluating source terms, releases, and environmental behavior of radioiodine, carbon-14, and tritium following an accident. This research will not affect accident frequencies at nuclear power plants. However, the results of these studies are assumed to be used to revise the Standard Review Plan and Regulatory Guides.

It is assumed that these Regulatory Guide revisions will result in reducing the size of current emergency planning zones (EPZs) from a ten-mile radius to a two-mile radius. This assumption is based upon a reduction by a factor of ten in source terms resulting from a core-melt accident, which translates into a reduction in dose concentration at a particular distance from the nuclear reactor--also by a factor of ten. Assuming neutral weather conditions with a 30-meter-high plume, the offsite dose predicted at two miles from the accident scene, using the reduced source term assumption, would be the same as that currently predicted at ten miles from the reactor.

The reduction in the size of the EPZ is assumed to reduce the amount of siren warning equipment needed by a licensee. This, in turn, should reduce the cost of siren system maintenance over the life of a plant. In addition, there should be cost reductions associated with the emergency planning program regarding state and local agency involvement, public information, and potassium iodide distribution (if applicable).

It is assumed that this safety issue resolution (SIR) would result in negligible cost savings for currently installed in-plant equipment. A reduction in core-melt accident source terms is assumed not to justify changing future in-plant equipment requirements (i.e., smaller filtering capacity, ECCS equipment, etc.). It might, however, avoid requirements for future equipment that is being considered because of the current source term assumptions (e.g., underground containment pressure relief filtering systems).

Studies that could be involved with this SIR are contained in NUREG-0660 (NRC 1980). Another possibility would be small-scale fuel melting experiments. This type of research is estimated to cost $7.5 million if conducted at existing facilities. Fuel-melt research could greatly improve the understanding of radioiodine partitioning and fission product behavior and is the main factor in determining total NRC cost.

2.0 SAFETY ISSUE RISK AND DOSE

The estimated public risk reduction and occupational dose associated with better understanding of radioiodine partitioning in post-accident situations and carbon-14 and tritium behavior in the environment are described in the following two sections.

PUBLIC RISK REDUCTION

Two aspects of public risk must be considered for this SIR. First is the "true" risk based on the public dose that would be received from core-melt accident releases. Second is the "perceived" risk, based on a modeled theoretical accident. Changes in theoretical models will not impact the true risk; however, changes in models may significantly impact the perceived risk which is the basis for design and procedural requirements. It is anticipated that this SIR will have no impact on the true risk but that it can be expected to significantly reduce the perceived risk (see Table 1). Again, the true public risk and core-melt frequency reductions are zero.

This Safety Issue involves the development of better modeling capability, particularly with regard to a more realistic understanding of radioiodine partitioning. For predominant core-melt accident scenarios, radioiodines are the major contributors to public dose. Present modeling of radioiodine releases is detailed in Regulatory Guides 1.3 and 1.4 and postulates 25% of the core radioiodine inventory to be available for immediate release. Preliminary results from the TMI-2 accident indicate that 25% might be an unrealistically conservative assumption. It is now suspected that iodine retention in coolant and particulate plateout on containment surfaces would result in iodine release a factor of 10 to 100 lower than the previously postulated 25%.

The anticipated true public risk reduction from this SIR is zero. If an accident occurred, the existing design of the plant and the timeliness of public protective action recommendations would limit public dose and the resulting true risk. Assuming that protective action recommendations are appropriate, the public dose would be received regardless of any theoretical predictions. Changes in the theoretical model would be irrelevant to the events then in process; hence, no reduction in true public risk is assumed to result.

Major changes are predicted for the perceived public risk. As mentioned above, radioiodine releases are now generally believed to be overstated by a factor of 10 to 100. A reduction in the source term for accident consequence assessment by such a factor would translate directly into a corresponding reduction in perceived public dose and risk. Thus, the accident consequence data presented in Table D.1 of NUREG/CR-2800 (Andrews et al. 1983) are probably overstated. For the perceived (theoretical) risk calculations, these are reduced by a factor of ten to reflect a more realistic source term. The public risk reduction for all affected plants listed in Step 17 of Table 1 thus represents elimination of undue conservatism in the risk models.

OCCUPATIONAL DOSE

No change in occupational dose is anticipated from the resolution of this safety issue. This assumes that plant design and operation will not be affected by source terms theoretically reduced to reflect a more realistic core-melt scenario. Thus, the Occupational Dose Work Sheet is omitted.

TABLE 1. (Perceived) Public Risk Reduction Work Sheet

(Note: See Section 2.0 for discussion of perceived versus true risk.)

1. Title and Identification Number of Safety Issue:

   Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis (III.D.2.2)

2. Affected Plants (N) and Average Remaining Lives ($\bar{T}$):

   All 44 BWRs and 90 PWRs will be affected, but not until SIR implementation is complete in 1986. The average lives of operating plants are thus four years less than the values given in Appendix C of NUREG/CR-2800 (Andrews et al. 1983).

	$\bar{T}$ (yr)
24 operating BWRs =	21.2
47 operating PWRs =	23.7

   Since planned plants will not be affected until 1986, those beginning operation prior to 1986 will have lives <30 yr for this SIR. Review of

TABLE 1. (contd)

2. Affected Plants (N) and Average Remaining Lives ($\bar{T}$) (contd):

Appendix C indicates that the average life of the 63 planned plants relative to 1986 is $\bar{T}$ = 29 yr. The average lives for the 44 BWRs and 90 PWRs are as follows for this issue:

	$\bar{T}$ (yr)
44 operating BWRs =	24.7
90 operating PWRs =	26.2

3. Plants Selected for Analysis:

Grand Gulf - representative BWR

Oconee 3 - representative PWR

4. Parameters Affected by SIR:

The dose factors R for PWR release categories 1-7 and BWR release categories 1-4 are assumed to be affected.

5. Base-Case Values for Affected Parameters:

Original values are used (Appendix D, Andrews et al. 1983).

6-7. Steps Related to Affected Accident Sequences and Release Categories:

These are not affected. Original frequencies (Appendices A and B, Andrews et al. 1983) are used for calculations in Steps 8 and 9.

8. Base-Case, Affected Core-Melt Frequency ($\bar{F}$):

$\bar{F}_{BWR}$ = 3.7E-5/py $\bar{F}_{PWR}$ = 8.2E-5/py

9. Base-Case, Affected Public Risk (W):

W_{BWR} = 250 man-rem/py W_{PWR} = 207 man-rem/py

10. Adjusted-Case Values for Affected Parameters:

The dose factors R are assumed to decrease by 10%.

Category	R (man-rem)
PWR-1	4.86E+6
PWR-2	4.32E+6
PWR-3	4.86E+6

TABLE 1. (contd)

10. Adjusted-Case Values for Affected Parameters (contd):

Category	R (man-rem)
PWR-4	2.43E+6
PWR-5	9.00E+5
PWR-6	1.35E+5
PWR-7	2.07E+3
BWR-1	4.86E+6
BWR-2	6.39E+6
BWR-3	4.59E+6
BWR-4	5.49E+5

11-12. Steps Related to Affected Accident Sequences and Release Categories:

These are not affected. Original frequencies are used for calculations in Steps 13 and 14.

13. Adjusted-Case, Affected Core-Melt Frequency ($\bar{F}^*$):

No change from base case (see Step 8).

14. Adjusted-Case, Affected Public Risk (W^*):

W^*_{BWR} = 225 man-rem/py W^*_{PWR} = 186 man-rem/py

15. Reduction in Core-Melt Frequency ($\Delta\bar{F}$):

None

16. Per-Plant Reduction in Public Risk (ΔW):

ΔW_{BWR} = 25 man-rem/py ΔW_{PWR} = 21 man-rem/py

17. Total Public Risk Reduction, $(\Delta W)_{Total}$:

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
7.7E+4	2.4E+7	0

3.0 SAFETY ISSUE COSTS

This SIR would not affect accident frequency at nuclear reactor power plants nor related costs or savings. The SIR presumably results in Regulatory Guide revisions, which are assumed to result in the use of lower radioiodine source terms in accident scenarios. This, in turn, is assumed to result in an EPZ reduction from a ten- to a two-mile radius, and translates to utility cost savings for warning siren equipment, operation, and maintenance. It would also lower a plant's emergency planning program costs regarding public information, and local and state agency involvement. The following assumptions are used in determining the SIR costs and savings:

Industry Costs

Implementation

Cost for 10-mile EPZ (sirens, warning system, radios, labor, etc.)	= $2.0E+6/plant
Cost for 2-mile EPZ	= $1.6E+5/plant
Cost Savings for Change from 10 to 2-mile EPZ[a] (Difference of Above)	$1.84E+6/plant

Operation/Maintenance

Cost Savings on Reduced Size of Warning System	= $6.1E+4/py
Cost Savings on Reduced Local and State Agency Involvement	= $3.9E+4/py
Total of Above[b]	= $1.00E+5/py

NRC Costs

Development

Research Cost (all plants)	= $7.0E+6
Cost for Research Administration and Regulatory Guide Revision (all plants)	= $5.0E+5
Total of Above[a]	= $7.50E+6

Implementation Support

Check Revised EPZ Plans and Dose Calculations (2 man-wk/plant)	= $6.08E+5 (all plants)

(a) Applicable only to planned plants becoming operational in 1986 or beyond (7 BWRs and 19 PWRs, 26 in all).

(b) Applicable to all plants, but not until 1986.

Operation/Maintenance Review

None

The cost analyses are summarized in Table 2.

TABLE 2. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis (III.D.2.2)

2. Affected Plants (N):

For industry implementation cost, only planned plants becoming operational in 1986 or beyond are affected (7 BWRs and 19 PWRs, 26 in all). For industry operation/maintenance cost, all plants (44 BWRs and 90 PWRs, 134 in all) are affected, but operation/maintenance is assumed not to begin until 1986 (when the SIR implementation is complete). For NRC costs, all 134 plants are affected.

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

Since operating plants will not be affected until 1986, the average remaining lives are four years less than the values listed in Appendix C of NUREG/CR-2800 (Andrews et al. 1983).

	$\bar{T}$ (yr)
24 operating BWRs =	21.2
47 operating PWRs =	23.7

Since planned plants will not be affected until 1986, those beginning operation prior to 1986 will have lives <30 yr for this SIR. Review of Appendix C indicates that the average life of the 63 planned plants relative to 1986 is $\bar{T}$ = 29 yr. The average life for all 134 plants becomes 25.7 yr.

Industry Costs (Steps 4 through 12)

4-5. Steps Related to Industry Cost Savings Due to Accident Avoidance:

This SIR does not impact accident frequency or cost of cleanup and refurbishing. Thus, no cost savings result.

6. Per-Plant Industry Resources for SIR Implementation:

Costs are estimated directly in the next step.

TABLE 2. (contd)

7. Per-Plant Industry Cost for SIR Implementation (I):

$I = -\$1.84E+6/\text{plant}$ (cost savings for reduction in size of EPZ)

This applies only to the 26 planned plants becoming operational in 1986 or beyond.

8. Total Industry Cost for SIR Implementation (NI):

$NI = (26)(-\$1.84E+6/\text{plant}) = -\$4.78E+7$

9. Per-Plant Industry Labor for SIR Operation and Maintenance:

Costs are estimated directly in the next step.

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_O):

$I_O = -\$1.0E+5/\text{plant-yr}$ (cost savings for reduced size of warning system and less local and state agency involvement).

This applies to all 134 plants.

11. Total Industry Cost for SIR Operation and Maintenance ($N\bar{T}I_O$):

$N\bar{T}I_O = (134)(25.7\ \text{yr})(-\$1.0E+5/\text{py}) = -\$3.44E+8$

12. Total Industry Cost (S_I):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
-$3.9E+8	-$2.2E+8	-$5.7E+8

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Costs are estimated directly in the next step.

14. Total NRC Cost for SIR Development (C_D):

$C_D = \$7.50E+6$ (all plants) for research, research administration, and Regulatory Guide revision.

15. Per-Plant NRC Labor for Support of SIR Implementation:

The NRC is assumed to expend 2 man-wk/plant to check revised EPZ plans and plant dose calculations.

TABLE 2. (contd)

16. Per-Plant NRC Cost for Support of SIR Implementation (C):

(2 man-wk/plant)($2270/man-wk) = $4540/plant

This cost is the same for PWRs and BWRs.

17. Total NRC Cost for Support of SIR Implementation (NC):

NC = ($4540/plant)(134 plants) = $6.08E+5

18-20 Steps Related to NRC Cost for Review of SIR Operation and Maintenance:

No requirements beyond existing programs are anticipated. Thus, $C_o = 0$.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$8.1E+6	$1.2E+7	$4.3E+6

REFERENCES

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

Director, Office of Nuclear Regulatory Research, et al. 1981. Technical Bases for Estimating Fission Product Behavior During LWR Accidents. Prepared for the U.S. Nuclear Regulatory Commission by Battelle Memorial Institute, Columbus Laboratories, Columbus, Ohio.

U.S. NRC. 1975. Reactor Safety Study. WASH-1400, U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. NRC. 1980. NRC Action Plan Developed as a Result of the TMI-2 Accident. NUREG-0660, U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: III.D.2.5, Offsite Dose Calculation Manual

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Calculation of some public dose resulting from radiation releases is currently subject to variation by site. This task would produce a uniform and consistent manual for use by all NRC and plant personnel in estimating public dose during an accident.

AFFECTED PLANTS			
	BWR:	Operating = 24	Planned = 20
	PWR:	Operating = 47	Planned = 43

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	B50
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	0

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	1.2
SIR Operation/Maintenance =	0
Total of Above =	1.2
Accident Avoidance =	0
NRC COSTS:	
SIR Development =	0.13
SIR Implementation Support =	0.12
SIR Operation/Maintenance Review	0
Total of Above =	0.25

OFFSITE DOSE CALCULATION MANUAL

ISSUE III.D.2.5

1.0 SAFETY ISSUE DESCRIPTION

This safety issue is part of TMI action plan Task III.D.2, Public Radiation Protection Improvement. The purpose of this task is to improve public radiation protection in the event of a nuclear power plant accident. Item 5 requires that NRR prepare a manual to be used by the NRC and plant personnel to estimate maximum individual and population doses during an accident. The manual would include formulations with which to combine source term and meteorological measurements, thus determining offsite dose rates in a manner that would be standard among all parties making decisions on public protection and emergency response. This appears to add additional requirements for the criteria listed in Appendix 2 to NUREG-0654 (NRC 1980), which establishes criteria for automated assessment of radiation doses in the event of an accident.

It is uncertain whether this Safety Issue Resolution (SIR) would directly impact public risk. Given in Attachment 1 is a discussion assuming a 0.1 percent reduction in public dose factors. This follows the Public Risk Reduction Work Sheet. Resolution of this issue will not impact core-melt or release category frequencies. Only the public dose factors, as given in Table D.1 of the Guidelines (Andrews et al. 1983), will be reduced. Occupational dose will be unaffected.

2.0 SAFETY ISSUE RISK AND DOSE

Results of the analysis for public risk reduction are summarized in Table 1. Note that this issue does not impact core-melt frequency. It is assumed that the public dose factors can be lowered by 0.1 percent through issue implementation.

The dose calculation manual is aimed strictly at public exposure. Issue implementation has no impact on occupational exposure. Thus, no Occupational Dose Work Sheet is provided.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Offsite Dose Calculation Manual (III.D.2.5)

TABLE 1. (contd)

2. Affected Plants (N) and Average Remaining Lives ($\bar{T}$):

	N	$\bar{T}$ (yr)
PWR	90	28.8
BWR	44	27.4

3. Plants Selected for Analysis:

Oconee 3 - representative PWR

Grand Gulf 1 - representative BWR

4. Parameters Affected by SIR:

Dose factors R for the seven PWR and four BWR release categories are assumed to be affected, see Appendix D of NUREG/CR-2800 (Andrews et al. 1983).

5. Base-Case Values for Affected Parameters:

For the base case, the dose factors for PWR-1 through PWR-7 and BWR-1 through BWR-4 are the same as those given in Appendix D (Andrews et al. 1983).

6-8. Steps Related to Affected Accident Sequences, Release Categories, Their Base-Case Frequencies, and Base-Case Core-Melt Frequency:

SIR has no effect upon accident sequence frequencies. Thus, these steps are omitted, as are Steps 11-13.

9. Base-Case, Affected Public Risk (W):

The original release category frequencies for Oconee and Grand Gulf as given in PNL-4297 (Andrews et al. 1983) are used to estimate W.

Oconee: W = 207 man-rem/py

Grand Gulf: W = 250 man-rem/py

10. Adjusted-Case Values for Affected Parameters:

The dose factors R for the seven PWR and four BWR release categories are assumed to decrease by 0.1% (see Attachment 1). As measured to two significant figures, no change from the base- to the adjusted-case value occurs for the public dose factors.

TABLE 1. (contd)

11-13. Steps Related to Adjusted-Case, Affected Accident Sequences, Release Categories, Core-Melt Frequencies:

As in Steps 6-8, these steps are skipped.

14. Adjusted-Case, Affected Public Risk (W*):

As measured to three significant figures, no change from the base- to the adjusted-case values occurs for the affected public risk.

15. Reduction in Core-Melt Frequency ($\Delta\bar{F}$):

None.

16. Per-Plant Reduction in Public Risk (ΔW):

ΔW is calculated directly as 0.1% of the base-case, affected public risk (since all public dose factors are assumed to decrease uniformly by 0.1%). Thus,

Oconee: ΔW = (0.001)(207 man-rem/py) = 0.21 man-rem/py

Grand Gulf: ΔW = (0.001)(250 man-rem/py) = 0.25 man-rem/py

17. Total Public Risk Reduction, $(\Delta W)_{Total}$:

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
850	2.5E+7	0

ATTACHMENT 1

It is implied that access to accurate, consistent information on dose rates would allow authorities to reduce public exposure during a reactor accident. This further implies that the information would be used to direct public movement, and that the public would respond. The following chain of events is then required for dose rate reduction:

- Accurate information on future releases are relayed to authorities.
- The dose rate manual is used exclusively by authorities.
- More uniform information then becomes available to authorities.
- Timely, consistent decisions are made by authorities.
- Decisions are relayed in timely fashion to the public.
- The public acts on the above as directed.

This is essentially the chain of events followed with current procedures, with some incremental improvement in performance assumed. Note that public protection must rely on relatively long-term (24-48 hour) projection of release. Use of the dose rate manual may provide consistent information to officials. However, its final enactment in terms of dose reduction must still rely on decisions which must be made and relayed to a responsive public in advance of releases if exposure reduction is to occur. Any impact that a dose manual may have in reducing public exposure will most likely be diluted in the confusion and indecisions which could realistically be expected during a core-melt accident.

Note that efforts to reduce exposure by focusing attention on the decision-making process and relaying this information to the public (via television, etc.) can potentially provide positive effects by coordinating public response. However, any resulting increase or decrease in public dose would still depend on the accuracy and timely nature of the decisions made. Presented below is an attempt to weight the various steps required in the process of preventing public exposure through informed action. If all steps in the chain function properly, it is assumed that a total net reduction of approximately 3 percent is possible. The contribution due to each step is then determined, based on its judged relative effectiveness. A total reduction is given, then, of: $1-(.98)(.99)(.999)(9995)^3 = .032$, or 3.2%.

STEP	Estimated Effectiveness in Reducing Public Exposure: Best	Better	Good	Marginal	Estimated Percent Reduction
Reduce reactor accident emissions.	✓				--
Improve 24 to 48-hr. forecasting of releases after accident.		✓			2
Improve monitoring.			✓		1
Improve dose calculations (via resolution of Issue III.D.2.5).			✓		0.1[(a)]
Improve decision-making process.				✓	0.05
Improve information relay to public.				✓	0.05
Improve public response.				✓	0.05

(a) For SIR, this 0.1% reduction is assumed to affect the public risk through the release categories' dose factors.

3.0 SAFETY ISSUE COSTS

The NRC has already completed work on development of a portable computerized system for dose calculations to be used by the NRC Regional Offices as part of the program for NUREG-0654 (NRC 1980). This program has been developed to the point of field trials for the computerized system. As a result, NRC implementation costs for this issue can be reduced substantially. Based on the current development costs, an additional $125,000 to develop this package into a manual form for use by utilities will be assumed. It is estimated that NRC site representatives could spend a minimal amount of time (approximately 2 days) to evaluate initial utility performance with the package. This comes to $900 per site.

For the utilities, three man-weeks training for implementation are assumed. No additional annual recurring costs for training review classes beyond these already in place will be required.

The current NRC program is on a portable Osborne computer. The program can likely be adapted to other systems, but to be conservative, it will be assumed that site equipment requirements will be approximately $2000 for the minicomputer.

TABLE 2. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Offsite Dose Calculation Manual (III.D.2.5)

2. Affected Plants (N):

   All 134 plants will be affected.

	N
PWR	90
BWR	44

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

	$\bar{T}$ (yr)
PWR	28.8
BWR	27.4
All	28.3

TABLE 2. (contd)

Industry Costs (Steps 4 through 12)

4-5. Steps Related to Industry Cost Savings from Accident-Avoidance (ΔH):

Since there is no change in core-melt frequency, no cost savings result.

6. Per-Plant Industry Resources for SIR Implementation:

Labor = 3 man-wk/plant

Equipment = Minicomputer installation (costs estimated directly in next step).

7. Per-Plant Industry Cost for SIR Implementation (I):

Labor	= (3 man-wk/plant)($2270/man-wk)	=	$6810/plant
Equipment	= (minicomputer)	=	$2000/plant
		I =	$8810/plant

8. Total Industry Cost for SIR Implementation (NI):

NI = (134)($8810/plant) = $1.18E+6

9-11. Steps Related to Industry Cost for SIR Operation and Maintenance:

No SIR operation/maintenance beyond current levels is foreseen. Thus, $I_0 = 0$.

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$1.2E+6	$1.8E+6	$5.9E+5

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Costs estimated directly in next step.

14. Total NRC Cost for SIR Development (C_D):

C_D = $125,000 for development and issuance of offsite dose manual.

TABLE 3. (contd)

15. Per-Plant NRC Labor for Support of SIR Implementation:

2 man-days/plant (= 0.4 man-wk/plant)

16. Per-Plant NRC Cost for Support of SIR Implementation (C):

C = (0.4 man-wk/plant)($2270/man-wk) = $908/plant

17. Total NRC Cost for Support of SIR Implementation (NC):

NC = (134)($908/plant) = $1.22E+5

18-20. Steps Related to NRC Cost for Review of SIR Operation and Maintenance:

No additional review beyond current levels is foreseen. Thus, $C_o = 0$.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$2.5E+5	$3.3E+5	$1.6E+5

REFERENCES

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

U.S. NRC. November 1980. Criteria for the Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants. NUREG-0654, U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: III.D.3.2, Worker Radiation Protection Improvement: Health Physics Improvements

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Improvements in worker radiation protection can be made by increasing assurance of adequate workers' protection and allowing better utilization of workers, without impact on occupational dose or public risk. These improvements include requiring testing and certification of personnel dosimetry processors, radiation monitoring instruments, and air purifying respirators for radioiodine applications.

AFFECTED PLANTS	BWR:	Operating = 24	Planned = 20
	PWR:	Operating = 47	Planned = 43

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 0

OCCUPATIONAL DOSES:

SIR Implementation = 0

SIR Operation/Maintenance = 0

Total of Above = 0

Accident Avoidance = 0

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:

SIR Implementation = 2.0

SIR Operation/Maintenance = 33

Total of Above = 35

Accident Avoidance = 0

NRC COSTS:

SIR Development = 0.46

SIR Implementation Support = 0

SIR Operation/Maintenance Review = 0

Total of Above = 0.46

WORKER RADIATION PROTECTION IMPROVEMENT: HEALTH PHYSICS IMPROVEMENTS

ISSUE III.D.3.2

1.0 SAFETY ISSUE DESCRIPTION

This safety issue is part of TMI Action Plan Task III.D.3, "Worker Radiation Protection Improvements" (NRC 1980a). Four specific items were identified for safety issue resolution (SIR):

(1) Requirement for Use of Certified Personnel Dosimeter Processors

(2) Audible Alarm Dosimeter Regulatory Guide

(3) Develop Standard Performance Criteria for Radiation Survey and Monitoring Instruments

(4) Develop Air Purifying Respirator Radioiodine Cartridge Testing and Certification Criteria.

With the exception of Item #2, each of these SIRs is evaluated in this analysis. Item #2 was excluded since a regulatory guide has been issued in final form prior to this evaluation.

The safety issues and their proposed resolutions do not impact public risk, nor are they expected to affect occupational dose. They relate to the rights of workers to be assured of adequate radiation protection, and would reduce stress during the performance of work in radiation zones.

Each of the four items identified for the issue is discussed in the following subsections.

1.1 REQUIREMENT FOR USE OF CERTIFIED PERSONNEL DOSIMETRY PROCESSORS

The proposed SIR would amend 10 CFR 20 to require that only nationally certified dosimetry processors be used by NRC licensees for personnel radiation dosimetry. Processors would be required to meet ANSI N13.11 (or its replacement standard) criteria for testing. Certification of processors would be performed by the National Voluntary Laboratory Accreditation Program (NVLAP), administered under the auspices of the Department of Commerce (DOC).

This safety issue and its resolution do not specifically address core-melt accidents, nor the public risk, occupational dose, or accident-avoidance costs associated with such accidents. Rather, the issue is related to the worker's right to accurate measurements of occupational dose. The proposed resolution would require accurate and precise determinations of individual worker doses using dosimeters, readout systems, and processing procedures certified to be capable of meeting minimum criteria defined in a national standard. The administrative and regulatory limits for occupational dose would be unaffected by this work.

A draft standard (ANSI N13.11) for dosimeter testing was issued for trial use in 1978. This standard has undergone substantial testing and remains only to be finalized for issuance as a new ANSI standard. Once issued, it will form the basis for amending 10 CFR 20. Testing and certification of dosimeter processors for criteria contained in this standard will be performed by NVLAP under DOC.

The anticipated impact on the commercial nuclear power plant industry is relatively minor for this SIR. However, major impact can be expected on small materials licensees who might be required to undertake major equipment or service expenditures. The impact on those licensees is beyond the scope of this analysis.

1.2 AUDIBLE ALARM DOSIMETER REGULATORY GUIDE

This element of the safety issue is considered resolved without the need for further analysis. The subject regulatory guide was issued as Regulatory Guide 8.28, "Audible-Alarm Dosimeters," in August 1981. This item is not discussed further.

1.3 DEVELOP STANDARD PERFORMANCE CRITERIA FOR RADIATION SURVEY AND MONITORING INSTRUMENTS

Testing of radiation survey and monitoring instruments will provide a high degree of quality assurance that instruments are capable of performing intended functions under specified conditions. This will allow consistent utilization of workers without impacting current individual or collective occupational dose. A draft standard for health physics instrumentation testing (ANSI N42.17-D2) has been developed.

This standard will undergo a field trial period, using off-the-shelf instruments, to determine its adequacy. This trial period is presently estimated to continue through FY84 and is jointly funded by NRC and the Department of Energy (DOE) at $400,000 each. Following the trial period, a final standard will be adopted by NRC, and only those instruments meeting this standard will be acceptable for use in NRC licensed facilities.

At this time, a plan for implementing the testing program has not been developed. It is anticipated, however, that independent testing laboratories would, for a fee, test instruments submitted by vendors or reactor licensees.

The testing laboratories would be certified by NVLAP under DOC. Costs associated with NVLAP certification and instrument testing fees would be passed on to industry in the form of higher instrument prices.

1.4 DEVELOP AIR PURIFYING RESPIRATOR RADIOIODINE CARTRIDGE TESTING AND CERTIFICATION CRITERIA

Air purifying respirators are not currently acceptable for radioiodine protection due to the lack of accepted test procedures for certifying cartridge filtering efficiency. The result is that bulky, self-contained breathing apparatus (SCBA) must be worn by workers in radioiodine environments. Such environments are expected during and after core-melt accidents. The result of wearing SCBA is to substantially reduce worker efficiency, due to physical stress and the relatively short working time limited by air tank capacity. Use of air purifying respirators would reduce worker stress and improve worker efficiency.

It is expected that operator dose would be unaffected by the availability of respirators. Immediately after an accident, because of immediate hazards, SCBA would still be used. During long-term recovery activities, respirators would be used. However, any reduction in external dose resulting from efficient use of time in radiation zones is expected to be offset by the reduced effectiveness of the respirators, compared to SCBA, in avoiding internal exposures. Criteria and test procedures for radioiodine cartridges have been under development by Los Alamos National Laboratory using NRC funds. The technology has been developed and is in the process of being transferred to the National Institute of Occupational Safety and Health (NIOSH). When transfer is complete, it is anticipated that NIOSH will amend 30 CFR 11 to incorporate the testing methods and criteria into respirator test and certification schedules. Respirator and cartridge manufacturers would submit products for certification testing, and periodic quality control checks would be performed.

Following establishment of certification programs, NRC evaluation is anticipated regarding the need to specify the quantity and types of respirators necessary for normal and emergency use at a typical power reactor.

This safety issue will have no impact on public risk associated with core-melt accidents. The occupational dose impact is also considered to be zero, the benefit to workers being reduced stress, improved comfort and, consequently, better worker performance.

2.0 SAFETY ISSUE RISK AND DOSE

Collectively, the safety issue items will not impact core-melt accident frequency or severity. Therefore, no public risk reduction will result from their resolution. Furthermore, they will neither collectively impact the occupational dose received for core-melt accident cleanup, nor will they require additional dose to implement or maintain their resolution. Therefore, the occupational dose impact is zero. Resolution of these items will provide worker benefit, but not in terms of dose reduction. Workers will have added confidence in dose and dose rate measurements, and improved comfort with lower physical stress levels while using respiratory protection in certain radioiodine environments. No work sheets for public risk reduction or occupational dose are prepared.

3.0 SAFETY ISSUE COSTS

The costs associated with resolution of each of the three outstanding items of this safety issue are discussed in the following subsections. The results of the analyses are summarized in Table 1.

3.1 REQUIREMENT FOR USE OF CERTIFIED PERSONNEL DOSIMETRY PROCESSORS

Costs for resolution of this item (#1) include those to finalize the dosimeter testing standard, amend 10 CFR 20 to require certification of dosimeter processors, and power industry costs to comply with certification requirements. Because most power reactor licensees use commercial dosimetry processors or have in-house capability to meet the certification criteria, this resolution is not expected to greatly impact the power reactor industry. However, significant impact could be incurred by small materials licensees (e.g., hospitals and radiographers,) as a result of requiring certification. The evaluation of that impact is beyond the scope of this analysis, which is limited to commercial nuclear power plants.

NRC development costs for the dosimeter testing standard have already been incurred, with only nominal finalization costs remaining. These have been estimated at $20,000 (approximately 0.2 man-years of NRC labor required). Included in this estimate is the cost for writing and publishing a 10 CFR 20 amendment requiring certification in the Federal Register.

Costs for establishing the certification program will consist primarily of staffing a testing laboratory. These costs are estimated at $200,000/yr, including staff salaries ($160,000/yr), operating expenses ($25,000/yr) and equipment replacement ($15,000/yr). These costs are based on estimates contained in NUREG/CR-1064, Performance Testing of Personnel Dosimetry Services (NRC 1980b) and escalated from 1979 to 1982 dollars. It was assumed that existing office and laboratory facilities would be available for use, therefore, construction of such facilities is not included in the estimates. Initially, these costs would be incurred by NVLAP under DOC. However, they would be recovered ultimately by passing them on to users in the form of testing fees and increased processing costs. Thus, the costs to industry for establishing the certification program are estimated as operation/maintenance costs in this analysis. For reactor licensees using commercial dosimeter processing services, there would be no implementation cost. Added costs would, however, be incurred in the annual operating expenses for dosimetry services. These costs are estimated at one dollar per worker monitored each year. The number of workers requiring dosimeters at a typical nuclear plant is estimated at 2,000 per year, based on data contained in NUREG-0713, Occupational Radiation Exposure at Commercial Nuclear Power Reactors--1980 (NRC 1981). This number represents the arithmetic average of all monitored workers at 68 reactors. The annual industry operation and maintenance cost is, therefore, estimated at $2,000/py. This cost is assumed to apply to all plants.

A few plants electing to become certified may require complete dosimetry system replacement as an implementation cost. For purposes of this analysis, ten plants were assumed to take this action. Each plant was assumed to require a new dosimeter readout system ($100,000) and 5,000 dosimeters ($100,000 at $20 each). The total implementation cost for each plant would then be $200,000, or $2 million for ten plants.

3.2 DEVELOP STANDARD PERFORMANCE CRITERIA FOR RADIATION SURVEY AND MONITORING INSTRUMENTS

The costs for resolution of this item (#3) include development costs being jointly funded by NRC and DOE. These cover field trials of draft ANSI Standard N43.17-D4 for the fiscal years 1981-1984. Total funding level is $800,00D, divided between the NRC and DOE. The NRC share is $400,000. The outcome of this effort is expected to be an approved standard for testing of radiation protection instruments.

An additional $20,000 is estimated to be required by the NRC for the rule-making process requiring that this standard be met. No added NRC cost is anticipated with respect to SIR implementation, operation or maintenance. Present NRC inspection practices would not require modification.

To meet this standard, a testing program would have to be established by industry. While a plan to establish such a program does not currently exist, it is anticipated that commercial testing laboratories would provide this capability. These laboratories would be certified by NVLAP under DOE.

Annual costs for a testing laboratory are estimated as follows:

Salaries (professional + technicians)	$130,000
Operating Expenses	50,000
Equipment Replacement	15,000
NVLAP Fees	5,000
Total	$200,000/yr

It is estimated that four such laboratories would be required to service the industry; hence, the total added industry expense is estimated at $800,000/yr.

The implementation cost to equip these laboratories is reflected in the equipment replacement cost factor since it would be recovered from industry users over time. Therefore, implementation costs are considered negligible. Assuming that the $800,000/yr industry cost is evenly distributed among all 134 reactors, the added cost per reactor would be approximately $5,970/yr. It is not anticipated that this standard would impose additional labor requirements on the reactor licensees. Neither is it anticipated that licensees would be required to replace entire survey instrument inventories existing prior to implementation of this standard.

3.3 DEVELOP AIR PURIFYING RESPIRATOR RADIOIODINE CARTRIDGE TESTING AND CERTIFICATION CRITERIA

The costs for resolution of this item (#4) include NRC development costs, annual operation/maintenance expense and the cost of respirators and cartridges. There is no anticipated implementation cost for industry or the NRC. The NRC development cost is estimated at $20,000 and includes providing NIOSH with equipment (already developed and procured), operating procedures, and amending 30 CFR 11.

The annual industry operation/maintenance costs are primarily staff labor and expenses for the NIOSH testing and certification facility. These are estimated at approximately 0.75 man-yr/yr ($75,000/yr) labor and $10,000/yr expenses. These costs would eventually be transferred to reactor licensees as an included cost in the purchase price of respirators and cartridges. Assuming uniform distribution among all 134 reactors, the added cost per plant would be approximately $634/yr.

An increase in the inventory and use of respirator cartridges could be expected. However, this would be offset by reduced use and maintenance costs of SCBA. For this analysis, it is assumed that those costs are equivalent, and the added cost to industry would be only those costs associated with testing and certification.

No added operation/maintenance expense is anticipated for the NRC. Present inspection practices would adequately cover needs.

TABLE 1. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Worker Radiation Protection Improvement: Health Physics Improvements (III.D.3.2)

TABLE 1. (contd)

1. Title and Identification Number of Safety Issue (contd):

Items:

1) Requirement for Use of Certified Personnel Dosimetry Processors

2) Audible Alarm Dosimeter Regulatory Guide

3) Develop Standard Performance Criteria for Radiation Survey and Monitoring Instruments

4) Develop Air Purifying Respirator Radioiodine Cartridge Testing and Certification Criteria

2. Affected Plants (N):

All 134 plants. No distinction is needed between PWRs and BWRs or operating and planned plants.[a]

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

For all 134 plants, $\bar{T}$ = 28.3 yr

INDUSTRY COSTS (Steps 4 through 12)

4-5. Steps Related to Industry Cost Savings Due to Accident Avoidance

SIR does not affect core-melt accident frequency or severity. Therefore, $\Delta H = 0$.

6. Per-Plant Industry Resources for SIR Implementation:

Item #1

- Dosimeter Readout System
- 5000 Dosimeters (or Use of Commercial Vendor Service)

Items #3 and 4

None

(a) For industry implementation of Item #1 of the SIR, only 10 plants are presumed to be affected.

TABLE 1. (contd)

7. Per-Plant Industry Cost for SIR Implementation (I):

Item #1

Dosimeter Readout System	\$100K
5000 Dosimeters (\$20 each) or Use of Commercial Vendor Service	100K
I =	\$200K/plant

Items #3 and #4

None

8. Total Industry Cost for SIR Implementation (NI):

Item #1 = (10)(\$2.0E+5) = \$2.0E+6

Item #3 = 0

Item #4 = 0

NI = \$2.0E+6

9. Per-Plant Industry Labor for SIR Operation and Maintenance:

Costs are estimated directly in the next step.

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_o):

Item #1

I_o = \$2,000/py (see Section 3.1)

Item #3

I_o = \$634/py (see Section 3.3)

Item #4

I_o = \$634/py (see Section 3.3)

11. Total Industry Cost for SIR Operation and Maintenance ($N\bar{T}I_o$):

Item #1 = (134)(28.3)(\$2,000) = \$7.6E+6

Item #3 = (134)(28.3)(\$5,970) = \$2.2E+7

Item #4 = (134)(28.3)(\$634) = \$2.4E+6

$N\bar{T}I_o$ = \$3.26E+7

TABLE 1. (contd)

12. Total Industry Cost (S_I):

	Best Estimate		
Item #1	$9.6E+6		
Item #3	$2.3E+7		
Item #4	$2.4E+6	Upper Bound	Lower Bound
Total	$3.5E+7	$5.1E+7	$1.8E+7

NRC COSTS (Steps 13 through 21)

13. NRC Resources for SIR Development:

Item #1

Labor = 0.2 man-yr

Items #3 and 4

Costs are estimated directly in the next step.

4. Total NRC Cost for SIR Oevelopment (C_D):

Item #1 = (0.2 man-yr) ($1.0E+5/man-yr) =	$2.0E+4
Item #3	
Field Trial of ANSI N43.17-D4 ($4.0E+5)	$4.2E+5
Rule making ($2.0E+4)	
Item #4 =	$2.0E+4
C_D =	$4.6E+5

15- Steps Related to NRC Costs for Support of SIR Implementation and
20 Review of SIR Operation and Maintenance:

It is assumed that no additional NRC effort will be required to support SIR implementation or review SIR operation/maintenance. Thus, C and C_0 are both zero.

21. Total NRC Cost (S_N):

	Best Estimate		
Item #1	$2.0E+4		
Item #3	$4.2E+5		
Item #4	$2.0E+4	Upper Bound	Lower Bound
Total	$4.6E+5	$6.9E+5	$2.3E+5

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16. ABSTRACT (200 words or less)

This is the third in a series of reports to document the use of a methodology developed by the Pacific Northwest Laboratory to calculate, for prioritization purposes, the risk, dose and cost impacts of implementing resolutions to reactor safety issues (NUREG/CR-2800, Andrews et al. 1983). This report contains results of issue-specific analyses for 31 issues. Each issue was considered within the constraints of available information, as of Summer 1983, and two staff-weeks of labor. The results are referenced, as one consideration in setting priorities for reactor safety issues, in NUREG-0933, A Prioritization of Generic Safety Issues.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

In Situ Testing of Valves	Operating Procedures	Deficiency Reporting
Seismic Scram	Control Room Design	Design & Construction
Reactor Vessel	Support Personnel	Emergency Preparedness
Pipe Rupture	Quality Assurance	Radiation Effects
Pumps and Valves	Small Break LOCAS	Radiation Protection
Sabotage	Instrumentation & Controls	
Operating Personnel	Degraded Core Activities	

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