PNL-4297 Supp 1-

NUREG/CR-2800
PNL-4297
Supplement 1

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, R. G. Clark, W. E. Bickford, S. H. Bian, E. H. Carbaugh, B. A. Fecht, C. H. Henager, M. A. McKinnon, T. B. Powers

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

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ABSTRACT

This is the second 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. This report contains results of issue-specific analyses for 15 issues. Each issue was considered within the contraints of available information as of September 1982 and two staff-weeks of labor. The results will be referenced, as one consideration in setting priorities for reactor safety issues, in an NRC prioritization report to be published at a future date.

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 will be documented in an NRC prioritization report to be published at a future date.

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

Several minor differences may exist between assumptions used in PNL issue reports and those used in NUREG-0933. These arise primarily from changes in projected plant construction and cost bases. The effect on final results is small and has a negligible effect, on the utility of this information for safety issue prioritization.

The following is a listing 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

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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 15 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 will be documented in an NRC prioritization report to be published at a future date.

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 that could be completed within a several man-week effort. This will allow NRC to consider current and future prioritization criteria.

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

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 representative PWR and one representative BWR for all current and future plants. Risks for any particular plant could vary significantly from those of the representative plants, although these plants are believed to reasonably represent the industry as a whole.

Other major simplifications include the use of only dominant accident sequences. These sequences typically contribute 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

(a) Operated by Battelle Memorial Institute.

independent. When an initial ranking has been completed, additional analyses can be performed to identify interdependences.

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:

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

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

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

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

ΔW = $\Delta(FR)$ = 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

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

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; and 2) the utility's cost of design, procurement, installation, and testing to implement the requirement and its cost for O/M. Accident-avoidance results in cost savings to the utility. Information on both NRC and industry costs is considered since both represent costs that are paid by the public, either as taxpayers or ratepayers. Only future costs are relevant to current decisions, so sunk costs are ignored. All costs are considered to be 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:

$(S_N)_{Total}$ = 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 annual 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 implementaiton ($\$10^6$/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. et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Informtion Development, NUREG/CR-2800, PNL-4297. Pacific Northwest Laboratory, Richland, Washington.

2.0 ISSUE ANALYSES

Fifteen issue analyses are describe 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 the 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 the 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: 23, Reactor Coolant Pump Seal Failures

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

This issue is concerned with the rather high rate of failures of reactor coolant pump seals in PWRs. These seal failures, if serious enough, can create a small loss of coolant accident. The proposed resolution is to replace each pump seal annually, typically during a refueling outage.

AFFECTED PLANTS			
	BWR:	Operating = 0	Planned = 0
	PWR:	Operating = 47	Planned = 43

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	2.3E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	6.7E+4
Total of Above =	6.7E+4
Accident-Avoidance =	190

COST RESULTS ($$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	0.99
SIR Operation/Maintenance =	-390
Total of Above =	-390
Accident-Avoidance =	15
NRC COSTS:	
SIR Development =	0.036
SIR Implementation Support =	0.21
SIR Operation/Maintenance Review =	0.60
Total of Above =	0.85

REACTOR COOLANT PUMP SEAL FAILURES

ISSUE 23

1.0 SAFETY ISSUE DESCRIPTION

This issue is concerned with the high rate of failures of reactor coolant pump seals in pressurized water reactors. Such an event can create a small loss of coolant accident (LOCA).

Following are descriptions of typical reactor coolant pump seals (Makay and Adams 1979).

- BYRON-JACKSON supplies primary coolant pumps for the C-E and B&W reactor systems in the USA. For a B&W system they introduced a three-stage, mechanical-type face seal equally staged, while for a C-E system they supply the pumps with four seal stages. Three stages are equally staged and the fourth stage is used as a vapor seal at the top of the arrangement. The rotating face is titanium carbide, while the stationary face is carbon in both cases.

- BINGHAM originally had only two stages in their mechanical-type seal cartridge in both BWR and PWR applications. All their currently operating pump seals in PWRs were modified and now have three stages equally staged. The rotating face is tungsten carbide with carbon stationary faces. There is a restriction bushing on the top for vapor sealing.

- KSB uses both the hydrostatic and the hydrodynamic face seal types. In U.S. applications, with the exception of Forked River, KSB uses three unequally staged hydrodynamic face seals. The first two are equally staged, while the third seal takes only 16% staging pressure. The main seal leakage is 3.9 GPM, while the maximum backup seal leakage is 2.6 GPH, i.e., 320 oz/hr, which is 6.4 times the optimistic early U.S. predictions made before measurements were available. Minimum startup pressure is 200 psig. Each seal stage is supposed to with-stand full system pressure in case the others fail. The face materials are:

    1. Hydrostatic - Both rotating and stationary faces are made of chrome oxide.

    2. Hydrodynamic - Rotating face is carbon, while the stationary is tungsten carbide.

- WESTINGHOUSE - Westinghouse uses a three-stage seal design. The first seal stage is a tapered-land hydrostatic seal that takes the full system pressure, reducing the pressure from 2250 psi to 50 psi, at a maximum leakage rate of 5 GPM. The second stage is a mechanical seal designed to take full system pressure in case of first-stage failure. However, during normal operation, it is subjected to only 50 psi pressure breakdown. During normal operation, the No. 2 seal reduces pressure from 50 psi to not more than 5 psi, at a leakage rate of approximately 2 GPH. When there is an indication of a No. 1 seal failure, the No. 1 seal leakoff line is closed by a remotely-controlled air-operated valve. In this condition, the No. 2 seal operates as a hydrostatic seal with an estimated maximum leakage rate of 30 GPM. The pump should not be operated in this mode any longer than is absolutely necessary because a failure in No. 2 could result in gross leakage of reactor coolant from the pump. Westinghouse recommends not operating a pump for more than 30 minutes in this condition, which is long enough to ramp down reactor power level and turn off the pump. The No. 3 seal is a vapor seal and operates at a pressure of not more than 5 psi. The face materials are:

1. First stage - Both faces are made of aluminum oxide.

2. Second stage - Aluminum- oxide stationary ring with carbon rotating ring.

3. Third Stage - The faces were originally tungsten carbide at San Onofre No. 1, but they were changed in 1967 to aluminum oxide face with graphitar rotating face ring. The stainless steel ring holder was changed to a ceramic material for equal thermal expansion. Also, a bellows was added to provide spring pressure to the stationary face ring.

PROPOSED RESOLUTIONS

Potential solutions to the seal failure problem include improved pump design, improved seal design and more frequent seal replacement. The proposed resolution, used in order to provide what is believed to be an upper bound cost estimate, is more frequent seal replacement. This is assumed to be as frequent as each refueling outage which is typically an annual event.

AFFECTED PLANTS

This issue affects all 90 PWRs, 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:

   Reactor Coolant Pump Seal Failures (23)

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

   All PWRs

	N	$\bar{T}$ (yr)
Backfit PWRs:	47	27.7
Forward-fit PWRs:	43	30
All PWR Plants:	90	28.8

3. Plants Selected for Analysis:

   Arkansas Nuclear One-Unit 1 (ANO-1) is selected as the representative PWR.

4. Parameters Affected by SIR:

   The parameter identified from the ANO-1 Interim Reliability Evaluation Program (IREP) analysis which is affected by the proposed resolution is given as follows (Kolb et al. 1982):

Symbol	Description
B(1.2)	Reactor coolant pump seal rupture or small-small LOCA (D.38" < D < 1.2")

5. Base-Case Values for Affected Parameters:

   B(1.2) = 0.02/py

   This number comes from the ANO-1 IREP analysis.

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies:

Sequence		Base-Case Frequency (1/py)
$B(1.2)D_1$ -	α (PWR-1)	2.8E-10
	γ (PWR-2)	1.4E-6
	β (PWR-5)	2.0E-8
	ϵ (PWR-7)	1.4E-6
$B(1.2)D_1C$ -	α (PWR-1)	4.4E-10
	γ (PWR-2)	2.2E-6
	β (PWR-4)	3.1E-8
	ϵ (PWR-6)	2.2E-6

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 7.2E-10/py

PWR-2 = 3.6E-6/py

PWR-4 = 3.1E-8/py

PWR-5 = 2.0E-8/py

PWR-6 = 2.2E-6/py

PWR-7 = 1.4E-6/py

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

$\bar{F}$ =7.25E-6/py

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

W = 1.77E+1 man-rem/py

10. Adjusted-Case, Affected Values for Affected Parameters:

It is assumed for this study that the base-case frequency of B(1.2) is reduced by a factor of 2 for the adjusted case. This factor of 2 was chosen through consultation with PNL staff in which it was felt to be a reasonable reduction based on the selection of an acceptable issue resolution.

B(1.2) = 0.01/py

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Sequence		Adjusted-Case Frequency (1/py)
$B(1.2)D_1$ -	α (PWR-1)	1.4E-10
	γ (PWR-2)	7.0E-7
	β (PWR-5)	1.0E-8
	ϵ (PWR-7)	7.0E-7
$B(1.2)D_1^C$ -	α (PWR-1)	2.2E-10
	γ (PWR-2)	1.1E-6
	β (PWR-4)	1.5E-8
	ϵ (PWR-6)	1.1E-6

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 3.6E-10/py

PWR-2 = 1.8E-6/py

PWR-4 = 1.5E-8/py

PWR-5 = 1.0E-8/py

PWR-6 = 1.1E-6/py

PWR-7 = 7.0E-7/py

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

$\bar{F}^* = 3.6E-6/py$

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

$W^* = 8.9$ man-rem/py

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

3.6E-6/py

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

8.8 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.3E+4	1.4E+6	0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Reactor Coolant Pump Seal Failures (23)

2. Affected Plants (N): All PWRs

   Backfit PWRs: 47

   Forward-fit PWRs: 43

   90

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

   The average remaining life for all PWRs is 28.8 years.

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

   (19,900 man-rem)(3.6E-6/py) = 7.2E-2 man-rem/py

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

   (90)(28.8 yr)(7.2E-2 man-rem/py) = 1.9E+2 man-rem

   Upper bound = 2.2E+3 man-rem

   Lower bound = 0

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

   These steps are not applicable since SIR implementation involves policy and procedural decisions and no actual occupational dose.

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

   Dose estimated directly in next step

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

   Based on information in EPRI-NP-1138 (Clark and Barrow 1979) the average exposure for one pump seal replacement is 7 man-rems. Based on a review of plant design data, the average number of pumps per PWR (backfit and forward-fit) is estimated to be 3.7. Thus, for annual replacement:

   D_o = (7 man-rem/py)(3.7 pumps/plant) = 25.9 man-rem/py

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

   (90)(28.8 yr)(25.9 man-rem/py) = 6.7E+4 man-rem

TABLE 2. (contd)

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
6.7E+4	2.0E+5	2.2E+4

3.D SAFETY ISSUE COSTS

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

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Reactor Coolant Pump Seal Failures (23)

2. Affected Plants (N):

   All PWRs

Backfit PWRs:	47
Forward-fit PWRs:	43
	90

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

   The average remaining life for forward-fit PWRs is 30 years, for backfit PWRs is 27.7 years, and for all PWRs is 28.8 years.

Industry Costs (Steps 4 through 12)

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

   (\$1.65E+9)(3.6E-6/py) = \$5.9E+3

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

   (90)(28.8 yr)(\$5.9E+3/py) = \$1.5E+7

   Upper bound = \$1.9E+8

   Lower bound = 0

TABLE 3. (contd)

6. Per-Plant Industry Resources for SIR Implementation:

Labor - 2 man-wk/plant (administrative)

Replacement Power - none

Equipment - none

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

Labor for any PWR = (2 man-wk/plant)($2270/man-wk) = $4,540/plant license amendment for backfit PWRs only (due to change in technical specifications: assume a Class IV fee as per 10 CFR 170.22) = $12,300/plant

I (backfit) = $16,840/plant

I (forward-fit) = $ 4,540/plant

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

($16,840/backfit plant)(47) + ($4,540/forward-fit plant) (43) = $9.9E+5

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

Labor - 28 man-wk/py[a]

Replacement power - none

Equipment (seals) - $57,000/pump

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

Labor	= (28 man-wk/py)($2270/man-wk)	= $63,560/py
Equipment (seals)	= ($57,000/pump-yr)(3.7 pumps/plant)	= $210,900/py
	Total	= $274,460/py

(a) Assuming 300 man-hr/pump seal for annual replacement (EPRI-NP-1138) and 3.7 pumps/PWR gives:

(300 man-hr/pump-yr)(3.7 pumps/plant)/(40 man-hr/man-wk) = 28 man-wk/py.

TABLE 3. (contd)

11. Total Industry Cost for SIR Operation and Maintenance ($\bar{NTI}_0$):

This high cost for operation and maintenance is due primarily to the cost of the seals. It is recognized that with improved installation training and procedures and improved maintenance procedures that the seals would not need to be replaced as often because of a lower failure frequency. This would reduce costs for operation and maintenance substantially. Thus the cost estimates given here are believed to be an upper bound.

Also if seal failures were reduced, the industry could benefit from the reduced outage time. Based on information in a memo from R. Riggs to E. Adensom on December 9, 1980, the overall failure frequency for seals (major and minor requiring shutdown) is calculated to be 2.8E-1/py. If this failure frequency were reduced by a factor of 2, an average of 10 days per outage is assumed (McKay and Adams 1979, p. 5-18), and $300,000 per day for outage cost, then the industry could realize a cost savings of $1.1E+9.

($274,460/py)(90)(28.8 yr) -$1.1E+9 = -$3.9E+8

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
-$3.9E+8	$1.9E+8	$5.9E+8

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Generic issue resolution ≈ 16 man-wk (NRC staff labor)

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

(16 man-wk)($2270/man-wk) = $3.6E+4

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

1 man-wk/plant

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

(1 man-wk/plant)($2270/man-wk) = $2270/plant

TABLE 3. (contd)

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

    ($2270/plant)(90) = $2.1+5

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

    0.1 man-wk/py

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

    (0.1 man-wk/py)($2270/man-wk) = $2.3+2/py

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

    ($2.3E+2/py)(90)(28.8 yr) = $6.0E+5

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$8.5E+5	$5.3E+5	$1.2E+6

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Makay, E. and M. L. Adams. 1979. Operation and Design Evaluation of Main Coolant Pumps for PWR. EPRI-NP-1194. Prepared for Electric Power Research Institute, Palo Alto, California, by Energy Research and Consultants Corporation, Morrisville, Pennsylvania.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: B-6, Loads, Load Combinations, Stress Limits

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

Nuclear power plants are currently designed to withstand scenarios which include safe shutdown earthquakes (SSEs) and loss-of-coolant accidents (LOCAs), double-ended pipe break and asymmetric blowdowns in PWRs. Due to recent research showing the probability of these events to be small, a reevaluation of the combined load requirements for commercial nuclear power plants suggests the following SIRs: 1) decoupling the SSE and LOCA load requirements reducing the number of snubbers required, 2) removing pipe whip restraints in connection with the leak-before-break philosophy and 3) eliminating the need to design for asymmetric blowdown in all forward-fit PWRs.

AFFECTED PLANTS:	BWR:	Operating = 16	Planned = 20
	PWR:	Operating = 41	Planned = 43

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	4.0E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	6.8E+4
SIR Operation/Maintenance =	-1.1E+6
Total of Above =	-9.8E+5
Accident-Avoidance =	340

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	-920
SIR Operation/Maintenance =	-240
Total of Above =	-1200
Accident-Avoidance =	28
NRC COSTS:	
SIR Development =	0.10
SIR Implementation Support =	3.4
SIR Operation/Maintenance Review =	0
Total of Above =	3.5

LOADS, LOAD COMBINATIONS AND STRESS LIMITS

ISSUE B-6

1.0 ISSUE DESCRIPTION

The Code of Federal Regulations requires that structures, systems and components important to the safety of nuclear power plants in the United States be designed to withstand appropriate combinations of effects of natural phenomena coupled with the effects of normal and accident conditions (10 CFR 50, Appendix A). An example load combination requirement mandated for commercial nuclear power plants includes coupling the effects of safe shutdown earthquakes (SSEs) with a large loss-of-coolant accident (LOCA). In a recent evaluation, these combined loads were increased to further account for phenomena such as asymmetric blowdowns in PWRs and the better understanding of seismic hazards and probabilities.

Because these changes have raised questions with regard to implementation of new regulations, increased construction costs and reduced reliability of stiffer systems under normal operating transients, design requirements are being reevaluated. Several investigations have been undertaken, one such study being the Load Combination Program at LLNL where the objective has been to estimate the probability that a large LOCA and an earthquake will occur simultaneously. Several conclusions as a result of such investigations were included in a memorandum to H. R. Denton, "Research Information Letter No. 117," and are included here.

> "The following are believed to be the significant conclusions which may and should have near-term impact on licensing:
>
> 1. It is concluded from results 3 and 4 that for reasonable and representative conditions relating to fatigue crack growth in primary system piping, through-wall cracks are about a million times more likely to occur than double-ended guillotine breaks. This appears to offer substantial quantitative support in a probabilistic format for the leak-before-break hypothesis. This estimate may be less sensitive to input assumptions than other results since it is the ratio of two related computations of probabilities.
>
> 2. Fatigue crack growth due to all transients, including earthquakes, is an extremely unlikely mechanism for inducing large LOCA. The contribution of earthquakes to the occurrence of this unlikely event is a few percent of the total probability. Thus, fatigue-induced large LOCAs are very remote events, and earthquake-induced large LOCAs by fatigue are even more so.

3. An upper bound estimate of the probability of asymmetric blowdown loads (resulting from rupture of in-cavity piping) due to direct and indirect mechanisms is 10^{-4} over the 40-year plant life, the primary contribution to this estimate being indirect seismically-induced asymmetric blowdown. It is felt that the best estimate of the probability is several orders of magnitude lower. It is believed that additional study of indirect seismically-induced asymmetric blowdown has the potential for reducing the upper bound because of the very limited number of scenarios leading to asymmetric blowdown."

Proposed Safety Issue Resolution

The SIR for issue B-6 has three parts: removal of some snubbers, removal of pipe whip restraints and deletion of asymmetric blowdown analyses. In this analysis, all three are assumed to be implemented.

If the SSE-LOCA load requirements were decoupled, many plants would require reanalysis to determine which snubbers could be removed. Following implementation of this portion of the SIR (i.e., removal of appropriate snubbers), the advantage would be the elimination of inspection and maintenance on these systems. In addition, systems that have been previously designed to withstand extreme load conditions may have a reduced probability of failure under normal transient conditions due to the reduction in stiffness.

The probability of a leak occurring during a 40-year plant life is on the order the 10^{-6} considering only fatigue crack growth. Assuming the leak-before-break scenario, the second part of the SIR would suggest that pipe whip restraints be removed. Following initial removal, general plant access would be greatly improved, particularly during in-service inspections (ISIs), where pipe whip restraints must often be removed and subsequently replaced to gain access to systems under inspection or maintenance.

The final portion of the SIR deletes the required design analysis for asymmetric blowdown loads. This would affect only forward-fit plants and would eliminate the additional stiffening of the reactor pressure vessel.

This issue affects both PWRs and BWRs. The probability of pipe fracture in the primary coolant loop has been determined by LLNL on a representative PWR plant only. BWRs are assumed similar for this analysis. This assumption may need revision if additional studies for BWRs are completed (Lee 1981).

Oconee (PWR) and Grand Gulf (BWR) are assumed representative of all affected plants in this issue.

2.0 SAFETY ISSUE RISK AND DOSE

This section presents results of public risk and occupational dose calculations.

2.1 PUBLIC RISK REDUCTION

It is assumed that there will be a small amount of risk reduction to the public due to the removal of appropriate snubbers in systems designed to withstand SSE + LOCA. This reduction in system stiffeners should help preclude potential lockup of snubbers during normal operation transients, thus reducing large stresses on piping under normal operating conditions. Table 1 summarizes the results of this analysis.

2.2 OCCUPATIONAL DOSE

Additional radiation will be accrued by personnel during removal of snubbers and pipe restraints. The exposure, however, during operation and maintenance is reduced because removed systems will no longer require inspection and maintenance, and other systems will be more accessible during ISIs. Table 2 summarizes the results of this analysis.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Loads, Load Combinations, Stress Limits (B-6)

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

   All PWRs and BWRs built since 1972. (Design for SSE + LOCA and pipe whip has been mandated for approximately 10 years)

	N	$\bar{T}$ (yr)
PWR	84	29.0
BWR	36	28.8

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf 1 - representative BWR

TABLE 1. (contd)

4. Parameters Affected by SIR:

Symbol	Description
Oconee:	
S_1:	Rupture of reactor coolant system piping >10" but $\leq$13.5"
S_2	Rupture of RCS piping with diameter >4" but $\leq$10"
S_3	Rupture of RCS piping with diameter $\leq$4"
Grand Gulf:	
S	Small LOCA (rupture area <1 ft^2)

The analysis for large LOCA was performed in the Grand Gulf RSSMAP study, and did not fall into the dominant accident sequences.

5. Base-Case Values for Affected Parameters:

Original values are used as specified in Appendices A and B (Andrews 1983).

6. Affected Accident Sequences and Base-Case Frequencies:

Accident Sequence

PWR:

$S_3H - \gamma$ (PWR-3)
$S_3H - \beta$ (PWR-5)
$S_3H - \varepsilon$ (PWR-7)

$S_1D - \alpha$ (PWR-1)
$S_1D - \gamma$ (PWR-3)
$S_1D - \beta$ (PWR-5)
$S_1D - \varepsilon$ (PWR-7)

$S_3FH - \gamma$ (PWR-2)
$S_3FH - \beta$ (PWR-4)
$S_3FH - \varepsilon$ (PWR-6)

TABLE 1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies:

Accident Sequence

PWR (contd):

$S_2FH - \alpha$ (PWR-1)

$S_2FH - \beta$ (PWR-4)

$S_2FH - \varepsilon$ (PWR-6)

$S_2D - \alpha$ (PWR-1)

$S_2D - \gamma$ (PWR-3)

$S_2D - \beta$ (PWR-5)

$S_2D - \varepsilon$ (PWR-7)

$S_3D - \gamma$ (PWR-3)

$S_3D - \beta$ (PWR-5)

$S_3D - \varepsilon$ (PWR-7)

BWR:

$SI - \alpha$ (BWR-1)

$SI - \delta$ (BWR-2)

Original frequencies are used as specified in Appendices A and B (Andrews 1983).

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 1.00E-7/py

PWR-2 = 2.10E-6/py

PWR-3 = 7.40E-6/py

PWR-4 = 4.05E-8/py

PWR-5 = 1.47E-7/py

PWR-6 = 3.10E-6/py

PWR-7 = 1.27E-5/py

BWR-1 = 4.60E-8/py

BWR-2 = 4.60E-6/py

TABLE 1. (contd)

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

$\bar{F}_{PWR}$ = 2.56E-5/py

$\bar{F}_{BWR}$ = 4.65E-6/py

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

W_{PWR} = 51 man-rem/py

W_{BWR} = 33 man-rem/py

10. Adjusted-Case, Affected Values for Affected Parameters:

It has been suggested that removing snubbers associated with a potentially stiffer system in the event of a combined LOCA plus SSE would reduce the stiffness and potential lockup of snubbers during normal operation. As a result, this could reduce the probability of pipe rupture during normal operating transients (e.g., start up, thermal transients, etc.). A best estimate is that probability of pipe rupture may be reduced by 25% across the board. Adjusted frequencies are given below:

	Element	Adjusted Frequency
PWR:		
	S_1	7.5E-5/py
	S_2	3.0E-4/py
	S_3	9.8E-4/py
BWR:		
	S	1.05E-3/py

11. Affected Accident Sequences and Adjusted-Case Frequencies:

	Sequence	Frequency (1/py)
PWR:		
	S_3H-γ	3.2E-6
	S_3H-β	5.5E-8
	S_3H-ϵ	3.8E-6

TABLE 1. (contd)

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

	Sequence	Frequency (1/py)
PWR:		
	S_1D-α	5.0E-8
	S_1D-γ	9.8E-7
	S_1D-β	3.7E-8
	S_1D-ε	4.0E-6
	S_3FH-γ	1.6E-6
	S_3FH-β	2.3E-8
	S_3FH-ε	1.6E-6
	S_2FH-α	9.8E-9
	S_2FH-β	7.1E-9
	S_2FH-ε	7.5E-7
	S_2D-α	1.5E-8
	S_2D-γ	3.0E-7
	S_2D-β	1.1E-8
	S_2D-ε	1.2E-6
	S_3D-γ	5.2E-7
	S_3D-β	7.5E-9
	S_3D-ε	5.2E-7
BWR:		
	SI-α	3.4E-8
	SI-δ	3.4E-6

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 7.5E-8/py

PWR-2 = 1.6E-6/py

PWR-3 = 5.6E-6/py

TABLE 1. (contd)

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

PWR-4 = 3.0E-8/py

PWR-5 = 1.1E-7/py

PWR-6 = 2.3E-6/py

PWR-7 = 9.5E-6/py

BWR-1 = 3.4E-8/py

BWR-2 = 3.4E-6/py

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

$\bar{F}^*_{PWR}$=1.9E-5/py

$\bar{F}^*_{BWR}$=3.5E-6/py

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

W^*_{PWR}:38 man-rem/py

W^*_{BWR}:25 man-rem/py

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

$\Delta\bar{F}_{PWR}$=6.4E-6/py

$\Delta\bar{F}_{BWR}$=1.2E-6/py

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

ΔW_{PWR}=13 man-rem/py

ΔW_{BWR}=8.2 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
4.0E+4	4.8E+6	0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Loads, Load Combinations, Stress Limits (B-6)

2. Affected Plants (N):

   All PWRs and BWRs built since 1972

	N
PWR backfit	41
PWR forward-fit	43
BWR backfit	16
BWR forward-fit	20

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

	$\bar{T}$(yr)
PWR backfit	28.0
PWR forward-fit	30.0
Avg. for 84 plants	29.0
BWR backfit	27.4
BWR forward-fit	30.0
Avg. for 36 plants	28.8

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

   PWR:

   $$\Delta\bar{F}D_R = (6.4E\text{-}6/py)(19{,}900 \text{ man-rem}) = 0.13 \text{ man-rem/py}$$

   BWR:

   $$\Delta\bar{F}D_R = (1.2E\text{-}6/py)(19{,}900 \text{ man-rem}) = 0.024 \text{ man-rem/py}$$

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
340	8000	0

TABLE 2. (contd)

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

Implementation of SIR would require removal of the portion of the pipe snubbers associated with decoupling LOCA and SSE as well as removing unnecessary pipe whip restraints to follow the leak-before-break concept. The following is a best estimate of labor hours required as obtained through PNL contacts with reactor personnel.

a. Removal of snubbers associated with decoupling LOCA and SSE.

# of snubbers in representative plant: (Landers 1981)

PWR ~800

BWR ~950

# of snubbers to be removed (50%):

PWR = 400

BWR = 475

Time to remove average snubber:

6 man-hr/snubber

Time to remove snubbers in representative plant:

PWR: 2400 man-hr/plant

BWR: 2850 man-hr/plant

b. Removal of pipe whip restraints (those which interfere with ISIs or general plant access).

# of pipe whip restraints in representative plant:

PWR ~100

BWR ~140

# of restraints to be removed (50%):

PWR = 50

BWR = 70

Time to remove average restraint:

40 man-hr/rest

TABLE 2. (contd)

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

Time to remove restraints in representative plant:

PWR: 2000 man-hr/plant

BWR: 2800 man-hr/plant

Addition of a and b: PWR: 4400 man-hr/plant (backfit plants only)

BWR: 5650 man-hr/plant (backfit plants only)

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

It will be assumed that radiation fields of 0.25 R/hr are encountered. (Landers 1981)

D_{PWR}: (4400 man-hr/plant) (0.25 R/hr) = 1100 man-rem/plant

D_{BWR}: (5650 man-hr/plant) (0.25 R/hr) = 1410 man-rem/plant

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

ND = (41)(1100 man-rem/plant) + (16)(1410 man-rem/plant)

= 6.8E+4 man-rem

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

Assume a decrease in labor hours due to a decrease in the number of snubbers to be inspected and maintained. In addition, there will be access to pumps, valves, etc., due to removal of pipe whip restraints.

a. Assume standard snubber inspection can be done at a rate of 4 snubbers/man-hr and done on the average (considering accessible and inaccessible snubbers) of 3 times/year. It is still assumed that a representative population of accessible snubbers is inspected every month.

PWR = 300 man-hr saved/yr

BWR = 356 man-hr saved/yr

b. Assume periodic testing, maintenance, removal and replacement of potentially defective snubbers. It is assumed 5% of the total snubbers are replaced per year. The following labor time is saved when removing 50% of the snubbers.

TABLE 2. (contd)

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

PWR = (20 snubbers)(20 man-hr/plant) = 400 man-hr saved/plant

BWR = (24 snubbers)(20 man-hr/plant) = 480 man-hr saved/plant

c. Assume the time required to remove pipe whip restraints is saved each time an inspection is made on the system. In addition, the time required for pump and valve inspection as well as general plant access is decreased.

Assume that the time saved is 6 times that required to remove the restraint (from Step 6):

PWR = 6(2000 man-hr/plant)/(29.0 yr) = 414 man-hr saved/plant

BWR = 6(2800 man-hr/plant)/(28.8 yr) = 583 man-hr saved/plant

Total labor hours in radiation zones for maintenance and operation of SIR:

PWR = -1110 man-hr/py
(negative sign indicates reduction)
BWR = -1420 man-hr/py

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

$(D_O)_{PWR}$ = (-1110 man-hr/py)(0.25 R/hr) = -278 man-rem/py

$(D_O)_{BWR}$ = (-1420 man-hr/py)(0.25 R/hr) = -355 man-rem/py

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

$N\bar{T}D_O$ = (84)(29.0 yr)(-278 man-rem/py) + (36)(28.8 yr)(-355 man-rem/py)
= -1.05E+6 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
-9.8E+5	-3.3E+5	-2.9E+6

(negative sign indicates decrease)

3.0 SAFETY ISSUE COSTS

Results of NRC and industry cost calculations are included in this section.

Best estimates were used for labor time required for removal of snubbers and restraints as well as time saved in later inspection and maintenance procedures. Additional estimates were made with regard to the number of forward-fit plants that would have to be totally or partially redesigned. Table 3 summarizes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Loads, Load Combinations, Stress Limits (B-6)

2. Affected Plants (N):

   All PWRs and BWRs built since 1972

	N
PWR operating	41
PWR planned	43
BWR operating	16
BWR planned	20

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

	$\bar{T}$ (yr)
PWR operating	28.0
PWR planned	30.0
Avg for 84 plants	29.0
BWR operations	27.4
BWR planned	30.0
Avg. for 36 plants	28.8

TABLE 3. (contd)

Industry Costs (Steps 4 through 12)

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

$$\Delta(\bar{F}A)_{PWR} = (6.4E\text{-}6/py)(\$1.65E+9) = \$1.1E+4/py$$

$$\Delta(\bar{F}A)_{BWR} = (1.2E\text{-}6/py)(\$1.65E+9) = \$2.0E+3/py$$

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

Best Estimate	Upper Bound	Lower Bound
\$2.8E+7	\$6.7E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

The labor estimates in this section are based on confirmatory analyses performed at PNL and in conjunction with reviews prior to the granting of operating licenses.

For backfit plants:

	PWR	BWR
Labor:		
Analysis	250 man-wk/plant	250 man-wk/plant
Crafts and Services	110 man-wk/plant	141 man-wk/plant
Replacement power:	None	None
(Assume work done during scheduled outages)		
Equipment:	None	None
Totals	360 man-wk/plant	391 man-wk/plant

All forward-fit plants will experience an implementation cost savings since they will not have to install as many snubbers and pipe whip restraints as a result of the SIR. From step 6 of the Occupational Dose Work Sheet, it can be assumed that the following numbers of snubbers and restraints will NOT have to be installed at forward-fit plants:

	Snubbers	Restraints
PWR	400	50
BWR	475	70

Assuming the average time to install a snubber or restraint is at least equivalent to that for removal (6 man-hr/snub and 40 man-hr/rest), the amounts of installation labor that will be saved at all forward-fit plants are:

TABLE 3. (contd)

6. Per-Plant Industry Resources for SIR Implementation: (contd)

Labor Saved (man-hr/plant):	PWR	BWR
Snubbers	2400	2850
Restraints	2000	2800
Total	4400	5650

However, ~50% of all forward-fit plants are assumed to require redesign for the reduced number of snubbers and restraints. While this requires no physical removal of these snubbers and restraints, staff labor will be required to perform a reanalysis. The estimate for backfit plants, 250 man-wk/plant, is assumed applicable for reanalysis due to the SIR at half of the forward-fit plants. The SIR will be included in the initial analysis at the remaining half of the forward-fit plants at no additional cost.

Equipment costs will also be saved by not installing these snubbers and restraints. These cost savings are estimated in the next step. Like backfit plants, no additional down-time requiring replacement power will result at forward-fit plants due to SIR implementation.

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

For backfit plants:

I_{PWR} = (360 man-wk/plant)($2270/man-wk) = $8.2E+5/plant

I_{BWR} = (391 man-wk/plant)($2270/man-wk) = $8.9E+5/plant

For forward-fit plants:

Cost savings due to not installing snubbers and restraints (all 63 forward-fit plants):

- Equipment[a]

PWR = (400 snubs)($10,000/snub) + (50 rests/plant)($10,000/rest)
= $4.5E+6/plant

BWR = (475 snubs/plant)($10,000/snub) + (70 rests/plant)
($10,000/rest) = $5.2E+6/plant

(a) Based on industry contacts, PNL found equipment costs for snubbers and restraints could vary widely. An average value of a $10,000 per snubber or restraint is assumed for this analysis.

TABLE 3. (contd)

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

- Labor

  PWR = (4400 man-hr/plant)($2270/man-hr) = $1.0E+7/plant

  BWR = (5650 man-hr/plant)($2270/man-hr) = $1.3E+7/plant

- Total

  PWR = $1.4E+7/plant (cost savings)

  BWR = $1.8E+7/plant (cost savings)

Reanalysis cost (22 forward-fit PWRs and 10 forward-fit BWRs):

Labor = (250 man-wk/plant)($2270/man-wk) = $5.7E+5/plant

For 22 forward-fit PWRs requiring reanalysis:

I = $5.7E+5/plant - $1.4E+7/plant = -$1.4E+7/plant

For 10 forward-fit BWRs requiring reanalysis:

I = $5.7E+5/plant - $1.8E+7/plant = -$1.7E+7/plant

For remaining 21 forward-fit PWRs and 10 forward-fit BWRs (not requiring reanalysis):

I_{PWR} = -$1.4E+7/plant

I_{BWR} = -$1.8E+7/plant

(Note--negative signs indicate reductions. Also, all forward-fit costs include redesign to eliminate asymmetric blowdown loads; the cost of additional concrete and labor to strengthen the vessel is not included.)

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

PWRs:	backfit	= 41($8.2E+5)	= $3.3E+7
	forward-fit	= 22 (-$1.4E+7) + 21 (-$1.4E+7)	= -$6.1E+8
BWRs:	backfit	= 16 ($8.9E+5)	= $1.4E+7
	forward-fit	= 10 (-$1.8E+7) + 10 (-$1.8E+7)	= -$3.6E+8
		NI	= -$9.2E+8

TABLE 3. (contd)

9. Per-Plant Industry Labor for SIR Operation and Maintenance:
(refer to step 9 in Occupational Dose Work Sheet)

PWR: -1110 man-hr/py
BWR: -1420 man-hr/py
(negative sign indicates reduction)

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

$$(I_O)_{PWR} = (-1110 \text{ man-hr/py})(1 \text{ man-wk}/40 \text{ man-hr})(\$2270/\text{man-wk}) = -\$6.3E+4/\text{py}$$

$$(I_O)_{BWR} = (-1420 \text{ man-hr/py})(1 \text{ man-wk}/40 \text{ man-hr})(\$2270/\text{man-wk}) = -\$8.1E+4/\text{py}$$

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

$$N\bar{T}I_O = (84)(29.0 \text{ yr})(-\$6.3E+4/\text{py}) + (36)(28.8 \text{ yr})(-\$8.1E+4/\text{py}) = -\$2.4E+8$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
-\$1.2E+9	-\$6.8E+8	-\$1.6E+9

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Development of the generic issue resolution is estimated to require one man-yr of NRC staff labor.

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

$$C_D = (1 \text{ man-yr})(\$1.0E+5/\text{man-yr}) = \$1.0E+5$$

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

backfit plants = 15 man-wk/plant
forward-fit plants = 10 man-wk/plant

TABLE 3. (contd)

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

backfit plants: C = (15 man-wk/plant)($2270/man-wk)
= $3.4E+4/plant

forward-fit plants: C = (10 man-wk/plant)($2270/man-wk)
= $2.3E+4/plant

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

NC = (57)($3.4E+4/plant) + (63)($2.3E+4/plant) = $3.4E+6

18-20. Steps Related to NRC Cost to Review SIR Operation and Maintenance:

No change over current NRC inspection requirements is assumed to result for SIR operation and maintenance. Therefore, $C_O = 0$.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$3.5E+6	$5.2E+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.

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Landers, D. F. et al. 1981. Effects of Postulated Event Devices on Normal Operation of Piping Systems in Nuclear Power Plants. NUREG/CR-2130. Teledyne Engineering Services, Los Angeles, California.

Lee, S., R. D. Streit and C. K. Chou. 1981. Probability of Pipe Fracture in the Primary Coolant Loop of PWR Plant. NUREG/CR-2189. Vol. 1. Lawrence Livermore National Laboratory, Livermore, California.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: B-10, Behavior of BWR Mark III Containments

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

Mark III suppression pool dynamic loads from a LOCA have been found to be larger than first postulated, requiring some modifications to piping, piping supports, grating, floors, equipment location and containment vessel stiffeners. Structural fixes made or planned by Grand Gulf as a result of NRC's proposed acceptance criteria are used as the issue resolution for evaluation purposes.

AFFECTED PLANTS			
	BWR:	Operating = 0	Planned = 8
	PWR:	Operating = 0	Planned = 0

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 2.6E+4

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	130
Total of Above =	130
Accident-Avoidance =	72

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:

SIR Implementation =	93
SIR Operation/Maintenance =	5.5
Total of Above =	98
Accident-Avoidance =	5.9

NRC COSTS:

SIR Development =	0
SIR Implementation Support =	0.22
SIR Operation/Maintenance Review =	1.1
Total of Above =	1.3

BEHAVIOR OF BWR MARK III CONTAINMENTS

ISSUE B-10

1.0 SAFETY ISSUE DESCRIPTION

The problem description for issue B-10 given in NUREG-0471 (1978) is as follows:

> "This is an ACRS generic concern. Evaluation and approval is required of various aspects of the Mark III containment design which differ from the previously reviewed Mark I and Mark II designs. This task involves the completion of the staff evaluation of the Mark III containment and documentation of the method used to validate the analytical models and assumptions needed to predict the containment pressures in the event of a LOCA."

Mark III suppression pool dynamic loads were reviewed by the NRC at the construction permit (CP) stage for Grand Gulf 1 and 2 and at the preliminary design analysis stage for GESSAR-238NI. It was concluded at the time that the information available was sufficient to adequately define the pool dynamic loads for those nuclear plants under review for CPs. Since the issuance of the GESSAR-238NI Safety Evaluation Report in December 1975, GE has conducted further tests and analyses to confirm and refine the original load definitions. To keep the NRC and Mark III applicants apprised of the current status of these tests, GE issued an Interim Containment Loads Report (22A4365) in April of 1978 and revised this report several times before GESSAR-II was provided to the NRC staff in March of 1980. GESSAR-II is GE's final design analysis submittal for their standard balance of plant (BOP) design and is to be referenced by Mark III OL applicants. Appendix 3B of GESSAR-II provides the finalized pool dynamic load definition for Mark III containments and is the basic document used for review by the NRC staff and its consultants.

The NRC staff is currently reviewing GE's pool dynamic load definitions to arrive at a finalized hydrodynamic load definition that can be utilized by Mark III containment applicants for operating licenses. The pool dynamic loads are being reviewed under USI A-39, "Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits for BWR Containment." The end product of these two generic programs will be applicable to Grand Gulf.

Loss of Coolant Accident Pool Dynamics:

Following a postulated loss of coolant accident (LOCA), escaping steam forces the weir annulus portion of the suppression pool out of the drywell into the wetwell. This action results in pool swell and loads from vent clearing jets, chugging, impact of water, impact from froth impingement, pool fallback, condensation loads and containment pressure (Speis 1982).

Resolution of This Safety Issue Involves Two Actions:

1. Review of the affected Mark III plants which are planned and under construction to determine if the structures as designed meet the NRC Acceptance Criteria for Mark III LOCA-related pool dynamic loads (or NUREG) when issued.

2. Implementing structural fixes where necessary to resist the dynamic loads.

The structural fixes required to resist the LOCA-related pool dynamic loads at Grand Gulf 1 and 2 were selected as typical fixes for the generic issue and evaluated for risk, dose and cost. These structural fixes were as follows.

1. Deleted solid circumferential concrete floor at elevation 120 feet and added a steel grating catwalk at the same elevation; due to pool swell, relocated equipment to above elevation 135 feet

2. Relocated and strengthened main steam tunnel floor above pool swell zone (~5 feet)

3. Added suppression pool makeup system

4. TIP station floor projected down into suppression pool to eliminate pool swell loads

5. Relocated piping to the region above bulk pool swell

6. Changed piping submerged in pool to smaller sizes and heavier walls to accommodate submerged structure loads.

Some equipment modifications were made as summarized below (abbreviated descriptions from a 9/24/81 Mississippi Power and Light Co. presentation slide).

- Polar crane
- Aftercooler for purge compressors
- CRD hydraulic system modifications
- 107 valve operators modified
- 723 pipe supports modified
- 236 pipe supports added
- Stiffened polar crane rail support brackets
- Some minor modifications to floor steel.

The above described fixes were selected as typical of modifications likely to be made in any future plants since Grand Gulf is nearest completion.

There were other modifications made in other Mark III containments, due to changes in Generic Mark III pool dynamic load criteria. These were made in 4 plants selected for examination: Clinton 1 (OL Stage, 80% complete); River Bend (OL Stage, 35% complete); Black Fox (CP stage); Allen's Creek (CP stage).

The changes made in these designs (from a presentation chart by Mel Fields, Containment Systems Branch, Nuclear Regulatory Commission) were as follows:

<u>Clinton 1 and 2</u>

- Suppression pool liner strengthened.
- General modification of hydraulic control unit floor, equipment moved from grating onto concrete, piping raised.
- SRV piping and supports modified, emergency core cooling systems (ECCS) suction strainers and supports redesigned.
- Polar crane girders and brackets redesigned.
- General upgrading of piping and pipe supports.

<u>River Bend</u>

- Steel hoops and stiffeners added to outside of free-standing steel containment, up to the elevation of the suppression pool surface.
- Will fill the annulus between the concrete shield building and steel containment with concrete to a level 5 feet above suppression pool surface.

<u>Black Fox 1 and 2</u>

- Modified stud patterns on weir wall
- May add stiffeners to free-standing steel containment
- Will fill the annulus between the concrete shield building and steel containment up to a level of 25 feet above suppression pool bottom.

<u>Allen's Creek</u>

- Added vertical stiffeners to outside of free-standing steel containment in the suppression pool region.
- Modified dome design from ellipsodial to hemispherical.
- Relocated all piping out of solid impact area.

The changes made at these four sites differ substantially from Grand Gulf in that three of them planned to add stiffeners to the free-standing steel containment shell or otherwise stiffen the shell to resist the vibratory loading of the LOCA-caused dynamic pool loads.

Various other options for fixes may also exist. For the purposes of quantifying the effects of B-10 with a single set of fixes, the Grand Gulf pipe support and floor modification fixes were selected as the representative resolution.

2.0 SAFETY ISSUE RISK AND DOSE

Results of public risk reduction and occupational dose analyses are summarized in this section.

2.1 PUBLIC RISK REDUCTION

The proposed resolution of Generic Safety Issue, B-10, "BWR Mark III Containments," is the implementation of the structural fixes discussed in the previous section. The applicable plants include all GE BWR-6 plants with Mark III containments, beginning with Grand Gulf 1. There are eight BWR-6 plants listed as under construction in Appendix C of PNL-4297 (Andrews 1982). These plants are all forward-fit, thus fixes will be made before plant start-up. The Grand Gulf plant is selected as representative. It is a GE BWR-6 with Mark III containment typical of the generic plant in question.

The parameters in the plant risk equations assumed to be affected by the BWR Mark III containment modifications are related to the ECCS. The LOCA is taken to have already occurred, i.e., the dynamic loads are a result of the LOCA. To have any adverse effect, the suppression pool swell resulting from the LOCA must attain a height sufficient to cause dislodgement of piping, equipment, or walkways. These must fall back into the suppression pool and subsequently plug suction piping for the ECCS (RCICS, LPCSS, LPCIS, HPCSS, SPMS--see PNL-4297).

Base-Case Redefinition

Six parameters are identified from Appendix B (Andrews 1982) as related to loss of flow through various ECCS feeding from the suppression pool: L, LA2, LB2, LC, SA, and SB. None of these, as originally defined, incorporated loss of flow due to the effects of pool swell (debris blockage or pipe crimping, e.g.). Thus, to account for this possibility, a common-cause factor X is added to each parameter to represent the system unavailability due to pool swell effects. Each parameter is, therefore, redefined as follows:

$$L = L_o + X$$
$$LA2 = LA2_o + X$$
$$LB2 = LB2_o + X$$

$LC = LC_o + X$
$SA = SA_o + X$
$SB = SB_o + X$

where the terms with the "o" subscripts represent the original parameters.

All minimal cut sets containing the terms L, LA2, LB2, LC, SA, or SB are modified by replacing each term by the above redefinition. In effect, this adds a new minimal cut set for each replacement, as follows:

Original Cut Set

$S \cdot LA2 \cdot VGB2$

Substitution

$S \cdot (LA2_o + X) \cdot VGB2 = (S \cdot LA2_o \cdot VGB2) + (S \cdot X \cdot VGB2)$

New Cut Set

$S \cdot X \cdot VGB2$

(Note, since $LA2_o$ is the original value for LA2, the cut set [$S \cdot LA2_o \cdot VGB2$] is the original cut set)

Even if two parameters are replaced in one cut set, only one new cut set results:

$$S \cdot SA \cdot SB = S \cdot (SA_o+X) \cdot (SB_o + X)$$

$$= \underbrace{(S \cdot SA_o \cdot SB_o)}_{\text{original}} + \underbrace{(S \cdot X)}_{\text{new}}$$

This is a consequence of X being a common-cause factor.

This replacement results in adding only one new minimal cut set to the following sequences:

Affected Sequence	Affected Cut Set
$T_1PQI - \alpha,\delta$	$T_1 \cdot P \cdot LOPNRE \cdot LOPNRL \cdot X \cdot RECOVERY$
$T_{23}PQI - \alpha,\delta$	$T_{23} \cdot P \cdot Q1 \cdot X \cdot RECOVERY$
$SI - \alpha,\delta$	$S \cdot X$

For the following affected sequences, several new, affected minimal cut sets are added: $T_1PQE - \gamma,\delta$ and $T_{23}PQE - \gamma,\delta$.

A previously non-dominant accident sequence for a large LOCA, as originally given in the Grand Gulf RSSMAP study (Hatch 1981), is presumed to be affected since it, too, would contain the redefined parameters discussed above. The original sequence is AI-α,δ (assumed to have the same minimal cut sets as SI-α,δ, except for the different initiator). Its affected cut set is $(A \cdot X)$.

Affected Parameter Values

Since X was not considered in the original study, a base-case probability must be estimated for it. This is done by defining X as follows:

$$X = X_1 \cdot X_2 \cdot X_3$$

where X_1 = Pool swell, given a LOCA, reaches height sufficient to potentially dislodge equipment

X_2 = Equipment is dislodged by pool swell and falls back into pool

X_3 = Dislodged debris in pool somehow causes loss of flow through ECCS suction lines (e.g., by plugging or line crimping).

Likelihoods of each event are estimated for both a small and large LOCA initiator. The results are as follows:

LOCA Initiator	X_1	X_2	X_3	X
A (large)	0.9	0.5	0.05	0.02
S (small)	0.5	0.3	0.03	0.005

Thus, given a large (small) LOCA, X takes on a base-case value of 0.02 (0.005).

For the adjusted case, the following is assumed.

- After the LOCA, because of structural fixes, no damage occurs to the ECCS piping. Therefore, the probability of common-cause loss of flow due to pool swell (X) is essentially zero.

Table 1 summarizes results of the public risk reduction calculations.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Behavior of BWR Mark III Containments (B-10)

TABLE 1. (contd)

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

Eight forward-fit BWRs are assumed to be affected ($N = 8$, $\bar{T} = 30$ years)

3. Plants Selected for Analysis:

Grand Gulf 1 - Representative BWR

4. Parameters Affected by SIR:

A parameter X, a common-cause factor representing the probability of pool-swell-induced flow blockage, is incorporated into small and large LOCA sequences. See the explanation in Section 2.1, Public Risk Reduction.

5. Base-Case Values for Affected Parameters:

X = 0.02 given a large LOCA (A)

X = 0.005 given a small LOCA (S)

6. Affected Accident Sequences and Base-Case Frequencies:

Small LOCA:	$T_1PQI - \alpha(BWR\text{-}1)$	= 4.6E-9/py
	$T_1PQI - \delta(BWR\text{-}2)$	= 4.6E-7/py
	$T_{23}PQI - \alpha(BWR\text{-}1)$	= 5.6E-8/py
	$T_{23}PQI - \delta(BWR\text{-}2)$	= 5.6E-6/py
	$T_1PQE - \gamma(BWR\text{-}3)$	= 5.6E-9/py
	$T_1PQE - \delta(BWR\text{-}4)$	= 5.6E-9/py
	$T_{23}PQE - \gamma(BWR\text{-}3)$	= 4.5E-9/py
	$T_{23}PQE - \delta(BWR\text{-}4)$	= 4.5E-9/py
	$SI - \alpha(BWR\text{-}1)$	= 7.0E-8/py
	$SI - \delta(BWR\text{-}2)$	= 7.0E-6/py
Large LOCA:	$AI - \alpha(BWR\text{-}1)$	= 2.0E-8/py
	$AI - \delta(BWR\text{-}2)$	= 2.0E-6/py

(Note: A has an original value of 1E-4/py)

TABLE 1. (contd)

7. Affected Release Categories and Base-Case Frequencies:

    BWR-1 = 1.5E-7/py

    BWR-2 = 1.5E-5/py

    BWR-3 = 1.0E-8/py

    BWR-4 = 1.0E-B/py

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

    $\bar{F}$ = 1.5E-5/py

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

    W = 110 man-rem/py

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

    The SIR will presumably eliminate the potential for pool-swell-induced flow blockage. Therefore, X ≈ 0 for both large and small LOCAs. Consequently, the adjusted-case affected accident sequences, release categories, core-melt frequency and public risk will be ≈0.

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

    $\Delta\bar{F}$ = 1.5E-5/py

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

    ΔW = 110 man-rem/py

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

Best Estimate (man-rem)	Upper Bound (man-rem)	Lower Bound (man-rem)
2.6E+4	7.9E+5	0

2.2 OCCUPATIONAL DOSE

Table 2 summarizes the results for occupational dose calculations.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Behavior of BWR Mark III Containments (B-10)

2. Affected Plants (N):

   8 BWRs, all forward-fit

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

   30 years

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

   $\Delta(\bar{F}D_R)$ = (1.5E-5/py)(19,900 man-rem) = 0.30 man-rem/py

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

Best Estimate (man-rem)	Upper Bound (man-rem)	Lower Bound (man-rem)
72	430	0

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

   Since SIR implementation involves only forward-fit plants prior to their operation, no dose will be accumulated (D = 0).

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

   A utilization factor of 75% is assumed.

Repainting gratings	= 2 man-day/yr @ 75% =	12 man-hr/py
Inspecting 236 pipe supports	= 2 man-day/yr @ 75% =	12 man-hr/py
Misc. equipment inspections	= 2 man-day/yr @ 75% =	12 man-hr/py
		36 man-hr/py

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

   The typical dose rate above the suppression pool at Grand Gulf is 15 mR/hr.

   D_o = (0.015 R/hr)(36 man-hr/py) = 0.54 man-rem/py

TABLE 2. (contd)

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

$N\bar{T}D_0$ = (8)(30 yr)(0.54 man-rem/py) = 130 man-rem.

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Upper Bound (man-rem)	Lower Bound (man-rem)
130	390	43

3.0 SAFETY ISSUE COSTS

The results of NRC and industry cost calculations are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Behavior of BWR Mark III Containments (B-10)

2. Affected Plants (N):

   8 BWRs, all forward-fit

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

   30 years

Industry Costs (Steps 4 through 12)

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

   $\Delta(\bar{F}A)$ = (\$1.65E+9)(1.5E-5/py) = \$2.5E+4

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

Best Estimate	Upper Bound	Lower Bound
\$5.9E+6	\$3.6E+7	0

6. Per-Plant Industry Resources for SIR Implementation:

   Costs estimated directly in next step.

TABLE 3. (contd)

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

Fix (Structure)	Equip. Cost	Labor Cost	Total
Replace catwalk	$ 16,000	$ 10,000	$ 56,000
New supports	10,000	20,000	
Relocate equip.	10,000	50,000	60,000
Strengthen tunnel floor	20,000	40,000	60,000
Add suppression pool makeup system	200,000	200,000	400,000
TIP sta. floor mods	30,000	30,000	60,000
Relocate piping	10,000	50,000	60,000
Smaller piping	50,000	150,000	200,000
723 pipe supports (mod)	300,000	450,000	750,000
236 pipe supports (new)	200,000	300,000	500,000
Stiffer crane brackets	40,000	60,000	100,000
Floor steel mods	20,000	30,000	50,000
Equip. Mods			
Polar crane	40,000	60,000	100,000
Aftercooler	20,000	30,000	50,000
CRD hyd. system	20,000	30,000	50,000
107 valve operators (mod)	40,000	60,000	100,000
Total (per plant)	$1.03E+6	$1.57E+6	$2.60E+6

Of the 8 BWRs affected, 5 were originally scheduled to commence operation prior to 1984, the remainder between 1984-6. The first 5, being near completion, would experience a maximum construction delay of 2 months/plant due to structural fixes and equipment modifications. The remaining three should show minimal (1 week/plant), if any, such delay. An average delay of 30 days/plant is assumed, requiring replacement power at $3.0E+5/day, or a total of $9.0E+6/plant.

I = $2.6E+6/plant (equip. and labor) + $9.0E+6/plant (repl. power)
= $1.16E+7/plant.

TABLE 3. (contd)

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

$NI = (8)(\$1.16E+7/plant) = \$9.28E+7$

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

10 man-wk/py for aftercooler and CRD hydraulic system mods

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

$I_O = (10 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$22,700/\text{py}$

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

$N\bar{T}I_O = (8)(30 \text{ yr})(\$22,700/\text{py}) = \$5.5E+6$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$9.8E+7	\$1.4E+8	\$5.2E+7

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

For this issue, resolution is primarily on a plant-specific basis; therefore, no NRC resources are foreseen to develop a generic SIR.

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

Zero

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

12 man-wk/plant

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

$C = (12 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$27,200/\text{plant}$

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

$NC = (8)(\$27,200/\text{plant}) = \$2.18E+5$

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

2 man-wk/py

TABLE 3. (contd)

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

$$C_O = (2\ \text{man-wk/py})(\$2270/\text{man-wk}) = \$4540/\text{py}$$

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

$$N\bar{T}C_O = (8)(30\ \text{yr})(\$4540/\text{py}) = \$1.09E+6$$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$1.3E+6	$1.9E+6	$7.5E+5

REFERENCES

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

Hatch, S., et al. 1981. RSSMAP: Grand Gulf No. 1 BWR Power Plant, NUREG/CR-1659/4. Sandia National Laboratories, Albuquerque, New Mexico.

Speis, T. 1982. "Containment Systems Branch Input to the Safety Evaluation Report, Grand Gulf Nuclear Station, Units 1 and 2, Docket Nos. 50-416/417," attachment to March 25, 1982 memorandum from T. Speis to R. Tedesco, U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: B-26, Structural Integrity of Containment Penetrations

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

Containment penetrations must be accessible to ensure that inservice examination requirements as specified in the ASME Code can be completed. Issue B-26 calls for an evaluation to determine accessibility of high-energy, fluid system penetrations in operating plants as well as in plants under construction and up for licensing reviews. In the event that penetration designs are found inadequate with respect to accessibility for conducting current inservice inspections, alternative surveillance or analysis methods would be implemented to ensure that inspections can be completed.

AFFECTED PLANTS

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 370

OCCUPATIONAL DOSES:

SIR Implementation =	710
SIR Operation/Maintenance =	-4600
Total of Above =	-3900
Accident-Avoidance =	0

COST RESULTS ($$10^6$)

INDUSTRY COSTS:

SIR Implementation =	1.5
SIR Operation/Maintenance =	3.5
Total of Above =	5.0
Accident-Avoidance =	0

NRC COSTS:

SIR Development =	0.81
SIR Implementation Support =	0
SIR Operation/Maintenance Review =	0
Total of Above =	0.81

STRUCTURAL INTEGRITY OF CONTAINMENT PENETRATIONS

ISSUE B-26

1.0 SAFETY ISSUE DESCRIPTION

"Containment penetration assemblies provide a means to maintain the integrity of the containment pressure boundary and prevent overstressing of the penetration nozzle due to thermal stresses. A typical penetration assembly may consist of a flued head, a guard pipe, an expansion bellows and an impingement ring. The flued head may be fabricated from a forging which may be welded into the process line or onto the outer surface of the process piping. This task involves an evaluation to assess the adequacy of specific containment penetration designs from the point of view of structural integrity and inservice inspection requirements." (U.S. NRC 1978)

Issue B-26 requires a review of specific containment penetration designs. The specific penetrations under investigation include only the high-energy fluid systems (from personal communication with M. Hum, Materials Engineering Branch, US NRC). High-energy fluid systems are defined as those that are in operation or are pressurized during normal plant conditions (i.e., during reactor startup, power operation and cold shutdown, but excluding test modes) where either or both of the following are satisfied (Regulatory Guide 1.46):

a. Maximum temperature exceeds 200°F
b. Maximum pressure exceeds 275 psig.

Under Issue B-26 it shall be determined whether or not the "configuration and accessibility of the welds in the proposed design and the procedures proposed for performing volumetric examination will permit the inservice examination requirements of Section XI of the ASME Code to be met (U.S. NRC 1978)". In the event that penetration designs are found to be inadequate with respect to accessibility for conducting current inservice inspections, alternative surveillance or analysis methods would most likely be implemented to ensure that inspections can be completed. In some cases, minor modifications in the penetration configuration may be required.

The SIR for B-26 involves the development of new surveillance or analysis methods applicable to containment penetrations which are identified as inaccessible. The issue is applied to all BWRs and PWRs currently operating as well as those plants under construction and up for licensing review, which would encompass all forward-fit and back-fit plants.

2.0 SAFETY ISSUE RISK AND DOSE

Upon satisfactory resolution of inspectability concerns, this issue should not affect public risk. However, should it be impractical for a plant to assure the above stated inservice examination requirement in accordance with Standard Review Plan item 3.6.2, no specific guidance is provided as to what measures provide an acceptable resolution. In these cases, NRC staff approval, on a case-by-case review basis, may result in inconsistent penetration requirements from plant to plant. Such inconsistencies, should they occur, could result in increased risk to the public. To account for this possibility, the potential for pubic risk reduction is considered for the SIR of B-26 by assuming that the likelihood for radioactive release via containment leakage may be reduced. 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:

   Structural Integrity of Containment Penetrations (B-26)

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

	N	$\bar{T}$(yr)
PWR	90	28.8
BWR	44	27.4
	134	28.3

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf 1 - representative BWR

4. Parameters Affected by SIR:

   Oconee 3: Elements of the dominant minimal cut sets for the dominant accident sequences do not change. The containment failure mode β (containment leakage) is assumed to be affected by this SIR.

   Grand Gulf 1: Elements of the dominant minimal cut sets for the dominant accident sequences do not change. The containment failure mode β (containment leakage) is affected by this SIR but contributes only to non-dominant accident sequences. Analysis is performed directly from NUREG/CR-1659/4 (Hatch 1981).

TABLE 1. (contd)

5. Base-Case Values for Affected Parameters:

The original values of β for Oconee 3 and Grand Gulf are assumed for the base case.

6. Affected Accident Sequences and Base-Case Frequencies:

PWR - All accident sequences (dominant and non-dominant) contributing to release categories PWR-4 and 5 (as given in PNL-4297 [Andrews 1983]) are affected, with the original frequencies taken as those for the base case.

BWR - Affected sequences (non-dominant) taken directly from NUREG/CR-1659/4.

Sequence	Frequency (1/py)
AI - β	1.8E-9
AC - β	5.4E-12
SI - β	3.2E-8
SC - β	7.7E-11
SDI - β	2.1E-12
T_1PQI - β	1.1E-8
T_{23}PQI - β	2.6E-8
T_1QW - β	4.3E-8
T_1C - β	8.4E-10
T_1QUW - β	2.4E-10
T_{23}C - β	3.8E-8
T_{23}QW - β	8.4E-8
T_{23}QUW - β	4.9E-10

All accident sequences contribute to release category BWR-4 only.

7. Affected Release Categories and Base-Case Frequencies:

PWR - Original frequencies for PWR-4 and 5 are taken as the base-case values

BWR - From above, BWR-4 = 2.4E-7/py.

TABLE 1. (contd)

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

The minimal cut set values are not affected in this analysis. Thus, this step is not required.

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

PWR: W = (9.7E-8/py)(2.7E+6 man-rem) + (4.6E-7/py)(1.0E+6 man-rem)
= 7.2E-1 man-rem/py

BWR: W = (2.4E-7/py)(6.1E+5 man-rem) = 1.5E-1 man-rem/py

10. Adjusted-Case Values for Affected Parameters:

For SIR it is assumed that all penetration assembly designs meet code accessibility requirements or approved analysis/surveillance techniques. The result is adequate completion of inservice inspections as well as elimination of unresolved conditions affecting plant start up.

a. Number of penetrations per plant: An average of 40 high-energy penetrations/plant are assumed in the following analysis. This number will vary depending on plant type and design, and is only an estimate based on information available in Section 3.6.2 of several BWR and PWR FSARS (including tables of high-energy lines, identification of systems requiring boundary guard pipes and complete listings of penetration data).

b. Number of penetrations considered in analysis: It is further assumed that only 20% of all high-energy penetrations/plant need attention as specified by Issue B-26. Since requirements for inservice inspection are known, industry, where possible, attempts to build in inspectability features.

Number of penetrations in need of special investigation (i.e., new surveillance or analysis techniques):
(40 pent./plant)(0.20) = 8 pent./plant.

c. Penetrations requiring modification or analysis development: There are analysis and augmented inspection procedures currently available to accommodate many of the inaccessible penetrations. It is estimated that 20% of those penetrations under consideration may require the development of new analysis procedures.

TABLE 1. (contd)

Number of penetrations requiring new procedures:

(8 pent./plant)(0.20) = 1.6 pent./plant; assume 2 pent./plant for this analysis.

Of the 40 penetrations, it is assumed that these 2 penetrations/-plant would be 5 times more likely to fail than the remaining 38. Upon resolution of the issue all 40 penetrations have an equal failure probability. This results in a 17% reduction in the containment leakage probability. The adjusted value for the containment failure mode β, is therefore:

$$\beta = (0.83)(.007) = .006$$

11. Affected Accident Sequences and Adjusted-Case Frequencies:

PWR:

Sequence	Frequency (1/py) in Release Category PWR-4	PWR-5
$T_2MLU - \beta$		7.2E-9
$T_1MLU - \beta$		1.2E-8
$T_1(B_3)MLU - \beta$		1.3E-8
$T_2MQH - \beta$		6.6E-8
$S_3H - \beta$		6.0E-8
$S_1D - \beta$		4.0E-8
$T_2MQFH - \beta$	3.0E-8	
$S_3FH - \beta$	2.5E-8	
$S_2FH - \beta$	7.8E-9	
$T_2MLUO - \beta$		4.9E-8
$T_2KMU - \beta$		4.7E-8
$S_2D - \beta$		1.2E-8
$S_3D - \beta$		8.4E-9
$T_1MLUO - \beta$		3.2E-8
$T_3MLUO - \beta$		6.6E-9
$T_2MQD - \beta$		9.0E-9
non-dom.	1.6E-8	1.6E-8

TABLE 1. (contd)

BWR:

Sequence	Frequency (1/py) in Release Category BWR-4
AI - β	1.6E-9
AC - β	4.6E-12
SI - β	2.8E-8
SC - β	6.6E-11
SDI - β	1.8E-12
T_1PQI - β	9.6E-9
T_{23}PQI - β	2.2E-8
T_1QW - β	3.7E-8
T_1C - β	7.2E-10
T_1QUW - β	2.0E-10
T_{23}C - β	3.2E-8
T_{23}QW - β	7.2E-8
T_{23}QUW - β	4.2E-10

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-4 = 7.9E-8/py

PWR-5 = 3.0E-7/py

BWR-4 = 2.0E-7/py

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

Not applicable to this analysis.

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

PWR: W* = (7.9E-8/py)(2.7E+6 man-rem) + (3.8E-7/py)(1.0E+6 man-rem)
= 5.9E-1 man-rem/py

BWR: W* = (2.0E-7/py)(6.1E+5 man-rem) = 1.2E-1 man-rem/py

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

None

TABLE 1. (contd)

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

PWR: ΔW = 1.3E-1 man-rem/py

BWR: ΔW = 3.0E-2 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
370	6.1E+4	0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Structural Integrity of Containment Penetrations (B-26)

2. Affected Plants (N):

	N
PWR operating:	47
PWR planned:	43
BWR operating:	24
BWR planned:	20

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

	$\bar{T}$ (Yr)
PWR operating:	27.7
PWR planned:	30.0
BWR operating:	25.2
BWR planned:	30.0

4. Per-Plant Occupational Oose Reduction Due to Accident Avoidance (ΔFD_R):

None

TABLE 2. (contd)

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

None

(Error bounds not estimated.)

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

It is assumed that implementation of issue B-26 will involve new analysis procedures or surveillance techniques for the most part. It is further assumed that most of the labor time in radiation zones will occur during scheduled inservice inspections with no increase in exposure time. However, it is anticipated that 1 of the 2 penetrations per plant requiring new procedures (see Step 10, Public Risk Reduction Work Sheet) will require minor modification. It is assumed that this would require 40 man-hr in average radiation fields of 250 mR/hr.

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

$$D = (40 \text{ man-hr/plant})(0.25 \text{ R/hr}) = 10 \text{ man-rem/plant}$$

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

If one assumes that all forward-fit plant penetration problems are resolved before operating licenses are granted, only back-fit plants will require SIR implementation that will result in an occupational dose increase.

$$ND = (71 \text{ backfit plants})(10 \text{ man-rem/plant}) = 710 \text{ man-rem}$$

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

It is assumed that, prior to SIR, failure rates are as currently encountered--at most one failure/year in all operating plants. If no resolution occurs, this will apply to all plants. The labor involved might amount to 20 man-wks/failure.

$$\text{Labor} = (20 \text{ man-wk/failure})\left(\frac{1 \text{ failure/yr}}{134 \text{ plants}}\right) = 0.15 \text{ man-wk/py}$$

It is assumed that, subsequent to SIR, failure rates are 5 times lower than before. Labor hours for repair remain the same.

$$\text{Labor} = (20 \text{ man-wk/failure})(1/5)\left(\frac{1 \text{ failure/yr}}{134 \text{ plants}}\right) = 0.03 \text{ man-wk/py}$$

TABLE 2. (contd)

All inspection for maintenance and operation remains the same both before and after SIR.

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

Again assuming radiation fields of 250 R/hr.

D_o = (.03-.15 man-wk/py)(40 man-hrs/man-wk)(0.25 R/hr) = -1.2 man-rem/py

(Negative sign indicates dose reduction.)

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

Applied to all plants.

$N\bar{T}D_o$ = [90(28.8) + 44(27.4)](-1.2) = -4.56E+3 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
-3.9E+3	-1.3E+3	-1.2E+4

(Negative sign indicates dose reduction.)

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost calculations are included in this section. Best estimates were used to determine labor time required to analyze penetration assemblies. These were based on experience in areas of structural modeling, fracture mechanics modeling and incorporating nondestructive test procedures into industrial trials. These estimates are based solely on example alternatives which might be undertaken to resolve the issue. Table 3 summarizes the results of the cost analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Structural Integrity of Containment Penetrations (B-26)

TABLE 3. (contd)

2. Affected Plants (N):

	N
PWR operating:	47
PWR planned:	43
BWR operating:	24
BWR planned:	20

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

	$\bar{T}$ (yr)
PWR operating:	27.7
PWR planned:	30.0
BWR operating:	25.2
BWR planned:	30.0

Industry Costs (Steps 4 through 12)

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

None

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

None

(Error bounds not estimated.)

6. Per-Plant Industry Resources for SIR Implementation:

It is assumed that 2 man-months/plant of effort are required to develop the supporting analysis for a given plant. In addition, it is assumed that 2 penetrations per plant are involved and that one of the penetrations requires the modifications. The cost of penetration modification is assumed to be \$5000/pent. This is applicable to backfit plants only.

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

$$\text{Labor} = \left(\frac{2 \text{ man-mo-plant}}{12 \text{ man-mo/man-yr}}\right)(\$1.0E{+}5/\text{man-yr}) = \$1.67E{+}4/\text{plant}$$

$$\text{Equipment} = (1 \text{ pent./plant})(\$5000/\text{pent.}) = \underline{\$5000/\text{plant}}$$

$$I = \$2.17E{+}4/\text{plant}$$

TABLE 3. (contd)

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

Applicable to all backfit plants.

$$NI = (71 \text{ plants})(\$2.17E+4/\text{plant}) = \$1.54E+6$$

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

The assumption is that new inspection procedures or analysis and modification would require additional labor during every 10 year inspection period. Assuming that an additional 4 man-wk/plant are required, the time spent over and above current inspection time is:

$$(4 \text{ man-wk/plant})\left(\frac{3 \text{ inspection periods}}{30 \text{ years}}\right) = 0.4 \text{ man-wk/py}$$

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

$$I_0 = (0.4 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$908/\text{py}$$

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

Applicable to all plants.

$$N\bar{T}I_0 = [(90)(28.8) + 44(27.4)]\ (908) = \$3.45E+6$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$5.0E+6	\$6.9E+6	\$3.1E+6

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

The NRC cost to review the plant penetration design inspection procedure and to prepare a safety evaluation report is estimated at 5 man-wk/plant (backfit only).

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

$$C = (71 \text{ plants})\ (5 \text{ man-wk/plant})\ (\$2270/\text{man-wk}) = \$8.06E+5$$

TABLE 3. (contd)

15-20. No additional NRC labor above that currently expended is foreseen to result from SIR. Therefore, $C = C_0 = 0$.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$B.1E+5	$1.2E+6	$4.1E+5

REFERENCES

Andrews, W. 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. 1978. Generic Task Problem Descriptions--Category B, C and D Tasks. NUREG-0471. U.S. Nuclear Regulatory Commission, Washington, DC.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: B-55, Improved Reliability of Target Rock Safety Relief Valves

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

Target Rock safety relief valves are a feature of BWR pressure relief, automatic depressurization, and emergency core cooling systems. Because of the number of unanticipated events with the valves used in these systems, their reliability has been identified as a specific safety issue. Valve redesign and increased maintenance have improved their reliability. Further improvements are being sought with programs now underway.

AFFECTED PLANTS:			
	BWR:	Operating = 22	Planned = 9
	PWR:	Operating = 0	Planned = 0

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	2.6E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	180
SIR Operation/Maintenance =	1900
Total of Above =	2100
Accident-Avoidance =	77

COST RESULTS ($$10^6$)

INDUSTRY COSTS;	
SIR Implementation =	19
SIR Operation/Maintenance =	-94
Total of Above =	-75
Accident-Avoidance =	6.4
NRC COSTS;	
SIR Development =	0.05
SIR Implementation Support =	0.14
SIR Operation/Maintenance Review =	3.8
Total of Above =	4.0

IMPROVED RELIABILITY OF TARGET ROCK SAFETY RELIEF VALVES

ISSUE (B-55)

1.0 SAFETY ISSUE DESCRIPTION

The BWR pressure relief system is designed to limit reactor pressure during normal operational transients and to prevent overpressurization of the reactor coolant pressure boundary (RCPB) under the most severe abnormal operational transients (e.g., closure of the main steam line isolation valves or fast closure of the turbine stop valves at full power). These design functions are accomplished through the use of a plant-unique combination of safety valves (SVs), power-actuated relief valves (PARVs), and dual function safety relief valves (SRVs). The majority of the latter two valve types in BWRs are commonly referred to as Target Rock SRVs.

In addition to the RCPB overpressure protection design functions of the BWR pressure relief system, a specified number of the PARVs or SRVs utilized in the pressure relief system of each BWR facility are used in the automatic depressurization system (ADS), which is one of the emergency core cooling systems. In the event of certain postulated small-break, loss-of-coolant accidents (LOCA), the ADS is designed to reduce reactor coolant system pressure to permit the low pressure emergency core spray and/or low pressure coolant injection systems to function. The ADS performs this design function by automatically actuating certain preselected PARVs or SRVs following receipt of specific signals from the protection system.

Certain safety concerns result when (1) a valve fails to open properly on demand, (2) a valve opens spuriously and then fails to properly reseat, and (3) a valve opens properly but fails to properly reseat. The failure of a pressure relief system valve to open on demand results in a decrease in the total available pressure relieving capacity of the system. Spurious openings of pressure relief system valves or failures of valves to properly reseat after opening can result in inadvertent reactor coolant system blowdown with unnecessary thermal transients on the reactor vessel and the vessel internals, unnecessary hydrodynamic loading of the containment system's pressure suppression chamber (torus) and its internal components, and potential increases in the release of radioactivity to the environs. In addition, if the failed valve also serves as part of the ADS, a degradation of the capability of the ADS to perform its emergency core cooling function could result.

Approximately 160 reactor-years of operating experience have accumulated with a significant number of failures of the Target Rock valves occurring due to various causes.[a] Studies and testing of these valves by the Owners Group, in some cases at the suggestion of NRC, have resulted in design changes in the valves and the issuance of several formal generic installation, operating, and maintenance instructions.

In 1978 (US NRC 1978), it was concluded by NRC staff that the inadvertent blowdown events that have occurred to date as a result of pressure relief system valve malfunctions have neither significantly affected the structural integrity or capability of the reactor vessel, the reactor vessel internals, or the pressure-suppression containment system, nor resulted in any significant radiation releases to the environment. They concluded that such events, even if they were to occur at a more frequent rate than that indicated by operating experience, would not likely have any significant effects on the reactor vessel or the vessel internals. It was also concluded that pressure relief valve blowdown events will not result in offsite radiological consequences appreciably different from those encountered during a normal reactor shutdown.

With respect to the pressure-suppresssion containment system, the slow progressive nature of the material fatigue mode of failure associated with the dynamic loading conditions resulting from pressure relief valve blowdown events and the substantial fatigue life margin currently available in the affected structures have led the staff to conclude that additional short-term actions are not required to assure that the integrity and functional capability of the system will be maintained. In addition, current programs to provide additional containment system structural safety margins for the long-term (i.e., the anticipated lifetime of the BWR facilities) are acceptable. The performance of these valves, however, is under continuous surveillance and the consequences of their failures are subject to review.

PROPOSED RESOLUTION

The proposed resolution is to replace all the 3-stage Target Rock SRVs with 2-stage valves. This resolution will result in a reduction in the frequency of valves failing to reseat. This assumption is based on the continued success of the remedial programs currently underway for these valves at existing BWRs.

AFFECTED PLANTS

This issue resolution affects 22 operating BWRs and is assumed to affect 9 BWRs now under construction. The 22 operating BWRs utilize Target Rock 3-stage SRVs and the 9 BWRs, which are all more than 75% complete, are assumed to have already installed Target Rock 3-stage SRVs.

(a) One PWR (Beaver Valley 1) also employs three Target Rock SRVs on the primary system. However, since this issue is primarily concerned with Target Rock SRV performance in BWRs, the Beaver Valley PWR is not assumed to be affected.

2.0 SAFETY ISSUE RISK AND DOSE

2.1 PUBLIC RISK REDUCTION

In the analysis for potential public risk reduction, Grand Gulf 1 BWR risk parameters are used. It is assumed for this analysis that a final solution (negligible frequency of Target Rock valve malfunction) has not yet been achieved. Hence, failure rate data on these valves in existing reactors are applicable to this analysis. Reactors (BWR/6) with Mark III containments for which full operating licenses are pending will presumably not use Target Rock valves.

Analyses of the effects of malfunctioning valves as separate failures have indicated that, for the short term, public safety is not of concern. The resulting thermal transients, even at the current rate of these events, are not likely to create concerns over pressurized thermal shock. The potential for radioactive releases to the public following a malfunction resulting in an unplanned blowdown is no greater than for a normal shutdown. However, when valve failure to reseat occurs simultaneously with failures on other systems, some potential for a core-melt exists. Analysis of the dominant accident sequences at Grand Gulf 1 for these events was done as part of this report.

All minimal cut sets in the following four Grand Gulf accident sequences are affected:

T_1PQI

$T_{23}PQI$

T_1PQE

$T_{23}PQE$.

Results of the analysis for public risk reduction are summarized in Table 1.

2.2 OCCUPATIONAL DOSE

The results of the analysis for occupational dose are summarized in Table 2.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Improved Reliability of Target Rock SRVs (B-55)

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

22 operating BWRs and 9 BWRs under construction

	N	$\bar{T}$ (yr)
Backfit BWRs	22	26.2
Forward-fit BWRs	9	30
	31	27.3

The 9 BWRs now under construction are assumed to be involved in retrofitting procedures with the 3-stage Target Rock SRVs. It is assumed that modifications are required on the valves already installed in these plants.

3. Plants Selected for Analysis:

Representative BWR-Grand Gulf 1

4. Parameters Affected by SIR:

The issue involves malfunctioning of a specific type of relief valve in BWRs only, namely the Target Rock Valves. It is further assumed that the valve that fails to reseat is large enough to lead to a LOCA which, in combination with other events occurring simultaneously, can lead to a core-melt. The parameter P in the Grand Gulf risk equation is assumed to be affected.

5. Base-Case Values for Affected Parameters:

P is assumed to have its original value of 0.1 in the base case.

6. Affected Accident Sequences and Base-Case Frequencies:

The base-case frequencies are the original values.

$T_1PQI-(\alpha,\delta)$
$T_{23}PQI-(\alpha,\delta)$
$T_1PQE-(\gamma,\delta)$
$T_{23}PQE-(\gamma,\delta)$

TABLE 1. (contd)

7. Affected Release Categories and Base-Case Frequencies:

BWR-1 = 5.3E-8/py

BWR-2 = 5.3E-6/py

BWR-3 = 3.9E-7/py

BWR-4 = 3.9E-7/py

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

6.1E-6/py

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

40 man-rem/py

10. Adjusted-Case, Affected Values for Affected Parameters:

For this analysis, it is assumed that the issue resolution will result in a reduction in the frequency of valves failing to reseat by a factor of 4. Hence, P* = 0.025. This assumption is based on the continued success of the remedial programs currently underway for these valves at BWRs.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

$$T_1PQI- \begin{cases} \alpha(\text{BWR-1}) = 4.0\text{E-9/py} \\ \delta(\text{BWR-2}) = 4.0\text{E-7/py} \end{cases}$$

$$T_{23}PQI- \begin{cases} \alpha(\text{BWR-1}) = 9.3\text{E-9/py} \\ \delta(\text{BWR-2} = 9.3\text{E-7/py} \end{cases}$$

$$T_1PQE- \begin{cases} \gamma(\text{BWR-3}) = 3.0\text{E-8/py} \\ \delta(\text{BWR-4}) = 3.0\text{E-8/py} \end{cases}$$

$$T_{23}PQE- \begin{cases} \gamma(\text{BWR-3}) = 6.8\text{E-8/py} \\ \delta(\text{BWR-4}) = 6.8\text{E-8/py} \end{cases}$$

TABLE 1. (contd)

12. Affected Release Categories and Adjusted-Case Frequencies:

BWR-1 = 1.3E-8/py

BWR-2 = 1.3E-6/py

BWR-3 = 9.8E-8/py

BWR-4 = 9.8E-8/py

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

1.5E-6/py

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

9.9 man-rem/py

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

6.13E-6/py - 1.51E-6/py = 4.6E-6/py

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

40 man-rem/py - 9.9 man-rem/py = 30 man-rem/py

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

2.6E+4 man-rem

Upper bound = 1.0E+6 man-rem

Lower bound = 0

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Improved Reliability of Target Rock SRVs (B-55)

2. Affected Plants (N):

BWRs	31	
Backfit BWRs:	22	
Forward-fit BWRS:	9	(in final construction)

TABLE 2. (contd)

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

BWR: Backfit $\bar{T}$ = 26.2 yr
Forward-fit $\bar{T}$ = 30 yr
Average $\bar{T}$ = 27.3 yr

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

(4.6E-6/py)(19,860 man-rem) = 0.091 man-rem/py

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

7.7E+1 man rem

Upper bound = 6.2E+2 man-rem

Lower bound = 0

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

It is assumed (based on consultation with PNL staff) that the labor in radiation zones to modify 165 valves in 22 operating plants (an average of 8 SRVs per plant) will amount to 4 man-weeks per backfit plant.

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

The average dose rate around the SRVs during plant shutdown is estimated to be about 0.050 R/hr.

(4 man-wk/plant)(40 man-hr/man-wk)

(0.050 R/hr) = 8 man-rem/plant

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

(8 man-rem/plant)(22 plants) = 1.8E+2 man-rem

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

It is assumed that further modifications, adjustments, and maintenance will require approximately 45 man-hr/py (25% of original implementation labor). This labor for operation and maintenance will apply to backfit (22) and forward-fit (9) plants.

TABLE 2. (contd)

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

(45 man-hr/py)(0.050 R/hr) = 2.3 man-rem/py

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

(31 plants)(27.3 yr)(2.3 man-rem/py) = 1.9E+3 man-rem

12. Total Occupational Dose Increase (G):

2.1E+3 man-rem

Upper bound = 6.3E+3 man-rem

Lower bound = 7.0E+2 man-rem

3.0 SAFETY ISSUE COSTS

The industry and NRC costs associated with the SIR 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 Issues:

Improved Reliability of Target Rock SRVs (B-55)

2. Affected Plants(N)

31 BWRs

Backfit BWRs: 22

Forward-fit BWRs: 9 (in final construction)

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

BWR: Backfit $\bar{T}$ = 26.2 yr

Forward-fit $\bar{T}$ = 30 yr

Average $\bar{T}$ = 27.3 yr

TABLE 3. (contd)

Industry Costs (steps 4 through 12)

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

(4.6E-6/py)($1.65E+9) = $7.6E+3/py

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

$6.4E+6

Upper bound = $5.1E+7

Lower bound = 0

6. Per-Plant Industry Resources for SIR Implementation:

Modifying or refurbishing SRVs on high-temperature, high-pressure steam lines is expected to require engineering, prints, license review, testing, travel, labor (5 man-wk/plant), material, QA control and management review.

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

Resources	Cost ($/plant)
Engineering, Meetings, Travel	22K
Prints, Plans, Drawings	13K
Licensing, QA	20K
Management Review	5K
Labor (5 man-wks)	11K
Material	540K*
Total	611K

* Based on 266 valves in 31 plants (22 operating and 9 in final construction), using $60,000/valve @ an average of 9 valves/plant

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

(31 plants)($6.11E+5/plant) = $1.9E+7

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

It is estimated that 50 man-hr/py will be required for operation (testing) and maintenance. Furthermore, existing data on Target Rock SRV experience indicate that 53 blowdowns attributable to Target Rock SRV failures have occurred in 160 plant-years,[a] a blowdown rate of 53/160 = 0.33/py. Assuming each blowdown results in a 1.5-day outage, (0.33/py)(1.5 day) = 0.50 day/py of down-time is attributable to Target Rock SRV failures.

With a factor of four improvement in Target Rock SRV reliability, a blowdown rate of 0.33/4 = 0.083/py can be expected as a result of the SIR. This translates into (0.083/py)(1.5 day) = 0.12 day/py of down-time after SIR, a savings of 0.38 day/py of outage time.

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

Labor:

(50 man-hr/py)($2270/man-wk)/(40 man-hr/man-wk) = $2,840/py

Replacement Power:

(-0.38 day/py)($3E+5/day) = -$114,000/py

Total = -$111,160/py

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

(31 plants)(27.3 yr)(-$1.11E+5/py) = -$9.4E+7

12. Total Industry Cost (S_I):

-$7.5E+7

Upper bound = -$2.7E+7

Lower bound = -$1.2E+8

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development

It is assumed (based on consultation with PNL staff) that 0.5 man-year will be spent in development of the SIR.

(a) Five plant-years of Target Rock SRV experience at the Beaver Valley 1 PWR are included in this data base.

TABLE 3. (contd)

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

C_D = (0.5 man-year)($1.0E+5/man-year) = $5.0E+4

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

To support the implementation by the industry, 2 man-weeks per plant is assumed (based on consultation with PNL staff).

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

(2 man-wk/plant)(2,270/man-wk) = $4,540/plant

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

(31 plants)($4,540/plant) = $1.4E+5

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

A total of 2 man-wk/py of inspection time is estimated to follow-up on operating and maintenance, including testing.

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

(2 man-wk/py)($2,270/man-wk) = $4,540/py

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

(31 plants)(27.3 yr)($4,540/py) = $3.8E+6

21. Total NRC Cost (S_N):

$4.0E+6

Upper bound = $5.9E+6

Lower bound = $2.1E+6

REFERENCES

U.S. NRC. July 1978. Technical Report on Operating Experience with BWR Pressure Relief Systems. NUREG-0462. Division of Operating Reactors, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D. C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: B-58, Passive Mechanical Failure

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

Valves in nuclear plants fail by a variety of mechanisms. Valve failure data will be reviewed to determine the frequency of passive failure modes and mechanisms (e.g. corrosion) which render valves inoperable over a period of time. Additional studies will be reviewed to ultimately recommend solutions.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = <4000

OCCUPATIONAL DOSES:

SIR Implementation = 300

SIR Operation/Maintenance = -1100

Total of Above = -800

Accident-Avoidance = <60

COST RESULTS ($$10^6$)

INDUSTRY COSTS:

SIR Implementation = 19

SIR Operation/Maintenance = -320

Total of Above = -300

Accident-Avoidance = <5

NRC COSTS:

SIR Development = 0.05

SIR Implementation Support = 0

SIR Operation/Maintenance Review = 0

Total of Above = 0.05

PASSIVE MECHANICAL FAILURE

ISSUE (B-58)

1.0 SAFETY ISSUE DESCRIPTION

As stated in NUREG-0471 (US NRC, 1978), this task involves a review of valve failure data in a systematic manner to confirm the staff's present judgment regarding the likelihood of passive mechanical valve failures, categorize these and other valve failures as to expected frequency, specify acceptance criteria and determine if and how the results of this effort should be applied in licensing reviews.

This issue is related to a number of other issues dealing with valve reliability:

C-11	Assessment of Failure and Reliability of Pumps and Valves
II.D.2	Research on Relief and Safety Valve Test Requirements
II.E.6	In-Situ Testing of Valves

Safety Issue C-11 in particular is aimed at active failure of pumps and valves. The approach used in analyzing C-11 will be applicable here. Valve failure data collected at the Nuclear Safety Information Center were studied to identify failure frequency for active failure mechanisms (NUREG/CR-0848; Scott and Gallaher, 1979). These same data are examined here to identify passive failure mechanisms. Passive failure was assumed to be due primarily to corrosion which caused deterioration over a period of time. The distinction made between active and passive failure is that passive failure occurs before valve demand, whereas active failure occurs at the moment of demand (e.g., sudden fatigue) or while in use. The actual data are given in Attachment 2. It is assumed that a 50% reduction in such passive failures can be achieved, resulting in a ~6% reduction in hardware-related valve failure.

PROPOSED RESOLUTION

Issue B-58, as written, calls primarily for the review of data on valve failures. Risk reduction would require that resulting conclusions and recommendations be acted upon to reduce passive failure modes. It is assumed here that this task will ultimately lead to the implementation of some hardware modifications. This would require an extension of the scope beyond that originally stated for Issue B-58.

AFFECTED PLANTS

This issue affects all PWRs and BWRs, both complete and under construction.

2.0 SAFETY ISSUE RISK AND DOSE

Public risk reduction and occupational dose are discussed in this section.

2.1 PUBLIC RISK REDUCTION

The data on operating experience of nuclear valves from 1965 to 1978 (NUREG/CR-0848) are presented in Attachment 2. It is assumed that a program to study passive failure mechanisms in valves will lead to a 50% reduction in such failures. This is assumed to result in a 6% reduction in hardware-related failures. The result for public risk reduction using the Oconee PWR are given below in Table 1.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Passive Mechanical Failure, B-58

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

   All plants.

	Backfit	Forward-fit	Total	$\bar{T}$ (yr)
PWRs	47	43	90	28.8
BWRs	24	20	44	27.4
	71	63	134	

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf - 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:

   All Oconee terms containing valves with hardware failure modes: B, C, D, E, CONST1, CONST2, A1, B1, C1, D•E, W•X, B•W, C•X, D•X, E•W, B•D, E•C.

TABLE 1. (contd)

5. Base-Case Values for Affected Parameters:

All parameters have the original values as given in Table A.4 of PNL-4297 (Andrews, et al. 1983).

6. Affected Accident Sequences and Base-Case Frequencies:

Sequence		Base-Case Frequency (1/py)
PWR:		
T_2MLU -	γ (PWR-3)	4.8E-7
	β (PWR-5)	6.9E-9
	ε (PWR-7)	4.8E-7
T_1MLU -	γ (PWR-3)	9.5E-7
	β (PWR-5)	1.4E-8
	ε (PWR-7)	9.5E-7
T_2MQH -	γ (PWR-3)	6.9E-7
	β (PWR-5)	1.0E-8
	ε (PWR-7)	6.9E-7
S_3H -	γ (PWR-3)	5.9E-7
	β (PWR-5)	8.6E-9
	ε (PWR-7)	5.9E-7
S_1D -	α (PWR-1)	5.3E-8
	γ (PWR-3)	1.1E-6
	β (PWR-5)	3.8E-8
	ε (PWR-7)	4.2E-6
T_2MQFH -	γ (PWR-2)	1.1E-7
	β (PWR-4)	1.5E-9
	ε (PWR-6)	1.1E-7
S_3FH -	γ (PWR-2)	9.0E-8
	β (PWR-4)	1.3E-9
	ε (PWR-6)	9.0E-8
S_2FH -	α (PWR-1)	5.7E-10
	β (PWR-4)	4.2E-10
	ε (PWR-6)	4.6E-8

TABLE 1. (contd)

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

Sequence		Base-Case Frequency (1/py)
PWR: (contd)		
S_2D -	α (PWR-1)	6.9E-9
	γ (PWR-3)	1.4E-7
	β (PWR-5)	5.1E-9
	ε (PWR-7)	5.5E-7
S_3D -	γ (PWR-3)	6.3E-7
	β (PWR-5)	9.2E-9
	ε (PWR-7)	6.3E-7
T_2MQD -	γ (PWR-3)	7.1E-7
	β (PWR-5)	1.0E-8
	ε (PWR-7)	7.1E-7

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 6.0E-8

PWR-2 = 2.0E-7

PWR-3 = 5.3E-6

PWR-4 = 3.2E-9

PWR-5 = 1.0E-7

PWR-6 = 2.5E-7

PWR-7 = 8.8E-6

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

$\bar{F}$(PWR) = 1.5E-5/py

$\bar{F}$(BWR) = 6.6E-6/py[a]

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

W(PWR) = 30 man-rem/py

W(BWR) = 36 man-rem/py[a]

(a) See Attachment 1.

TABLE 1. (contd)

10. Adjusted-Case, Values for Affected Parameters:

A 6% reduction in valve hardware failures is assumed for the adjusted case (see Attachment 2). The affected parameters for Oconee take on the following values:

B	= 0.0033
C	= 0.0033
D	= 0.023
E	= 0.023
CONST1	= 2.0E-4
CONST2	= 6.3E-4
A1	= 0.0097
B1	= 0.035
C1	= 0.0097
D • E	= 4.9E-4
W • X	= 8.7E-5
B • W	= 2.7E-5
C • X	= 2.7E-5
O • X	= 2.1E-4
E • W	= 2.1E-4
B • D	= 6.3E-5
E • C	= 6.3E-5

Note that only CONST1, A1, C1, and W•X show any change from the base case for two significant figures.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Sequence		Adjusted-Case Frequency (1/py)
PWR:		
T_2MLU -	γ (PWR-3)	4.5E-7
	β (PWR-5)	6.6E-9
	ε (PWR-7)	4.5E-7

TABLE 1. (contd)

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

Sequence		Adjusted-Case Frequency (1/py)
PWR: (contd)		
T_1MLU -	γ (PWR-3)	9.5E-7
	β (PWR-5)	1.4E-8
	ϵ (PWR-7)	9.5E-7
T_2MQH -	γ (PWR-3)	6.9E-7
	β (PWR-5)	1.0E-8
	ϵ (PWR-7)	6.9E-7
S_3H -	γ (PWR-3)	5.9E-7
	β (PWR-5)	8.6E-9
	ϵ (PWR-7)	5.9E-7
S_1D -	α (PWR-1)	5.3E-8
	γ (PWR-3)	1.1E-6
	β (PWR-5)	3.8E-8
	ϵ (PWR-7)	4.2E-6
T_2MQFH -	γ (PWR-2)	1.1E-7
	β (PWR-4)	1.5E-9
	ϵ (PWR-6)	1.1E-7
S_3FH -	γ (PWR-2)	9.0E-8
	β (PWR-4)	1.3E-9
	ϵ (PWR-6)	9.0E-8
S_2FH -	α (PWR-1)	5.7E-10
	β (PWR-4)	4.2E-10
	ϵ (PWR-6)	4.6E-8
S_2D -	α (PWR-1)	6.9E-9
	γ (PWR-3)	1.4E-7
	β (PWR-5)	5.1E-9
	ϵ (PWR-7)	5.5E-7

TABLE 1. (contd)

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

Sequence		Adjusted-Case Frequency (1/py)
PWR: (contd)		
S_3D -	γ (PWR-3)	6.2E-7
	β (PWR-5)	9.1E-9
	ε (PWR-7)	6.2E-7
T_2MQD -	γ (PWR-3)	7.1E-7
	β (PWR-5)	1.0E-8
	ε (PWR-7)	7.1E-7

Note that only T_2MLU-γ, β, ε and S_3D-γ, β, ε show any change from the base case for two significant figures.

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 6.0E-8
PWR-2 = 2.0E-7
PWR-3 = 5.3E-6
PWR-4 = 3.2E-9
PWR-5 = 1.0E-7
PWR-6 = 2.5E-7
PWR-7 = 8.8E-6

Note that no release category shows a change from the base case for two significant figures.

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

$\bar{F}^*$(PWR) = 1.5E-5/py (no change from the base case for two significant figures).

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

W*(PWR) = 30 man-rem/py (no change from the base case for two significant figures).

TABLE 1. (contd)

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

There is no change in the PWR's affected core-melt frequency from the base to the adjusted case for two significant figures. This implies the following:

$\Delta\bar{F}$(PWR) < 1E-6/py

$\Delta\bar{F}$(BWR) < 5E-7/py[(a)]

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

There is no change in the PWR's affected public risk from the base to the adjusted case for two significant figures. This implies the following:

ΔW(PWR) < 1 man-rem/py

ΔW(BWR) < 1 man-rem/py[(a)]

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
<4000	3.6E+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. 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 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:

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

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

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

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

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Passive Mechanical Failure, B-58.

2. Affected Plants (N): All Plants

	Backfit	Forward-fit	Total
PWR	47	43	90
BWR	24	20	44
	71	63	134

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

	Backfit (yr)	Forward-fit (yr)	Total (yr)
PWR	27.7	30	28.8
BWR	25.2	30	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 man-rem)(1E-6/py) = .02 man-rem/py

   $\Delta(\bar{F}D_R)_{BWR}$ < (19,900 man-rem)(5E-7/py) = .01 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
<6D	5600	0

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

   312 man-hr/BWR

   96 man-hr/PWR

   (See Attachment 2)

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

   D(BWR) = (312 man-hr/plant)(.025 R/hr) = 7.8 man-rem/plant

   D(PWR) = (96 man-hr/plant)(.025 R/hr) = 2.4 man-rem/plant

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

   ND = (7.8 man-rem/BWR)(24 BWR) + (2.4 man-rem/PWR)(47 PWR)

   = 300 man-rem

TABLE 2. (contd)

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

$$\text{Labor(BWR)} = (0.65 - 1.3 \text{ failures/py})(24 \text{ man-hr/failure}) = -15.6 \text{ man-hr/py}$$

$$\text{Labor(PWR)} = (0.18 - 0.36 \text{ failures/py})(24 \text{ man-hr/failure}) = -4.3 \text{ man-hr/py}$$

(See Attachment 2. Negative signs indicate reductions in labor.)

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

Again assuming .025 R/hr,

D_O(BWR) = -0.39 man-rem/py

D_O(PWR) = -0.11 man-rem/py

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

$N\bar{T}D_O$ = -1,140 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
-840	-280	-2500

(Negative signs indicate reductions)

ATTACHMENT 2

As with Issue C-11, Assessment of Failure and Reliability of Pumps and Valves, the operating experience for nuclear valves collected in NUREG/CR-0848 (Scott and Gallaher 1979) is used here to estimate the contribution of valve failure due to passive mechanisms. The data and categories for failure are shown below:

Failure Mechanism	% Failure by Reactor Type BWR	PWR	Passive Failure	Number of Reports BWR	PWR
Administrative Error	3	7			
Age Effects	<1	1	x	7	10
Corrosion	1	2	x	25	33
Crud	6	-	x	119	-
Erosion	<1	<1	x	8	6
Design Error	11	8			
Fabrication Error	4	4			
Fatigue	1	1			
Inherent	20	28			
Installation Error	4	5			
Maintenance Error	17	16			
Operator Error	6	7			
Vibration	2	2			
Wear	1	2			
Weather	<1	1	x	3	20
Flaw	1	-			
Leak	13	-			
Lubrication	2	-			
Stress	1	-	x	16	-
Stress Corrosion	<1	-	x	6	-
Other	-	18			
				184	69

Those items thought to be indicative of passive failure are checked. The distinction is made here in that active failures typically occur during valve operation, while passive failures occur over a period of time, going unnoticed as the valve is rendered inoperable. Detection of failure then occurs after valve operation is demanded. Referring to the preceding table, one finds that passive failures accounted for ~12% and ~5% of all reported valve failures at BWRs and PWRs, respectively (failure percents listed as <1% are valued at 1% in this calculation). Removing non-hardware-related failure modes (i.e., administrative, installation, maintenance and operator errors) from the preceding table indicates that passive falures accounted for the following percents of all reported hardware-related valve failures:

BWR-- 12%/70% = 17%
PWR-- 5%/65% = 8%

Over the 1965-1978 reporting period for the valve failure data, 140 py and 190 py of operating experience were accumulated at BWRs and PWRs, respectively (Appendix C; Andrews et al. 1982). Thus, the average contribution of passive failures to all reported hardware-related valve failures becomes:

$$\frac{(17\%)(140 \text{ py}) + (8\%)(190 \text{ py})}{140 \text{ py} + 190 \text{ py}} = 12\%$$

Assuming a program to investigate passive failures would be 50% effective in reducing such passive failures, it follows that the number of hardware-related valve failures can be reduced by 6% due to SIR.

The preceding table indicates that the numbers of failures over the reporting period are 184 and 69, or 1.3/py and 0.36/py, for BWRs and PWRs, respectively. Assuming as was done in Issue C-11 that the number of valves requiring replacement during SIR implementation is 10 times their annual failure rate, it is estimatd that 13 BWR and 4 PWR valves will be replaced per plant during this SIR implementation. Assuming 24 man-hr per valve replacement gives 312 man-hr/BWR and 96 man-hr/PWR for work in a radiation zone.

For SIR operation/maintenance a 50% improvement in the passive failure rates, or 0.65 failures/py and 0.18 failures/py, respectively, for BWRs and PWRs, is assumed. A repair time of 24 man-hr per failure is again assumed. Since maintenance on the aforementioned valves is anticipated to occur in a number of locations with varying radiation dose rates (depending on when and where the work is done), an average radiation field of 0.025 R/hr is assumed for purposes of this analysis.

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:

   Passive Mechanical Failure, B-58.

2. Affected Plants (N): All Plants

	Backfit	Forward-Fit	Total
PWR	47	43	90
BWR	24	20	44
	71	63	134

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

	Backfit (yr)	Forward-fit (yr)	Total (yr)
PWR	27.7	30	28.8
BWR	25.2	30	27.4

Industry Costs (Steps 4 through 12)

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

$$\Delta(\bar{F}A)_{PWR} < (\$1.65E+9)(1E-6/py) = \$1700/py$$

$$\Delta(\bar{F}A)_{BWR} < (\$1.65E+9)(5E-7/py) = \$800/py$$

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

Best Estimate	Upper Bound	Lower Bound
<\$5E+6	\$4.6E+8	0

6. Per-Plant Industry Resources for SIR Implementation:

   For backfit plants, it is assumed that the time spent in radiation zones (Step 6, Table 2) represents 20% of the total utility staff commitment. With administrative and engineering support, the labor estimates become:

TABLE 3. (contd)

$$\text{BWR-- } (5)\left(\frac{312 \text{ man-hr/plant}}{40 \text{ man-hr/man-wk}}\right) = 39 \text{ man-wk/plant}$$

$$\text{PWR-- } (5)\left(\frac{96 \text{ man-hr/plant}}{40 \text{ man-hr/man-wk}}\right) = 12 \text{ man-wk/plant}$$

Backfit equipment is assumed to consist of 13 valves/BWR and 4 valves/PWR at a cost of $30,000/valve. Work will presumably be conducted during normal outages, so no additional down-time is foreseen.

It is also assumed that the nuclear industry will fund research totalling $500,000, a cost spread over all 134 plants.

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

	Cost($/plant)			
	BWR		PWR	
Item	Backfit	Forward-Fit	Backfit	Forward-Fit
Labor	88,500	-	27,200	-
Equipment	390,000	-	120,000	-
Research	3,700	3,700	3,700	3,700
I	482,200	3,700	150,900	3,700

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

NI = $1.9E+7

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

SIR is assumed to reduce the labor, equipment, and outage time requirements attributable to passive valve failures during operation and maintenance. The labor and equipment requirements prior to SIR are as follows (from Attachment 2--note that the labor estimates from Attachment 2, 24 man-hr/failure, are increased by a factor of 5 to 120 man-hr/failure to include support labor, such as engineering and administration):

Labor

$$\text{BWR} = \frac{(1.3 \text{ failures/py})(120 \text{ man-hr/failure})}{40 \text{ man-hr/man-wk}} = 3.9 \text{ man-wk/py}$$

$$\text{PWR} = \frac{(0.36 \text{ failures/py})(120 \text{ man-hr/failure})}{40 \text{ man-hr/man-wk}} = 1.1 \text{ man-wk/py}$$

TABLE 3. (contd)

Equipment

BWR = 1.3 valve replacements/py
PWR = 0.36 valve replacements/py   at $30,000/valve

Active failures of pumps and valves are estimated to account for 10% of the average 60 days/py of routine down-time at a plant, or 6.0 days/py. Dividing this equally between pump and valve failures, one can attribute 3.0 days/py of down-time to active valve failures. Active valve failures were reported at rates of 639 failures in 140 py (BWRs), or 4.6/py, and 678 failures in 190 py (PWRs), or 3.6/py (NUREG/CR-D848). If the amount of down-time attributable to active valve failures is assumed proportional to their failure rates, then it follows that the amount of down-time attributable to passive valve failures will be proportional to the ratio of passive to active valve failure rates, or 1.3/4.6 = 0.28 for BWRs and 0.36/3.6 = 0.10 for PWRs. Thus, prior to SIR, the down-time attributable to passive valve failures is assumed to be:

Oown-Time

BWR = (0.28)(3.0 days/py) = 0.84 day/py

PWR = (0.10)(3.0 days/py) = 0.30 day/py

Assuming SIR reduces the passive valve failure rate by 50%, the following reductions in labor, equipment and down-time for operation and maintenance result:

	BWR	PWR
Labor (man-wk/py)	2.0	0.55
Equipment (valve replacements/py)	0.65	0.18
Down-Time (days/py)	0.42	0.15

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

	BWR	PWR
Labor (at $2,270/man-wk)	-$ 4,500/py	-$ 1,200/py
Equipment (at $30,000/valve)	-$ 19,500/py	-$ 5,400/py
Down-Time (at $3.0E+5/day)	-$126,000/py	-$45,000/py
I_O	-$150,000/py	-$51,600/py

(negative signs indicate cost savings)

TABLE 3. (contd)

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

$N\bar{T}I_O$ = -\$3.15E+8

12. Total Industry Cost (S_I):

Best Estimate	UpperBound	Lower Bound
-\$3.0E+8	-\$1.4E+8	-\$4.5E+8

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Estimate included directly in the next step.

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

It is assumed that \$50,000 would be required of NRC for review of valve failure data and to establish appropriate categories for data entries in failure data records. It is assumed that the majority of SIR development costs are funded by industry research into the problem (\$500,000 assumed in Step 6) Thus, C_D = \$50,000.

15-20. No additional NRC labor above that currently expended is foreseen for support of SIR implementation or review of SIR operation/maintenance. Thus, $C = C_O = 0$.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$5.0E+4	\$7.5E+4	-\$2.5E+4

REFERENCES

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

Scott, R. and R. Gallaher. 1979. Operating Experience with Valves in Light-Water Reactor Nuclear Power Plants for Period 1965-1978, NUREG/CR-0848. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

U.S. NRC. 1978. Generic Task Problem Descriptions--Category B, C, and D Tasks, NUREG-0471. U. S. Nuclear Regulatory Commission, Washington, D. C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: C-8, Main Steam Line Leakage Control Systems

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

Dose calculations have shown that operation of the MSIV leakage control systems at BWRs may result in higher offsite accident doses than if the releases take place through the condenser off-gas systems. The proposed resolution is to develop procedures to make the condenser off-gas system the preferred pathway and the MSIV leakage control systems its backup.

AFFECTED PLANTS	BWR:	Operating = 24	Planned = 20
	PWR:	Operating = 0	Planned = 0

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	1.1
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	8.0
Total of Above =	8.0
Accident-Avoidance =	0

COST RESULTS ($\$10^6$):

INDUSTRY COSTS:	
SIR Implementation =	6.4
SIR Operation/Maintenance =	3.0
Total of Above =	9.4
Accident-Avoidance =	0
NRC COSTS:	
SIR Development = } SIR Implementation Support = }	0.50[a]
SIR Operation/Maintenance Review	0.048
Total of Above =	0.55

(a) NRC costs for SIR development and implementation support are given as a combined estimate for this issue.

MAIN STEAM LINE LEAKAGE CONTROL SYSTEMS

ISSUE C-8

1.0 SAFETY ISSUE DESCRIPTION

Operation of the main steam line isolation valve leakage control system (MSIVLCS) required for some BWRs may result in higher offsite accident doses than if the system is not used and the integrity of the steam lines and condenser is maintained. Dose calculations by the Accident Analysis Branch in 1975 indicated a potential 100-1000-fold increase in offsite releases of iodine for proper operation of an MSIVLCS when compared to the calculations of releases assuming the steam system is intact and MSIV leakage is eventually released through the condenser. These calculations assumed nonoperation of the MSIVLCS and took credit for 1) cold trapping of iodine and volatiles in the steam lines and condenser and 2) long holdup times and release either through stack filters via the condenser air ejector or leakage from the steam system. Leakage from the main steam condenser system would be small because normal operation requires that leakage be maintained at a low level.

While integrity of these systems is not assured during earthquakes (they are not designed for the safe shutdown earthquake), the probability of seismic failure of both the fuel and these systems is small. By contrast, the MSIVLCS draws a negative pressure downstream of the MSIVs to collect leakage past the valve seats and processes the collected leakage through the safety-grade standby gas filtation system (SGTS) for release to the environment. Relatively little cold trapping or holdup time occurs when the MSIV leakage control system is used. Therefore, calculated doses for release through the MSIVLCS are greater than calculated doses for releases through the steam system unless the integrity of the steam system is lost.

Generic Safety Issue C-8 (US NRC 1978) was initiated to investigate whether or not the MSIVLCSs currently recommended in Regulatory Guide 1.96 (US NRC 1976) are desirable. In the meantime, new concerns have arisen because operational experience has indicated a relatively high MSIVLCS failure rate and a variety of failure modes at some BWR plants. MSIVLCSs as prescribed by Regulatory Guide 1.96 may be increasing the overall risk to the public. In addition, the question of backfitting MSIVLCSs to BWRs that do not have the systems has been raised.

ISSUE RESOLUTION

At BWRs equipped with MSIVLCSs, MSIV leakage can be released to the environment in any one of three ways:

1. Through the MSIVLCS, with subsequent release via the SGTS

2. Through the main steam condenser system, with subsequent release via the steam and waste gas treatment system

3. Directly to the atmosphere with no holdup or treatment.

Currently, the first pathway is preferred, although evidence indicates the second reduces the level of radioactivity released. The third is always least desirable.

At BWRs without MSIVLCSs, MSIV leakage can be released to the environment only in two ways, these being numbers two and three listed above. The second pathway is preferred at such plants.

Resolution of issue C-8 is assumed to be the following:

1. At all BWRs currently equipped with, or planning to install, MSIVLCSs, change operating/emergency procedures such that the condenser release pathway is the preferred one for MSIV leakage. The MSIVLCS will serve only as a backup should the preferred pathway be lost.

2. At all BWRs currently without, or not planning to install, MSIVLCSs, install MSIVLCSs to serve as a backup to the preferred condenser release pathway for MSIV leakage.

AFFECTED PLANTS

All BWRs are assumed to be affected. The backfit BWRs and all forward-fit BWRs beginning operation prior to 1986 are assumed to be currently equipped with, or planning to install, MSIVLCSs. At these plants, resolution only involves procedural revision to make the condenser release pathway the preferred one. The remaining forward-fit BWRs are assumed to be currently without, or not planning to install, MSIVLCSs. At these plants, resolution involves only MSIVLCS installation. Procedural revision to make the condenser the preferred pathway is assumed to be incorporated into the initial procedure writing at these forward-fit BWRs beginning operation in 1986 or later.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose due to the SIR are estimated in this section. Results are summarized in Tables 1 and 2, respectively. The public risk reduction estimate requires some deviation from the standardized procedure of PNL-4297 (Andrews 1983). The analysis is described in Attachment 1 to the Public Risk Reduction Work Sheet.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Main Steam Line Leakage Control Systems (C-8)

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

All 44 BWRs are assumed to be affected. These are divided into two groups:

1. All BWRs currently equipped with, or planning to install, MSIVLCSs. These are assumed to include all 24 backfit BWRs ($\bar{T}$ = 25.2 yr) and the 13 forward-fit BWRs beginning operation by 1986 ($\bar{T}$ = 30 yr). For the 37 BWRs in this group, ($\bar{T}$ = 26.9 yr).

2. All BWRs currently without, or not planning to install, MSIVLCSs. These presumably include the remaining seven forward-fit BWRs commencing operation in 1986 or beyond ($\bar{T}$ = 30 yr).

3. Plants Selected for Analysis:

The WASH-1400 BWR is chosen as the representative BWR (see Attachment 1).

4-8. Steps Related to Affected Parameters and Base-Case Accident Sequences, Release Categories and Affected Core-Melt Frequency ($\bar{F}$):

Analysis involves deviation from the standardized procedure of PNL-4297. This analysis is presented in Attachment 1. A base-case, affected core-melt release frequency of 3E-8/py is estimated for each group of BWRs defined in Step 2.

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

BWR Group[a]	W(man-rem/py)
24 backfit 13 forward-fit	3.6E-4
7 forward-fit	.0047

(See Attachment 1 for details.)

(a) Refer to Step 2.

TABLE 1. (contd)

10-13. <u>Steps Related to Adjusted-Case Values of Affected Parameters, Accident Sequences, Release Categories, and Core-Melt Frequency ($\bar{F}$*):</u>

Analysis involves deviation from the standardized procedure of PNL-4297. This analysis is presented in Attachment 1. The adjusted-case, affected core-melt release frequency for each group of BWRs defined in Step 2 does not change from the base case.

14. <u>Adjusted-Case, Affected Public Risk (W*):</u>

W* = 2.1E-4 man-rem/py (same for both BWR groups)

(See Attachment 1 for details.)

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

Zero (same for both BWR groups)

16. <u>Per-Plant Reduction in Public Risk (ΔW):</u>

BWR Group[a]	ΔW(man-rem/py)
24 backfit 13 forward-fit	1.5E-4
7 forward-fit	.0045

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.1	40	0

ATTACHMENT 1

The Final Safety Analysis Report (FSAR) for the Limerick Generating Station indicates that the MSIVLCS at each BWR unit is designed to limit the leakage rate from an MSIV to 100 ft^3/hr. It is not required to operate below the technical specification leakage rate of 11.5 ft^3/hr. Thus, given the BWR containment free volume of 2.78E+5 ft^3, these leakage rates translate into:

$$\text{Upper Limit} = \left(\frac{100\ ft^3/hr}{2.78E+5\ ft^3}\right)(24\ hrs/day) = .0086\ volume/day$$

$$\text{Lower Limit} = \left(\frac{11.5\ ft^3/hr}{2.78E+5/ft^3}\right)(24\ hrs/day) = 9.9E-4\ volume/day$$

Thus, the effective range of the MSIVLCS is approximately 0.1% to 1% volume/day.

Table 5-3 of WASH-1400 lists the following dominant BWR accident sequences (and frequencies) involving containment leakage ("G") with release via the drywell ("δ")[US NRC 1975]:

AGJ - δ	= 6E-11/py
AEG - δ	= 7E-10/py
AGHI - δ	= 6E-11/py
S_1GJ - δ	= 2E-10/py
S_1GE - δ	= 2E-10/py
S_1GHI - δ	= 2E-10/py
S_2CG - δ	= 6E-11/py
S_2GHI - δ	= 6E-10/py
S_2EG - δ	= 3E-10/py
S_2GJ - δ	= 6E-10/py
S_2GI - δ	= 2E-10/py
Total	= 3E-9/py

All sequences contribute to release category BWR-4.

The above sequences assume containment leakage rates in excess of 100% volume/day (G, with a likelihood of .0057 [from Table II 1-5 of WASH-1400--Q_{median} for drywell leakage > 1 in^2]). Also assumed is leakage via the drywell pathway (δ, with a probability of 0.86 [from Table V-9 of WASH-1400]). Thus, in the above sequences, the product $G\delta$ has a probability of .005.

Since G refers to much higher leakage rates than applicable for the MSIVLCS, it is necessary to redefine both G and δ for the effective range of 0.1% to 1% volume/day. The likelihood of the product $G\delta$ is expected to increase. A recent report by M. Hallins, "Technical Evaluation of Browns Ferry Nuclear Plant MSIV Containment Integrity Leak Rates," estimates a probability of 0.95 for MSIV leakage occurring at a rate <11.5 ft^3/hr (0.1% volume/day). Thus, the likelihood is 0.05 that MSIV leakage will occur at a rate $\geq$0.1% volume/day. The value 0.05 is a conservative estimate of the likelihood that MSIV leakage occurs at a rate within the effective range of the MSIVLCS.

Redefining the product $G\delta$ to represent MSIV leakage in excess of 0.1% volume/day, a redefined value of 0.05 is assigned to this product. Since this is ten times higher than the original 0.005 value used in WASH-1400, it follows that the release frequency from the previously listed dominant sequences will be ten times higher when redefined for MSIV leakage in excess of 0.1% volume/day. Thus, the release frequency due to MSIV leakage in excess of 0.1% volume/day is taken to be (10)(3E-9/py) = 3E-8/py.

As discussed in Section 1.0, radioactivity escaping past the MSIVs can be released to the environment via the following pathways:

1. Through the MSIVLCS (if available) and SGTS,

2. Through the main steam condenser system and the steam and waste gas treatment systems

3. Directly to the atmosphere.

Consider three BWR-types:

1. BWRs where the MSIVLCS is the preferred pathway, with the condenser pathway serving as backup

2. BWRs where the condenser pathway is preferred, with the MSIVLCS serving as backup

3. BWRs without an MSIVLCS, leaving only the condenser pathway as the alternative to direct release.

Given unavailabilities of the MSIVLCS and the condenser pathways of .05 each (from discussions with NRC staff), the likelihoods of release via the various pathways at the three BWR-types become:

BWR-type	Pathway	Likelihood
MSIVLCS preferred	MSIVLCS	0.95
	Condenser	(.05)(.95) = .048
	Direct	1 - .95 - .048 = .002
Condenser preferred (MSIVLCS backup)	Condenser	0.95
	MSIVLCS	(.05)(.95) = .048
	Direct	1 - .95 - .048 = .002
No MSIVLCS	Condenser	0.95
	Direct	1 - .95 = .05

The dominant BWR accident sequences listed earlier all belong to release category BWR-4, which has an associated dose factor of 6.1E+5 man-rem (Andrews 1982). This release category presumes release occurs via the reactor building with treatment by the SGTS, essentially the same pathway as that for the MSIVLCS. However, the maximum design leakage rate for the MSIVLCS is ~1% volume/day, compared to the WASH-1400 definition of G for leakage rates >100% volume/day. It is assumed that the dose factor for release via the MSIVLCS and SGTS is 100 times less than that for BWR-4, i.e., (.01)(6.1E+5 man-rem) = 6100 man-rem. The dose factor for release via the condenser pathway is assumed to be ten times less, or 610 man-rem. This is believed to be conservative in light of existing dose calculations. The dose factor for direct release is assumed to be five times higher than that for BWR-4, or 3.1E+6 man-rem, since no treatment by the SGTS will occur.

For the three BWR-types defined, the affected public risks become:

1. BWRs with preferred MSIVLCSs

$$W_1 = (3E\text{-}8/py)[(.95)(6100 \text{ man-rem}) + (.048)(610 \text{ man-rem}) + (.002)(3.1E{+}6 \text{ man-rem})] = 3.6E\text{-}4 \text{ man-rem/py}$$

2. BWRs with backup MSIVLCSs

$$W_2 = (3E\text{-}8/py)[(.95)(610 \text{ man-rem}) + (.048)(6100 \text{ man-rem}) + (.002)(3.1E{+}6 \text{ man-rem})] = 2.1E\text{-}4 \text{ man-rem/py}$$

3. BWRs with no MSIVLCSs

$$W_3 = (3E\text{-}8/py)[(.95)(610 \text{ man-rem}) + (.05)(3.1E{+}6 \text{ man-rem})] = .0047/py$$

For BWRs currently equipped with, or planning to install, MSIVLCSs, SIR will result in changing the MSIVLCS pathway from preferred to backup. The resulting risk reduction is:

$$\Delta W = W_1 - W_2 = 1.5E\text{-}4 \text{ man-rem/py}$$

For BWRs currently without, or not planning to install, MSIVLCSs, SIR will result in adding the MSIVLCS as a backup pathway to the preferred condenser pathway. The resulting risk reduction is:

$$\Delta W = W_3 - W_2 = .0045 \text{ man-rem/py}$$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Main Steam Line Leakage Control Systems (C-8)

2. Affected Plants (N):

   As discussed in Step 2 of Table 1, the 44 affected BWRs are divided into the following groups:

   1. 37 BWRs currently equipped with, or planning to install, MSIVLCSs.

   2. 7 forward-fit BWRs (commencing operation in 1986 or beyond) currently without, or not planning to install, MSIVLCSs.

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

BWR Group[a]	$\bar{T}$(yr)
37 BWRs	26.9
7 forward-fit BWRs	30

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

   Since the reduction in core-melt frequency is zero, there is no occupational dose reduction due to accident-avoidance.

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
0	4.3	0

(a) Refer to Step 2.

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

Procedural modification will require no radiation exposure. Addition of MSIVLCSs where currently unplanned will take place only at seven forward-fit BWRs prior to their operation. Thus, no radiation exposure will result.

$$D = 0$$

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

Only at the seven forward-fit BWRs adding MSIVLCSs to their designs will utility labor in radiation zones result for SIR operation and maintenance. As given in Step 9 of Table 3, a value of 0.5 man-wk/py is assumed for SIR operation and maintenance. Since this should involve primarily testing and inspection (and any resulting maintenance), the following estimate of labor in radiation zones results (assuming a 75% utilization factor):

$$(0.5 \text{ man-wk/py})(0.75) = 0.38 \text{ man-wk/py}$$

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

Assuming that MSIVLCS testing, inspection and maintenance occur during scheduled outages, an exposure rate of 2.5 mR/hr is deemed appropriate.

$$D_0 = (.0025 \text{ R/hr})(0.38 \text{ man-wk/py})(40 \text{ man-hr/man-wk}) = .038 \text{ man-rem/py}$$

(This applies only to 7 post-1985 forward-fit BWRs.)

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

$$N\bar{T}D_0 = (7)(30)(.038) = 8.0 \text{ man-rem}$$

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
8.0	24	2.7

3.0 SAFETY ISSUE COSTS

The industry and NRC costs due to the SIR 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:

   Main Steam Line Leakage Control Systems (C-8)

2. Affected Plants (N):

   As discussed in Step 2 of Table 1, the 44 affected BWRs are divided into the following groups:

   1. 24 backfit BWRs and 13 forward-fit BWRs (commencing operation by 1986) currently equipped with, or planning to install, MSIVLCSs.

   2. Seven remaining forward-fit BWRs (commencing operation in 1986 or beyond) currently without, or not planning to install, MSIVLCSs.

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

BWR Group[a]		$\bar{T}$(yr)	
24 backfit	37 BWRs	25.2	26.9
13 forward-fit		30	
7 forward-fit		30	

Industry Costs (Steps 4 through 12)

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

   Since the reduction in core-melt frequency is zero, there is no industry cost savings due to accident-avoidance.

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

Best Estimate	Upper Bound	Lower Bound
0	\$3.6E+5	0

(a) Refer to Step 2.

6. Per-Plant Industry Resources for SIR Implementation:

Both groups of affected BWRs (see Step 2) will have to develop modify operating/emergency procedures which call for the condenser release pathway, rather than the MSIVLCS, to be used as the preferred release pathway in the event of MSIV leakage. Control room display equipment will also have to be procured and installed. However, at the seven BWRs commencing operation in 1986 or beyond (those currently without, or not planning to install, MSIVLCSs), this procedural development and control room display installation can be included as part of the initial licensing effort. Thus, this inclusion requires no additional resources beyond those which would be expended if the MSIVLCS, rather than the condenser pathway, were preferred.

Procedural modification is assumed to require 0.5 man-yr/plant for labor at the 37 BWRs currently equipped with, or planning to install, MSIVLCSs. The cost of procuring and installing additional control room display equipment at these 37 BWRs is estimated directly in the next step.

For the seven BWRs commencing operation in 1986 or beyond (those currently without, or not planning to install, MSIVLCSs), MSIVLCSs must be added, since they were not included in the original design. The cost of this, including equipment procurement and installation labor, is estimated directly in the next step.

For the 24 backfit BWRs currently equipped with MSIVLCSs, a class III license amendment is assumed necessary due to the procedural modification.

In summary, the resources required to implement SIR at the affected BWRs are as follows:

BWR Group[a]	Resources (per plant)
24 backfit BWRs	0.5 man-yr Control room display license amendment
13 forward-fit BWRs (prior to 1986)	0.5 man-yr Control room display
7 forward-fit BWRs (in 1986 or beyond)	MSIVLCS

(a) Refer to Step 2.

TABLE 3. (contd)

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

The following costs are estimated directly for resources listed in Step 6:

Control Room Display	= \$2.5E+4/plant
Class III License Amendment	= \$4000/plant
MSIVLCS	= \$5.0E+5/plant

For labor costs, a rate of \$1.0E+5/man-yr is assumed (Appendix E of PNL-4297). For the affected BWRs, the implementation costs are as follows:

BWR Group	Cost (\$/plant)
24 backfit BWRs	50,000
	25,000
	4,000
	I = 79,000
13 pre-1986 forward-fit BWRs	50,000
	25,000
	I = 75,000
7 post-1985 forward-fit BWRs	500,000
	I = 500,000

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

$$NI = (24)(\$7.9E+4) + (13)(\$7.5E+4) + (7)(\$5.0E+5)$$
$$= \$6.4E+6$$

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

Plant operators will be required to become initially familiar with the procedural modifications related to the MSIVLCS as well as to annually refresh their knowledge of the procedures. Per operator, this is assumed to require 0.5 day/yr. Assuming ten operators per plant gives a labor estimate of (0.5 man-day/yr)(10 operators/plant) = 5 man-day/py, or 1 man-wk/py. This applies to all 44 affected BWRs.

TABLE 3. (contd)

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

At the seven forward-fit BWRs commencing operation after 1985, annual operation/maintenance of the installed MSIVLCSs will be in addition to any currently expended (as at the 24 backfit BWRs) or planned to be expended (as at the 13 pre-1986 forward-fit BWRs). This additional labor is assumed to amount to 0.5 man-wk/py at the seven forward-fit BWRs.

In summary, the labor required for SIR operation/maintenance at the affected BWRs is as follows:

BWR Group[a]	Labor (man-rem/py)
24 backfit BWRs and 13 pre-1986 forward-fit BWRs (37 BWRs)	1
7 post-1985 forward-fit BWRs	1 0.5 1.5

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

BWR Group	I_O (\$/py)
37 BWRs	2270
7 post-1985 BWRs	3410

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

$$N\bar{T}I_O = (37)(26.9)(\$2270) + (7)(30)(\$3410)$$
$$= \$3.0E+6$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$9.4E+6	\$1.3E+7	\$5.9E+6

(a) Refer to Step 2.

TABLE 3. (contd)

NRC Costs (Steps 13 through 21)

13-17. Steps Related to NRC Cost for SIR Development and Support of SIR Implementation:

The estimated NRC effort for SIR development and implementation support is not broken down into specific development and support efforts. There may be considerable overlap and iteration between the two. Thus, only an overall cost of \$5.0E+5 for NRC staff labor and contractor support is estimated to result from both SIR development and implemention support (C_D + NC = \$5.0E+5).

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

Additional NRC labor beyond that currently expended (or anticipated to be expended) will possibly arise only at the seven post-1985 forward-fit BWRs assumed to add MSIVLCSs to their designs. Since these will be included with routine inspection, only a modest 0.5 man-day/py is assumed to be needed.

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-day)(\$2270/man-wk) = \$227/py
(This applies only to 7 post-1985 forward-fit BWRs).

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

$N\bar{T}C_O$ = (7)(30)(\$227) = \$4.8E+4

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$5.5E+5	\$8.0E+5	\$3.0E+5

REFERENCES

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: I.A.2.7, Accreditation of Training Institutions

SUMMARY OF PROBLEM ANO PROPOSED RESOLUTION:

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

AFFECTED PLANTS	BWR:	Operating = 22	Planned = 18
	PWR:	Operating = 42	Planned = 39

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 3.1E+4

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	-2.1E+4
Total of Above =	-2.1E+4
Accident-Avoidance =	210

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:

SIR Implementation =	36
SIR Operation/Maintenance =	340
Total of Above =	380
Accident-Avoidance =	17

NRC COSTS:

SIR Development =	0.25
SIR Implementation Support =	0.25
SIR Operation/Maintenance Review =	2.8
Total of Above =	3.3

ACCREDITATION OF TRAINING INSTITUTIONS

TMI ACTION PLAN ITEM I.A.2.7

1.0 SAFETY ISSUE DESCRIPTION

This safety issue as described in NUREG-0660 calls for a study by NRR on procedures and requirements for NRC accredition of training institutions for nuclear plant operations. This would result in an NRR information paper. SD would then be called on to prepare a commission paper describing the various options. Coordination with INPO was also called for.

The current status of this issue finds NRR rather than SO developing the commission paper. The participation of INPO has undergone some revisions. Their input will be sought but the magnitude of their participation may be less than originally envisioned.

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 envisioned the resolution of this safety issue as the formation of an accreditation board consisting of representatives from the NRC, industry and academe. This board would develop and apply criteria for accreditation. Thess would include training programs of utilities, university-related programs, and independent training institutions. While theoretically applying to training for all operations staff, the PNL panel felt the current thrust is focused on reactor operators. Therefore, the assessment was made assuming only operators would be affected.

The insights of the panel included the awareness that some training programs are very near to accreditation already. Either through association with universities or other means of providing high quality instruction, their programs would be likely to quickly acquire accreditation from the board. Other training programs are not so well prepared for accreditation and may require significant effort and expense. Some savings may be gained for multi-unit sites in sharing costs.

The issue summary work sheet which is shown on the cover page provides a summary of the analysis of the SIR. In the following sections the details of the analysis are further described.

2.0 SAFETY ISSUE RISK AND DOSE

A reduction in public risk through the improvement of operator performance is expected from the improved training accreditation. Likewise, a reduction in occupational exposure is also expected. This will be primarily for operators, who often supervise maintenance or perform other duties in radiation zones. However, some reduction in routine occupational exposure can be expected for other operations personnel from the increased awareness in operators.

These two terms, public risk reduction and occupational dose, are described in the following two sections.

2.1 PUBLIC RISK REDUCTION

As was previously discussed, the major result of the resolution of this safety issue was assumed to be an improvement in operator performance. For some utilities, judged to be approximately 10% of the total, this issue will have essentially no effect. This is because 1) their current training programs would be accredited with little effort and 2) the quality of their programs is sufficiently high that accreditation would result in no discernable improvement in their operators' performance. Other utilities will see a varying degree of improvement. Those with training programs that are below the accreditation standards will be brought up nearer to the high quality enjoyed by the outstanding utilities. Overall, the effect on operator error is estimated by the panel to be a reduction of 10% across the affected portion of the industry.

It is worthwhile to repeat that these estimates are the intuitive judgements of a panel of experienced experts. As such they are not hard numbers. However, due to the lack of specificity and unknowns associated with the issue resolution and the lack of firm data linking training with improved operator performance, these are the best estimates which could be obtained within the scope of the project.

Table 1 is the work sheet for the public risk reduction. It describes how the estimated 10% reduction in operator error is used to calculate public man-rem averted.

2.2 OCCUPATIONAL DOSE

The PNL panel felt this issue would have a small but finite effect on occupational dose. The major effect is expected to be in the reduction of dose to operators themselves. The improved training they receive as a result of accreditation is expected to include increased emphasis on radiation safety and health physics. With this increased awareness, the exposure operators receive in their routine duties is expected to be reduced. A secondary effect would be that this increased awareness of operators would be passed on to other operational personnel, thereby reducing their exposures as well. The overall effect is estimated to be small, on the order of a 1 to 2% reduction in plant-wide routine dose. A value of 1.5% is used in the calculations which follow. It is estimated that 300 to 500 man-rem of occupational exposure occurs at a typical facility annually. If we assume 400 man-rem as a best estimate, the 1.5% reduction results in an occupational dose reduction of 6 man-rem per plant year.

In addition to the operational dose, the prioritization formulation calls for estimates of implementation occupational dose and avoided occupational dose associated with cleanup of accidents (avoided due to accident frequency reduction). For this issue, there is no implementation dose. The accident-related avoided dose and the operational dose development are shown in Table 2.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Accreditation of Training Institutions (I.A.2.7)

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

   90% of all plants.

	N	$\bar{T}$(yr)
PWR	81	28.8
BWR	40	27.4

3. Plants Selected for Analysis:

   Oconee 3 - represenative PWR (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 1983).

6. Affected Accident Sequences and Base-Case Frequencies:

Sequence	Frequency (1/py)
T_2MLU - γ (PWR-3)	5.8E-7
T_2MLU - β (PWR-5)	8.5E-9
T_2MLU - ε (PWR-7)	5.8E-7

TABLE 1. (contd)

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

Sequence	Frequency (1/py)
$T_1MLU - \gamma$ (PWR-3)	9.8E-7
$T_1MLU - \beta$ (PWR-5)	1.4E-8
$T_1MLU - \varepsilon$ (PWR-7)	9.8E-7
$T_1(B_3)MLU - \gamma$ (PWR-3)	1.1E-6
$T_1(B_3)MLU - \beta$ (PWR-5)	1.6E-8
$T_1(B_3)MLU - \varepsilon$ (PWR-7)	1.1E-6
$T_2MQH - \gamma$ (PWR-3)	3.2E-6
$T_2MQH - \beta$ (PWR-5)	4.7E-8
$T_2MQH - \varepsilon$ (PWR-7)	3.2E-6
$S_3H - \gamma$ (PWR-3)	2.8E-6
$S_3H - \beta$ (PWR-5)	4.1E-8
$S_3H - \varepsilon$ (PWR-7)	2.8E-6
$S_1D - \alpha$ (PWR-1)	5.3E-8
$S_1D - \gamma$ (PWR-3)	1.1E-6
$S_1D - \beta$ (PWR-5)	3.9E-8
$S_1D - \varepsilon$ (PWR-7)	4.3E-6
$T_2MQFH - \gamma$ (PWR-2)	2.4E-6
$T_2MQFH - \beta$ (PWR-4)	3.6E-8
$T_2MQFH - \varepsilon$ (PWR-6)	2.4E-6
$S_3FH - \gamma$ (PWR-2)	2.0E-6
$S_3FH - \beta$ (PWR-4)	3.0E-8
$S_3FH - \varepsilon$ (PWR-6)	2.0E-6
$S_2FH - \alpha$ (PWR-1)	1.2E-8
$S_2FH - \beta$ (PWR-4)	8.9E-9
$S_2FH - \varepsilon$ (PWR-6)	9.8E-7

TABLE 1. (contd)

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

Sequence	Frequency (1/py)
$T_2KMU - \gamma$ (PWR-3)	3.9E-6
$T_2KMU - \beta$ (PWR-5)	5.7E-8
$T_2KMU - \epsilon$ (PWR-7)	3.9E-6
$S_2D - \alpha$ (PWR-1)	7.2E-9
$S_2D - \gamma$ (PWR-3)	1.4E-7
$S_2D - \beta$ (PWR-5)	5.2E-9
$S_2D - \epsilon$ (PWR-7)	5.7E-7
$S_3D - \gamma$ (PWR-3)	6.7E-7
$S_3D - \beta$ (PWR-5)	9.8E-9
$S_3D - \epsilon$ (PWR-7)	6.7E-7
$T_2MQD - \gamma$ (PWR-3)	7.2E-7
$T_2MQD - \beta$ (PWR-5)	1.1E-8
$T_2MQD - \epsilon$ (PWR-7)	7.2E-7

(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.)

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: 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.)

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

$\bar{F}_{PWR}$ = 4.91E-5/py $\bar{F}_{BWR}$ = 2.21E-5/py[(a)]

(a) See Attachment 1.

TABLE 1. (contd)

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

W_{PWR} = 116.3 man-rem/py W_{BWR} = 139.6 man-rem/py[a]

10. Adjusted-Case Values for Affected Parameters:

B = C	=	0.0032
D = E	=	0.023
CONST1	=	2.1E-4
CONST2	=	6.3E-4
A1 = C1	=	0.0098
B1	=	0.035
HHMAN	= HPMAN1 =	0.090
HPMAN	=	0.0135
HPRSCM	= WXCM =	0.0027
D•E	=	4.9E-4
B•W	= C•X	2.6E-5
D•X	= E•W =	2.1E-4
B•D	= E•C =	6.0E-5

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Sequence	Frequency (1/py)
$T_2MLU - \gamma$	5.2E-7
$T_2MLU - \beta$	7.7E-9
$T_2MLU - \varepsilon$	5.2E-7
$T_1MLU - \gamma$	8.9E-7
$T_1MLU - \beta$	1.3E-8
$T_1MLU - \varepsilon$	8.9E-7
$T_1(B_3)MLU - \gamma$	9.8E-7
$T_1(B_3)MLU - \beta$	1.4E-8
$T_1(B_3)MLU - \varepsilon$	9.8E-7

(a) See Attachment 1.

TABLE 1. (contd)

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

Sequence	Frequency (1/py)
$T_2MQH - \gamma$	3.0E-6
$T_2MQH - \beta$	4.3E-8
$T_2MQH - \varepsilon$	3.0E-6
$S_3H - \gamma$	2.6E-6
$S_3H - \beta$	3.8E-8
$S_3H - \varepsilon$	2.6E-6
$S_1D - \alpha$	5.3E-8
$S_1D - \gamma$	1.1E-6
$S_1D - \beta$	3.9E-8
$S_1D - \varepsilon$	4.3E-6
$T_2MQFH - \gamma$	2.2E-6
$T_2MQFH - \beta$	3.2E-8
$T_2MQFH - \varepsilon$	2.2E-6
$S_3FH - \gamma$	1.8E-6
$S_3FH - \beta$	2.7E-8
$S_3FH - \varepsilon$	1.8E-6
$S_2FH - \alpha$	1.1E-8
$S_2FH - \beta$	8.0E-9
$S_2FH - \varepsilon$	8.8E-7
$T_2KMU - \gamma$	3.5E-6
$T_2KMU - \beta$	5.1E-8
$T_2KMU - \varepsilon$	3.5E-6
$S_2D - \alpha$	7.2E-9
$S_2D - \gamma$	1.4E-7
$S_2D - \beta$	5.2E-9
$S_2D - \varepsilon$	5.7E-7

TABLE 1. (contd)

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

Sequence	Frequency (1/py)
$S_3D - \gamma$	6.7E-7
$S_3D - \beta$	9.8E-9
$S_3D - \epsilon$	6.7E-7
$T_2MQD - \gamma$	7.2E-7
$T_2MQD - \beta$	1.1E-8
$T_2MQD - \epsilon$	7.2E-7

(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 = 7.8E-8/py
PWR-2 = 5.2E-6/py
PWR-3 = 1.5E-5/py
PWR-4 = 8.3E-8/py
PWR-5 = 2.4E-7/py
PWR-6 = 6.4E-6/py
PWR-7 = 1.9E-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 ($\bar{F}^*$):

$\bar{F}^*_{PWR}$ = 4.54E-5/py

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

W^*_{PWR} = 107.8 man-rem/py

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

$(\Delta\bar{F})_{PWR}$ = 3.7 E-6/py

$(\Delta\bar{F})_{BWR}$ = 1.7 E-6/py[(a)]

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

$(\Delta W)_{PWR}$ = 8.5 man-rem/py

$(\Delta W)_{BWR}$ = 10. man-rem/py[(a)]

(a) See Attachment 1.

TABLE 1. (contd)

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

3.1E+4 man-rem

Upper bound = 1.3E+7 man-rem

Lower bound = 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 (Andrews 1983). Using the original release category frequencies and the public dose factors (Appendix D of PNL-4297), 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 (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 value 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:

   Accreditation of Training Institutions (I.A.2.7)

2. Affected Plants (N):

   90% of all plants.

	N
PWRs	81
BWRs	40

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, $\Delta(\bar{FO}_R)$:

   PWR = (19,860 man-rem)(3.7E-6/py) = 0.073 man-rem/py

   BWR = (19,860 man-rem)(1.7E-6/py) = 0.034 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
210	1.7E+4	0

6-8. These steps do not apply since the implementation occupational dose is zero.

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

   Dose estimated directly (See Step 10).

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

    -6 man-rem/py (reduction)

TABLE 2. (contd)

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

-2.1E+4 man-rem (reduction)

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
-2.1E+4	-6.8E+3	-6.2E+4

3.0 SAFETY ISSUE COSTS

The PNL panel also estimated the costs associated with the implementation and operation of the resolution to this safety issue. The cost to industry to implement the change was estimated to be in the range of $1E+5 to $1E+6 per reactor. Those with training programs closer to accreditable status would employ the smaller cost. The best estimate for the average plant was taken to be $3.0E+5. This represents an average effort of three person-years per facility. Activities included in this effort are assumed to be 1) review accreditation standards, 2) compare present utility practices, 3) plan upgrades, and 4) implement program upgrades to fulfill accreditation requirements.

Operation under the accreditation program was estimated to cost industry between $5.0E+4 and $2.5E+5 per facility annually. The best estimate was taken to be $1.0E+5 per plant annually. This represents one person-year of effort, which is assumed to be absorbed by 1) operation staff participation in upgraded training and 2) additional instruction time.

The cost to the NRC to develop and support implementation of the accreditation was estimated to be $5.0E+5. This represents an estimte of 5 person-years to develop the accreditation standards, put them into regulations, and see their adoption. This is based on the perception that accreditation from INPO is not forthcoming. Greater INPO participation would be likely to reduce NRC cost. At the time the estimates were made there was no requirement to separate development and implementation support costs, therefore the panel made no distinction. If we assume an equal division, each is assigned a cost of $2.5E+5.

The annual cost to the NRC for review of operation is estimated to be $1.0E+5 over all affected plants. This represents one person-year annually of IE effort to assure that accreditation standards are being met.

The remaining cost term is the cost savings to industry associated with accident-avoidance. Its development and those of the other cost terms are given in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Accreditation of Training Institutions (I.A.2.7)

2. Affected Plants (N):

   90% of all plants.

	N
PWRs	81
BWRs	40

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, $\Delta(\bar{F}A)$:

   PWR = ($1.65E+9)(3.7E-6/py) = 6.1E+3/py

   BWR = ($1.65E+9)(1.7E-6/py) = 2.8E+3/py

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

Best Estimate	Upper Bound	Lower Bound
$1.7E+7	$1.4E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

   Labor: 3 person-yr/plant

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

   $3.0E+5/plant

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

   $3.6E+7

TABLE 3. (contd)

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

   1.0 person-yr/py

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

   \$1.0E+5/py

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

   \$3.4E+8

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$3.8E+8	\$5.5E+8	\$2.1E+8

NRC Costs (Steps 13 through 21):

13. NRC Resources for SIR Development:

   Cost estimated directly in next step.

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

   \$2.5E+5

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

   Cost estimated directly in Step 17.

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

   Cost estimated directly in Step 17.

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

   \$2.5E+5

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

   Cost estimated directly in Step 20.

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

   Cost estimated directly in Step 20.

TABLE 3. (contd)

20. Total NRC Cost for Review of SIR Operation and Maintenance ($\bar{NTC}_0$):

$(\$1.0E+5/yr)(28.3\ yr) = \$2.8E+6$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$3.3E+6	$4.8E+6	$1.9E+6

REFERENCE

Andrews, W. 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.C.1(4), Short-Term Accident Analysis and Procedures Revision (Confirmatory Analysis of Selected Transients by NRC)

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

This TMI action item seeks to perform confirmatory analyses of selected transients by NRR to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons will assure the adequacy of the analytical methods being used to generate emergency procedures.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	9.7E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	-7.6E+4
Total of Above =	-7.6E+4
Accident-Avoidance =	570

COST RESULTS ($$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	9.1
SIR Operation/Maintenance =	60
Total of Above =	69
Accident-Avoidance =	47
NRC COSTS:	
SIR Development =	1.8
SIR Implementation Support =	0.11
SIR Operation/Maintenance Review=	1.4
Total of Above =	3.3

SHORT-TERM ACCIDENT ANALYSIS AND PROCEDURE REVISION (CONFIRMATORY ANALYSIS OF SELECTED TRANSIENTS BY NRC) TMI ACTION ITEM - I.C.1(4)

1.0 SAFETY ISSUE DESCRIPTION

The objective of this safety issue is to improve the quality of procedures through confirmatory analyses of selected transients by NRC/NRR to provide greater assurance that operator and staff actions are technically correct. The analyses, using the best available computer codes, will provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons, together with comparisons to other data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures. The issue summary work sheet 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

Based on the information available to PNL, NRC has performed a limited number of confirmatory transient analyses. The rest are currently being defined. Estimates are made by PNL on the total scope of the confirmatory analyses.

Benefits are given in terms of the reduction in operator errors and upgrading of operating systems. Table 1 gives the reduction in overall core-melt frequency from improvements in these two areas. It also gives the percent decrease in annual occupational dose upon the implementation of the safety issue resolution (SIR) as a result of the confirmatory analyses. These estimates were made by PNL staff with considerable experience in the areas of nuclear power plant and reactor systems analysis.

TABLE 1. Estimates of Safety Benefits

	Reduction in Overall Core-Melt Frequency (%)			Decrease in Annual Occupational Dose (%)		
	Lower Bound	Best Estimate	Upper Bound	Lower Bound	Best Estimate	Upper Bound
From improvement in human error rate for operators	3.5	7	17.5	--	--	--
From other operations improvements (set points for control systems, maintenance, hardware upgrade, etc.)	1.5	4.5	9	--	--	--
Total	5	11.5	26.5	0	5	10

2.1 PUBLIC RISK REDUCTION

Because of multifactor influences of the SIR on the safety benefits, it is judged appropriate to apply the total percent reduction to the base-case frequencies of all affected release categories. This assumes all sequences are affected. See Table 2 for results.

2.2 OCCUPATIONAL DOSE

This SIR is expected to affect all reactors and result in dose reduction during SIR operation and maintenance as compared to the dose currently being accrued. Results are summarized in Table 3.

TABLE 2. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Short-Term Accident Analysis and Procedures Revision (Confirmatory Analysis of Selected Transient by NRC) [I.C.1(4)]

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

All 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

4-6. Steps Related to Affected Parameters, Accident Sequences, and Their Base-Case Values:

It is judged that this SIR affects most of the parameters. Therefore, a uniform reduction of 11.5% is applied directly to the frequencies of the release categories. Thus, steps 4-6 are skipped.

7. Affected Release Categories and Base-Case Frequencies:

All release categories are affected by issue resolution. The original frequencies are assumed for the base case.

TABLE 2. (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

10-11. Steps Related to Adjusted-Case Values of Affected Parameters and Accident Sequences:

Analysis not performed for these steps since the 11.5% reduction is applied directly to the release category frequencies.

12. Affected Release Categories and Adjusted-Case Frequencies:

All affected release categories are assumed to be subject to an 11.5% reduction in frequency due to SIR.

PWR-1 = 9.8E-8/py	BWR-1 = 9.8E-8/py
PWR-2 = 8.9E-6/py	BWR-2 = 3.0E-5/py
PWR-3 = 2.6E-5/py	BWR-3 = 1.2E-6/py
PWR-4 = 8.6E-8/py	BWR-4 = 1.4E-6/py
PWR-5 = 4.1E-7/py	
PWR-6 = 6.5E-5/py	
PWR-7 = 3.1E-5/py	

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

$\bar{F}^*_{PWR}$ = 7.3E-5/py

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

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

W^*_{PWR} = 183 man-rem/py

W^*_{BWR} = 221 man-rem/py

TABLE 2. (contd)

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

$\Delta\bar{F}_{PWR}$ = 9.0E-6/py

$\Delta\bar{F}_{BWR}$ = 4.1E-6/py

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

ΔW_{PWR} = 24 man-rem/py

ΔW_{BWR} = 29 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
9.7E+4	2.5E+7	0

TABLE 3. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Short-Term Accident Analysis and Procedures Revision (Confirmatory Analysis of Selected Transients by NRC)[I.C.1(4)]

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

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

PWR = (19,860 man-rem)(9.0E-6/py) = 0.18 man-rem/py

BWR = (19,860 man-rem)(4.1E-6/py) = 0.082 man-rem/py

TABLE 3. (contd)

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
570	3.1E+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.

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

Dose estimated directly (see Step 1D)

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

With improved operating guidelines and upgraded control systems, it is felt that the annual operational doses can be reduced. Table 1 gives an operational reduction of 5% as a best estimate. Assuming a range of 300 to 500 man-rem per year for routine operational exposure of an average plant, the best-estimate annual operational dose increase is

$$D_o = -\left(\frac{300 + 500 \text{ man-rem/py}}{2}\right)(0.05) = -20 \text{ man-rem/py}$$

(Negative sign indicates reduction.)

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

$N\bar{T}D_o$ = -7.6E+4 man-rem (Reduction)

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
-7.6E+4	-2.5E+4	-2.3E+5

(Negative signs indicate reductions.)

3.0 SAFETY ISSUE COSTS

3.1 INDUSTRY COST

It is estimated that 30 man-wk per plant are needed for industry to implement the SIR. This covers three activities by industry: 1) review the NRC results and determine how the individual facility is affected; 2) modify and upgrade procedures and/or systems appropriately; and 3) familiarize operations staff with upgrades. Using the industry rate of $2270/man-wk, this cost is $68,000/plant. It is assumed that plants will not be shut down except as scheduled, therefore extra cost for replacement power is not included. The labor required for operation and maintenance is estimated at 7 man-wk/py. This gives a cost of $16,000/py. Results of the industry cost analysis are shown in Steps 4-12 of Table 4.

3.2 NRC COST

Using eight transient scenarios for each generic type of NSSS leads to a total of 32 cases to be analyzed by NRC/NRR for its independent verifications. Using a resource requirement of 0.5 man-yr and 20 computer hours for each case leads to a total of 16 man-yr and 640 computer hours. At $1.0E+5 per man-yr[a] and $300 per computer hour, labor cost is estimated at $1.6E+6 and computer cost at $1.9E+5. Total cost for the analysis portion is $1.8E+6.

The requirement for the NRC to implement the SIR is estimated to be 1.1 man-yr over all plants, which is equal to $110,000. The annual NRC effort to review all the licensees' documentations on follow-up activities is estimated to be 1 man-yr at the beginning. This is expected to reduce to zero at the end of plant life. Therefore, an average of 0.5 man-yr is used for cost analysis, or $50,000 per year. Over the remaining lifetimes of the completed and planned reactors, the cost if $1.4 million. Results of the NRC cost analysis are shown in Steps 13-21 of Table 4.

TABLE 4. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Short-Term Accident Analysis and Procedure Revision (Confirmatory Analysis of Selected Transients by NRC)[I.C.1(4)]

(a) Same rate is assumed whether work is done by the NRC or by NRC contractors.

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
All:	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)(9.0E-6/py) = \$1.5E+4/py

BWR = (\$1.65E+9)(4.1E-6/py) = \$6.8/E+3/py

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

Best Estimate	Upper Bound	Lower Bound
\$4.7E+7	\$2.5E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

Labor = 30 man-wk/plant

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

Labor = (30)(\$2270) = \$6.8E+4/plant

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

NI = (134)(\$6.8E+4) = \$9.1E+6

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

Labor = 7 man-wk/py

TABLE 3. (contd)

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

Labor = (7)($2270) = $1.6E+4/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)]($1.6E+4) = $6.0E+7

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$6.9E+7	$9.9E+7	$3.9E+7

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Labor = 16 man-yr

Computer Time = 640 hr

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

Labor[(a)] = $1.6E+6

Computer = $1.9E+5 (assuming $300/computer-hr)

C_D = $1.8E+6

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

Cost estimated directly in Step 17.

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

Cost estimated directly in Step 17.

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

NC = (1.1 man-yr)($1.0E+5/man-yr) = $1.1E+5

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

Cost estimated directly in Step 20

(a) NRC staff and/or NRC contractor.

TABLE 3. (contd)

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

Cost estimated directly in Step 20.

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

$$N\bar{T}C_O = (28.3\ \text{yr})(0.5\ \text{man-yr/yr})(\$1.0E+5/\text{man-yr}) = \$1.4E+6$$

21. Total NRC Cost (SN):

Best Estimate	Upper Bound	Lower Bound
$3.3E+6	$4.4E+6	$2.2E+6

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: II.B.6, Risk Reduction for Operating Reactors at Sites with High Population Densities

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

Concern exists over the potential for above-average risk due to accidents at reactor sites located near regions of high population densities. Risk assessments have been completed for the three key sites (Zion, Limerick, and Indian Point). Issue resolution is presumed to be the implementation of fixes to lower the frequencies of dominant accident sequences.

AFFECTED PLANTS BWR: Operating = 0 Planned = 0
PWR: Operating = 1 Planned = 0

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 5.1E+4

OCCUPATIONAL DOSES:

SIR Implementation = 26

SIR Operation/Maintenance = 1.0

Total of Above = 27

Accident-Avoidance = 130

COST RESULTS ($$10^6$)

INDUSTRY COSTS:

SIR Implementation = 4.0

SIR Operation/Maintenance = 0.061

Total of Above = 4.1

Accident-Avoidance = 11

NRC COSTS:

SIR Development = 0

SIR Implementation Support = 0.027

SIR Operation/Maintenance Review 0.0061

Total of Above = 0.033

RISK REDUCTION FOR OPERATING REACTORS AT SITES WITH HIGH POPULATION DENSITIES

TMI ACTION PLAN TASK II.B.6

In May 1980, the NRC established TMI Action Plan (TAP) Task II.B, "Consideration of Degraded or Melted Cores in Safety Review" (US NRC 1980). As part of this task, sub-task II.B.6, "Risk Reduction for Operating Reactors at Sites with High Population Densities," was defined.

1.0 SAFETY ISSUE DESCRIPTION

The original description of TAP Task II.B.6 is as follows:

> "To ensure that the public health and safety is adequately protected, NRC is undertaking a review of operating reactors located in areas of high population density to determine what additional measures and/or design changes can and should be implemented that will further reduce the probability of a severe reactor accident and will reduce the consequences of such an accident by reducing the amount of radioactive releases and/or by delaying any radioactive releases, and thereby provide additional time for evacuation near the sites." (US NRC 1980)

The Indian Point (IP) and Zion sites were identified as being located near regions of high population density. Risk studies were proposed for the four plants at these two sites (the "ZIP" studies). Subsequently, the Limerick power station was also selected for a similar risk study. All three of these sites fell into the category "Substantially Above Average" with respect to the median value of the "Site Population Factor" (SPF) [Dircks 1981]. The SPF weighted population by site proximity using average meteorological conditions and by plant power level. These three sites had SPF values 10-15 times that of the median. No other sites were identified in this category.

The Zion and Limerick studies were completed in 1981 (Commonwealth Edison Co. 1981 and Philadelphia Electric Co. 1981); the IP study was finished in 1982 (PASNY 1982). Although risk assessments of other sites have been, are being, and will be conducted for other NRC programs (e.g., Interim and National Reliability Evaluation Programs, IREP and NREP), no further risk studies are currently envisioned as part of TAP Task II.B.6. Future efforts in connection with this task will be related to reviews of the completed studies and possible implementation of site-specific fixes to reduce the risk at these sites. Currently, special hearings are being conducted to review possible design changes for IP.

ISSUE RESOLUTION

Risk can be measured in various ways, with the expected numbers of early and latent fatalities being of greatest interest in nuclear plant studies. Current NRC thinking on possible nuclear plant safety goals centers on these two risk measures along with the core-melt frequency (US NRC 1982). The Zion and Limerick plants have values for all three of these risk-related measures which are less than those for the corresponding WASH-1400 plants.

The expected numbers of early fatalities for the two IP units are both less than that for the WASH-1400 PWR. However, both the expected number of latent fatalities and the core-melt frequency for each unit exceed those for the WASH-1400 PWR. Although final values for the NRC quantitative safety goals are probably a few years away, these latter risk-related measures will be of most concern with respect to the IP site.

Risk reduction can only be achieved through actual design and/or procedural changes at a plant. While a risk assessment does not of itself achieve any reduction in risk, its results can indicate potential areas where design and/or procedural changes may generate a risk reduction. For purposes of this analysis, it is assumed that design and/or procedural changes will be implemented as a result of the risk studies performed at the high population density sites. Furthermore, it is assumed that reasonable estimates of the risk reduction, dose, and cost associated with resolution of TAP Task II.B.6 can be obtained by presuming fixes will be made at Indian Point 2 (IP2) to reduce the likelihood of those dominant accident sequences contributing the most to both the expected number of latent fatalities and the core-melt frequency as assessed in the IP2 risk study. IP2 is chosen as the representative plant because: 1) the IP site has larger risks than the Zion and Limerick sites; and 2) comparison of the risk curves for IP2 and Indian Point 3 indicates that the latent fatality risk and the core-melt frequency for the former are 5-10 and 2.5 times greater, respectively, than those for the latter.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with issue resolution are estimated in this section. The analyses are summarized in Tables 1 and 2, respectively. Analysis of the public risk reduction requires some deviation from the standardized method of PNL-4297 (Andrews 1982). This alternative approach is described in Attachment 1 to the Public Risk Reduction Work Sheet.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue;

   Risk Reduction for Operating Reactors at Sites with High Population Densities (II.B.6).

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

   For purposes of analysis, the Indian Point 2 (IP2) PWR is assumed to be affected (N=1). It has a remaining life of 27 yr.

3. Plants Selected for Analysis:

   IP2 (see Attachment 1).

4-5. Steps Related to Affected Parameters and Base-Case Values:

   These steps are not utilized for this issue (see Attachment 1).

6. Affected Accident Sequences and Base-Case Frequencies:

   Seismic Sequence (SS) = 1.4E-4/py (plant-year)

   Fire Sequence (FS) = 1.4E-4/py

   (See Attachment 1; these values are taken directly from the IP risk assessment.)

7. Affected Release Categories and Base-Case Frequencies:

   This step is not utilized for this issue (see Attachment 1).

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

   2.8E-4/py
   (This value is taken directly from the IP risk assessment.)

9. Base-Case, Affected Public Risk (W): 2,100 man-rem/py

   2,100 man-rem/py
   (This value assumes an average dose factor of 7.4E+6 man-rem, see Attachment 1.)

10. Adjusted-Case Values for Affected Parameters:

   This step is not utilized for this issue.

TABLE 1. (contd)

11. Affected Accident Sequences and Adjusted-Case Frequencies:

SS = 1.4E-5/py

FS = 1.4E-5/py

(See Attachment 1)

12. Affected Release Categories and Adjusted-Case Frequencies:

This step is not utilized for this issue.

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

2.8E-5/py

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

210 man-rem/py

(This value assumes an average dose factor of 7.4E+6 man-rem, see Attachment 1.)

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

2.5E-4/py

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

1,900 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
5.1E+4	1.7E+6	0

ATTACHMENT 1

Risk reduction estimates for resolution of TAP Task II.B.6 are obtained by presuming fixes will be made at IP2 to reduce the likelihood of those dominant accident sequences contributing the most to both the expected number of latent fatalities and the core-melt frequency as assessed in the IP2 risk study. Based on this study, each of the following two accident sequences is estimated to contribute 30% to the core-melt frequency and 43% to the expected number of latent fatalities:

1. Seismic Sequence (SS). An earthquake causes failure of the IP2 control building due to impact with the control building of Indian Point 1 (no longer operating). Coupled with seismic-induced failure of the ceramic insulators on the offsite power transformers, this leads to a loss of control and AC power at IP2. A small LOCA and turbine trip result along with complete loss of containment mitigative systems.

2. Fire Sequence (FS). A large-exposure fire occurs in the electrical tunnel or the switchgear room of IP2, damaging power and control cables. A small LOCA results due to failure of the reactor coolant pump seals. Power to the safety injection pumps, containment spray pumps, and fan coolers is also lost.

The following two fixes are assumed to reduce the likelihood of core-melt (and, thus, the expected number of latent fatalities) from these sequences.

Seismic Sequence Fix

The SS occurs only if the unit 2 control building is damaged by impact with the unit 1 control building and the ceramic insulators on the offsite power transformers fail. Little can be done to prevent the latter. However, introducing some sort of seismic damper between the two units' control buildings can reduce the likelihood of the former. These buildings are very close together (a few inches), so the damper need not be excessively thick. Thus, the placement of a seismic damper between the two buildings is the assumed fix for the SS.

Fire Sequence Fix

The FS occurs if a large-exposure fire takes place in either the electrical tunnel or the switchgear room at a location from which it can damage vital power and control cables. Neither room is equipped with an automatic extinguishing system, although automatic detectors and manual extinguishers are readily available. However, even the presence of an automatic system would not reduce the likelihood of a large-exposure fire significantly under the modeling assumptions used in the risk study.

A more effective fix would be the routing of redundant power and control cables for the vital systems. While this does not reduce the likelihood of a large-exposure fire in either the electrical tunnel or the switchgear room, it does reduce the likelihood of core-melt resulting from such a fire. Thus, routing of redundant power and control cables is the assumed fix for the FS.

Public Risk and Core-Melt Frequency Estimates

It is assumed that the public risk and core-melt frequency reductions can be estimated as decreases in the likelihood of the two dominant accident sequences SS and FS. Each has a base-case frequency of 1.4E-4/py (plant-year) as given in the original IP study. It is assumed that resolution of the issue (via the assumed fixes) has the effect of lowering the frequency of each sequence (SS and FS) by a factor of ten. This yields an adjusted-case frequency of 1.4E-5/py for each sequence. Given the base-case, affected core-melt frequency for IP2 of 2.8E-4/py (from the SS and FS sequences), the adjusted-case, affected core-melt frequency becomes (0.1)(2.8E-4/py) = 2.8E-5/py.

The IP2 risk assessment employs a set of PWR release categories different from that defined in WASH-1400 (and used in PNL-4297). Since the public dose factors associated with the IP2 release categories are not conveniently extractable from the risk study, the following definition of the overall plant public risk is used:

$$W_o = \bar{F}_o \bar{R}$$

$\bar{F}_o$ = overall plant core-melt frequency (1/py)

$\bar{R}$ = average public dose factor (man-rem)

The expected public dose for IP2 can be found from the complementary cumulative density function for whole-body dose, which gives a base-case, overall plant public risk of W_o = 3,500 man-rem/py. Since $\bar{F}_o$ = 4.7E-4/py, the average public dose factor for IP2 is $\bar{R}$ = 7.4E+6 man-rem. Thus, the base-case and adjusted-case, affected public risks become:

$$W = (2.8E\text{-}4/py)(7.4E{+}6 \text{ man-rem}) = 2{,}100 \text{ man-rem/py}$$

$$W^* = (2.8E\text{-}5/py)(7.4E{+}6 \text{ man-rem}) = 210 \text{ man-rem/py}$$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Risk Reduction for Operating Reactors at Sites with High Population Densities (II.B.6).

2. Affected Plants (N):

   IP2 PWR (N=1)

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

   27 yr.

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

   (2.5E-4/py)(19,900 man-rem) = 5.0 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
130	900	0

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

   PNL staff with expertise in nuclear reactor decommissioning estimate that 20%-30% of the total cost of decommissioning can typically be attributed to dedicated staff labor. In addition, where work in radiation zones is necessary, worker productivity is roughly 75%. Assuming that the dedicated staff labor in decommissioning involves work primarily in the plant, it follows that approximately (0.25)(0.75) ≈ 0.2, or 20%, of the total cost of decommissioning can be attributed to dedicated staff labor in radiation zones.

   The SS fix involves work outside the control building and is not expected to involve radiation exposure. The FS fix involves work inside the plant and will presumably involve some radiation exposure. For this analysis, it is assumed that the percentage of the total decommissioning cost attributable to dedicated staff labor in radiation zones (~20%) is representative of the contribution of dedicated staff labor in radiation zones to the total cost of the FS fix. Thus, from the implementation cost of the FS fix ($3E+6/plant, from Step 7 of Table 3), the amount of the labor in radiation zones is estimated to be (using $1.0E+5/man-yr):

   Labor = (0.2)($3E+6/plant)/($1.0+5/man-yr)

   = 6.0 man-yr/plant

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

An average radiation field of 2.5 mR/hr is assumed (work should be primarily outside containment).

$$D = (.0025\ R/hr)(6.0\ man\text{-}yr/plant)(44\ man\text{-}wk/man\text{-}yr)(40\ man\text{-}hr/man\text{-}wk)$$
$$= 26.4\ man\text{-}rem/plant$$

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

26.4 man-rem

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

An additional 1 man-wk/py of labor is assumed for normal plant maintenance and inspection of the seismic damper and rerouted cables (see Step 9 of Table 3). Assuming half of this is alloted to the cables and 75% involves work in radiation zones, the amount of labor in radiation zones becomes:

$$(1\ man\text{-}wk/py)(0.5)(0.75) = 0.38\ man\text{-}wk/py$$

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

Again assuming a 2.5 mR/hr average radiation field:

$$D_0 = (.0025\ R/hr)(0.38\ man\text{-}wk/py)(40\ man\text{-}hr/man\text{-}wk)$$
$$= .038\ man\text{-}rem/py$$

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

$$(27)(.038) = 1.0\ man\text{-}rem$$

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
27	82	9.1

3.0 SAFETY ISSUE COSTS

The industry and NRC costs associated with issue resolution are estimated in this section. The analysis is summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Risk Reduction for Operating Reactors at Sites with High Population Densities (II.B.6)

2. Affected Plants (N):

   IP2 PWR (N=1)

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

   27 yr

Industry Costs (Steps 4 through 12)

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

   (2.5E-4/py)($1.65E+9) = $4.1E+5/py

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

Best Estimate	Upper Bound	Lower Bound
$1.1E+7	$7.5E+7	0

6. Per-Plant Industry Resources for SIR Implementation:

   Labor and equipment costs for implementing the two fixes assumed for issue resolution are estimated directly in the next step. Installing the seismic damper would involve work outside the control building and, therefore, not require any plant down-time. Cable routing could be performed during scheduled outages, so it too would not require any additional plant down-time.

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

   Installation of Seismic Damper = $1E+6/plant

   Routing of Power and Control Cables = $3E+6/plant

   (These cost estimates are based on contacts with personnel in the nuclear industry who are cognizant of the range of costs that would be typical for such fixes on a plant layout similar to IP2. Both estimates include labor and equipment costs.)

TABLE 3. (contd)

The above fixes will presumably require a class III license amendment at a cost of $4,000 (10 CFR 170.22).

I = $1E+6/plant + $3E+6/plant + $4,000/plant

= $4E+6/plant

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

$4E+6

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

The seismic damper and the newly-routed cables should be included as part of normal plant maintenance and inspection procedures. An additional 1 man-wk/py of labor is assumed.

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

(1 man-wk/py)($2,270/man-wk) = $2,270/py

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

(27)($2270) = $6.1E+4

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$4.1E+6	$6.1E+6	$2.1E+6

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Review of the Zion, IP, and Limerick risk studies should be complete prior to fiscal year 1983. No new sites are currently anticipated to require analysis as part of this issue. Any NRC resources expended with respect to design and/or procedural fixes at individual plants will be in support of SIR implementation. Thus, SIR development is assumed to be essentially complete.

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

Zero

TABLE 3. (contd)

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

As evidenced by the special hearings being conducted to review possible design changes for IP, further NRC labor may be necessary in connection with possible fixes at the high population sites. As an estimate of this potential amount of labor, 12 man-wk/plant are assumed necessary for NRC to support implementation of the two proposed fixes at IP2.

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

(12 man-wk/plant)($2,270/man-wk) = $2.7E+4/plant

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

$2.7E+4

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

The seismic damper and the newly-routed cables should be included as part of NRC's routine inspection. An additional 0.1 man-wk/py of labor is assumed.

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

(0.1 man-wk/py)($2,270/man-wk) = $227/py

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

(27)($227) = $6,100

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$3.3E+4	$4.7E+4	$1.9E+4

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: II.C.2, Continuation of IREP (NREP)

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

This issue proposes extending the ongoing IREP effort to the eleven plants in the first group of phase III of the SEP. The assumed resolution involves two parts: 1) performance of an NREP study at each of these plants by the utilities and 2) implementation of fixes to reduce the likelihood of the most dominant core-melt sequences at each plant judged to have an overall core-melt frequency which is "too high."

AFFECTED PLANTS[a]			
	BWR:	Operating = N/A	Planned = N/A
	PWR:	Operating = N/A	Planned = N/A

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 3.8E+4

OCCUPATIONAL DOSES:

SIR Implementation =	54
SIR Operation/Maintenance =	5.4
Total of Above =	59
Accident-Avoidance =	230

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:

SIR Implementation =	25
SIR Operation/Maintenance =	1.4
Total of Above =	26
Accident-Avoidance =	19

NRC COSTS:

SIR Development =	0.10
SIR Implementation Support =	3.5
SIR Operation/Maintenance Review =	0.14
Total of Above =	3.7

(a) Eleven backfit LWRs are assumed to be affected. There is no breakdown into BWRs and PWRs in this analysis.

CONTINUATION OF IREP (NREP)

ISSUE II.C.2

1.0 SAFETY ISSUE DESCRIPTION

This safety issue is concerned with continuing the Interim Reliability Evaluation Program (IREP) studies to cover all remaining operating reactors that were not involved in the initial IREP studies or for which no risk/reliability assessment has been performed. The details of IREP continuation (known as NREP, the National Reliability Evaluation Program) will be based on the results of the initial IREP studies. Also, consideration will be given to expanding the coverage to include plants under construction, in which the design is sufficiently final to allow a meaningful evaluation (i.e., plants awaiting an operating license or with well-developed standardized designs).

The original IREP scope included a provision for recommending plant modifications to reduce the likelihood of especially high-risk accident sequences uncovered in the study. As stated in the TMI Action Plan (US NRC 1980):

> "Following each plant study in the IREP program, a set of plant-specific recommended alterations in design, procedures, and technical specifications will be prepared, as necessary to reduce the expected frequency of particularly high-risk accident sequences and to rectify any identified safety weaknesses."

This aspect of IREP will presumably carry over into the NREP effort.

One potential modification in NREP is a shift from the NRC to the utilities as the performers of the plant-specific studies. This appears likely if NREP is applied to a large number of plants.

PROPOSED RESOLUTION

For purposes of this analysis, the proposed resolution to this safety issue is assumed to consist of two parts: 1) performance of an NREP study by the utility at each designated plant currently without a risk/reliability assessment, and 2) implementation of some hardware/procedural fix which will significantly lower the frequency of the most dominant accident sequence with respect to core-melt at each plant where the core-melt frequency is determined to be "too high."

At this point, definition of "too high" can only be arbitrary since no safety goal exists. However, reviews of the latest NRC proposal on safety goals (US NRC 1982) indicate that core-melt frequency will be an important goal (along with the early and latent risk goals). Thus, it is assumed that "too high" will be judged with respect to some future safety goal or guideline on the core-melt frequency. This assumption is convenient since NREP studies are currently intended only to estimate core-melt frequency, not risk (although the extension can be made).

AFFECTED PLANTS

Although NREP may ultimately encompass all nuclear power plants, discussions with NRC indicate that current plans are to initially involve only the eleven plants selected for the first group of phase III of the Systematic Evaluation Program (SEP). Thus, it is assumed in this analysis that an NREP study will be performed at eleven backfit light water reactors (LWRs), with implementation of hardware/procedural fixes enacted only at those plants where the core-melt frequency is determined to be "too high." As developed in Attachment 1 to the Public Risk Reduction Work Sheet, four of these eleven LWRs are assumed to exhibit core-melt frequencies deemed "too high." Thus, while costs for performing NREP studies will be incurred at all eleven plants, public risk reduction will be realized only at the four LWRs implementing fixes as a result of their studies.

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. Note that the analysis is conducted for a representative LWR, rather than for a representative PWR and BWR, a consequence of the data base employed for this issue analysis.[a]

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Continuation of IREP (NREP) (II.C.2)

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

   Four of eleven backfit LWRs assumed to perform NREP studies are further assumed to implement hardware/procedural fixes as a result of these studies. Such fixes are assumed to be implemented in 1985. Therefore, for public risk reduction estimation:

   N = 4 backfit LWRs with $\bar{T}$ = 23.9 yr

(a) See Attachment 1.

TABLE 1. (contd)

3. Plants Selected for Analysis:

A hypothetical backfit LWR with a core-melt frequency of 3.3E-4/py and a most dominant accident sequence which contributes 39% of this frequency is assumed to be representative of the four affected backfit LWRs.[a]

4-7. Steps Related to Affected Parameters, Accident Sequences and Release Categories and Their Base-Case Values:

The base-case, affected core-melt frequency is estimated directly in the next step. Thus, these steps are omitted.

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

$\bar{F}$ = 3.3E-4/py[a]

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

W = 1090 man-rem/py[a]

10-12. Steps Related to Affected Parameters, Accident Sequences and Release Categories and Their Adjusted Case Values:

The adjusted-case, affected core-melt frequency is estimated directly in the next step. Thus, these steps are omitted.

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

$\bar{F}^*$ = 2.1E-4/py[a]

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

W* = 690 man-rem/py[a]

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

$\Delta\bar{F}$ = 1.2E-4/py

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

ΔW = 400 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
3.8E+4	3.1E+6	0

(a) See Attachment 1.

ATTACHMENT 1

Of the 15 risk/reliability studies currently available for LWRs, five plants have been assessed as having core-melt frequencies in excess of 1E-4/py: Indian Point Units 2 and 3 (IP2 and IP3), Crystal River 3 (CR3), Calvert Cliffs 2 (CC2), and Browns Ferry 1 (BF1). The table below lists the core-melt frequencies, frequencies of the most dominant accident sequences, and percent contributions to the core-melt frequencies from these most dominant sequences for these five LWRs (PASNY 1982; Garcia 1981; Hatch 1982; Mays 1982):

Plant	Core-Melt Frequency (1/py)	Dominant Seq. Freq. (1/py)	Dominant % Contribution
IP2	4.7E-4	1.4E-4	30
IP3	1.9E-4	8.2E-5	43
CR3	3.7E-4	1.7E-4	46
CC2[a]	4.0E-4	9.6E-5	25
BF1	2.0E-4	9.7E-5	49

(a) The values given here assume AFWS upgrades scheduled for 1982 are implemented. See the CC2 RSSMAP study for further discussion (Hatch 1982).

For these five plants, the average core-melt frequency is 3.3E-4/py and the average contribution of the most dominant sequence is 39%. For this analysis, it is assumed that these five plants have core-melt frequencies deemed "too high" (a determination based loosely on the proposed core-melt frequency safety goal of 1E-4/py from NUREG-0880). Thus, of the fifteen plants whose risks/reliabilities have so far been assessed, one-third exhibit core-melt frequencies which are "too high." Assuming this fraction to be representative of the eleven LWRs for which NREP studies will be conducted, it follows that $11/3 \approx 4$ of these plants will exhibit core-melt frequencies deemed "high enough" to warrant hardware/procedural fixes to lower the frequency of the most dominant accident sequence (with respect to core-melt) at each.

These four LWRs are assumed to each have a base-case, affected core-melt frequency of 3.3E-4/py, of which 39% is due to the most dominant accident sequence. Assuming issue resolution (through hardware/procedural fixes) reduces the likelihood of the most dominant sequence's frequency by 90%, the adjusted-case, affected core-melt frequency becomes:

$$\bar{F}^* = (3.3\text{E-}4/\text{py})[1-(0.90)(0.39)]$$
$$= 2.1\text{E-}4/\text{py}$$

To obtain the base and adjusted-case, affected public risks, the overall risk is written as follows:

$$W_0 = \bar{F}_0 R_0$$

where W_0 = overall risk

$\bar{F}_0$ = overall core-melt frequency

R_0 = average dose factor

Denoting the number of plants as N and their average remaining lives as $\bar{T}$, the average dose factor for an LWR can be estimated as follows:

$$(R_0)_{LWR} = \frac{(N\bar{T}W_0)_{PWR} + (N\bar{T}W_0)_{BWR}}{(N\bar{T}\bar{F}_0)_{PWR} + (N\bar{T}\bar{F}_0)_{BWR}}$$

Based on Appendices A-D of PNL-4297 (Andrews 1983),

$$(R_0)_{LWR} = 3.3E+6 \text{ man-rem}$$

where N = 90 (PWR) and 44 (BWR)

$\bar{T}$ = 28.8 yr (PWR) and 27.4 yr (BWR)

W_0 = 207 man-rem/py (PWR) and 250 man-rem/py (BWR)

$\bar{F}_0$ = 8.2E-5/py (PWR) and 3.7E-5/py (BWR)

Thus, for this issue,

W = (3.3E-4/py)(3.3E+6 man-rem)

= 1090 man-rem/py

W* = (2.1E-4/py)(3.3E+6 man-rem)

= 690 man-rem/py

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Continuation of IREP (NREP) (II.C.2)

2. Affected Plants (N):

   Four backfit LWRs (see Step 2 of Table 1). No occupational dose from accidents will be avoided unless a plant implements hardware/procedural fixes. Likewise, occupational dose will be accumulated from SIR implementation and operation/maintenance only if hardware fixes are imposed (see Step 6).

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

   $\bar{T}$ = 23.9 yr (see Step 2 of Table 1).

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

   $\Delta(\bar{FD}_R)$ = (19,900 man-rem)(1.2E-4/py) = 2.4 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
230	3800	0

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

   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. In addition, where work in radiation zones is necessary, worker productivity is roughly 75%. Assuming that the dedicated staff labor in decommissioning involves work primarily in the plant, it follows that approximately (0.25)(0.75) ≈ 0.2, or 20%, of the total cost of decommissioning can be attributed to dedicated staff labor in radiation zones.

   For this analysis, it is further assumed that this percentage is representative of the contribution of dedicated staff labor in radiation zones to the cost of a hardware fix. For an average implementation cost of \$2.0E+6/plant (see Step 8 of Table 3), the amount of labor in radiation zones is estimated to be (using \$1.0E+5/man-yr):

   Labor = (0.2)(\$2.0E+6/plant)/(\$1.0E+5/man-yr)

   = 4 man-yr/plant

However, since some plants might perform fixes that are more procedurally oriented (presumably involving little work in radiation zones), the above estimate is assumed to apply only to three of the four affected plants (all backfit).

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

An average exposure rate of 2.5 mR/hr is assumed to apply to the implementation labor.

$$D = (0.0025 \text{ R/hr})(4 \text{ man-yr/plant})$$
$$(44 \text{ man-wk/man-yr})(40 \text{ man-hr/man-wk})$$
$$= 18 \text{ man-rem/plant}$$

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

$$ND = (3 \text{ plants})(18 \text{ man-rem/plant}) = 54 \text{ man-rem}$$

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

It is assumed that 75% of the labor associated with operation/maintenance of a hardware fix (1 man-wk/py, see Step 9 of Table 3) will involve work in radiation zones. Thus,

Labor = 0.75 man-wk/py

As for implementation, this estimate is assumed to apply only to three of the four affected plants.

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

Again, a 2.5 mR/hr radiation field is assumed.

$$D_o = (0.0025 \text{ R/hr})(0.75 \text{ man-wk/py})$$
$$(40 \text{ man-hr/man-wk})$$
$$= 0.075 \text{ man-rem/py}$$

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

$$N\bar{T}D_o = (3 \text{ plants})(23.9 \text{ yr})(0.075 \text{ man-rem/py})$$
$$= 5.4 \text{ man-rem}$$

TABLE 2. (contd)

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
59	180	20

3.0 SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. Results are summarized in Table 3. Note that the cost of performing an NREP study will be incurred at all eleven backfit LWRs in phase III of the SEP. However, the cost of implementing, operating, and maintaining a hardware/procedural fix will be incurred only at four of these plants.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Continuation of IREP (NREP) (II.C.2)

2. Affected Plants (N):

   For SIR implementation, operation, and maintenance, all eleven backfit LWRs in phase III of the SEP are presumed to be affected. For industry cost savings due to accident-avoidance, only those four plants imposing hardware/procedural fixes will be affected.

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

   $\bar{T}$ = 23.9 yr (see Step 2 of Table 1).

Industry Costs (Steps 4 through 12)

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

   $\Delta(\bar{FA})$ = (\$1.65E+9)(1.2E-4/py) = \$2.0E+5/py

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

Best Estimate	Upper Bound	Lower Bound
\$1.9E+7	\$3.1E+8	0

TABLE 3. (contd)

6. Per-Plant Industry Resources for SIR Implementation:

Cost is estimated directly in next step.

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

Based on previous IREP experience, it is assumed that the cost to each utility to perform an NREP study will be \$1.5E+6/plant (excluding an in-depth systems interaction study--this is considered in Issue II.C.3, "Systems Interaction"). The cost to each utility to implement a hardware/procedural fix is assumed to be \$2.0E+6/plant (applicable to four of the eleven backfit LWRs). This estimate is based on the analysis of Issue II.B.6, "Risk Reduction for Operating Reactors at Sites with High Population Densities," in which it was assumed that the top two dominant accident sequences for the representative plant (IP2) could be reduced in likelihood by a factor of ten at a combined cost of \$4.0E+6. Since both sequences were equally dominant, it is assumed that their average cost (\$2.0E+6) is typical of that associated with implementing a hardware/procedural fix to reduce the likelihood of the most dominant accident sequence. In addition, a class III license amendment fee of \$4000 will presumably be incurred at each of the four plants implementing a fix.

Affected Plants	I (\$/plant)
7 backfit LWRs performing NREP studies only	1.5E+6
4 backfit LWRs performing NREP studies, implementing fixes, and amending licenses	1.5E+6 2.0E+6 + 4.0E+3 3.5E+6

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

$$NI = (7 \text{ plants})(\$1.5E+6/\text{plant}) + (4 \text{ plants})(\$3.5E+6/\text{plant})$$
$$= \$2.45E+7$$

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

Update of an existing NREP study for plant modifications is assumed to require 2 man-wk/py at all eleven backfit LWRs. Operation/maintenance of a hardware/procedural fix is assumed to require 1 man-wk/py at those four plants implementing such (for procedural fixes, operation/maintenance may take the form of reviews and updates).

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

Affected Plants	Labor (man-wk/py)
7 backfit LWRs updating NREP studies only	2
4 backfit LWRs updating NREP studies and operating/maintaining fixes	2 + 1 = 3

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

Affected Plants	I_o ($/py)
7 backfit LWRs--NREP only	4540
4 backfit LWRs--NREP and fixes	6810

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

$$N\bar{T}I_o = [(7\ \text{plants})(\$4540/\text{py}) + (4\ \text{plants})(\$6810/\text{py})](23.9\ \text{yr})$$
$$= \$1.4\text{E}{+}6$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$2.6E+7	$3.8E+7	$1.4E+7

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

The NREP studies will be similar to those for IREP except that the industry is assumed to perform the NREP studies. Thus, only one additional man-year of NRC labor is assumed for further development of guidelines.

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

$$C_D = (1\ \text{man-yr})(\$1.0\text{E}{+}5/\text{man-yr})$$
$$= \$1.0\text{E}{+}5$$

TABLE 3. (contd)

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

To support SIR implementation, the following amounts of NRC labor are assumed:

a. Monitor and review NREP study = 3 man-yr/plant (all eleven backfit LWRs)

b. Monitor and review hardware/procedural fix = 0.5 man-yr/plant (four of eleven backfit LWRs)

Thus,

Affected Plants	Labor (man-yr/plant)
7 backfit LWRs-- NREP only	3.0
4 backfit LWRs-- NREP and fixes	3.0 + 0.5 = 3.5

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

Affected Plants	C ($/plant)
7 backfit LWRs-- NREP only	3.0E+5
4 backfit LWRS-- NREP and fixes	3.5E+5

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

NC = (7 plants)($3.0E+5/plant) + (4 plants)($3.5E+5/plant)

= $3.5E+6

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

To review SIR operation/maintenance, the following amounts of NRC labor are assumed:

a. Review NREP updates = 0.2 man-wk/py (all eleven backfit LWRs)

b. Inspect/review hardware/procedural fix = 0.1 man-wk/py (four of eleven backfit LWRs)

TABLE 3. (contd)

Thus,

Affected Plants	Labor (man-wk/py)
7 backfit LWRs--NREP only	0.2
4 backfit LWRs--NREP and fixes	0.2 + 0.1 = 0.3

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

Affected Plants	C_o (\$/py)
7 backfit LWRs--NREP only	454
4 backfit LWRs--NREP and fixes	681

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

$$\bar{NTC}_o = [(7\ \text{plants})(\$454/\text{py}) + (4\ \text{plants})(\$681/\text{py})](23.9\ \text{yr})$$
$$= \$1.4E{+}5$$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$3.7E+6	\$5.5E+6	\$2.0E+6

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: II.C.3/A-17, Systems Interaction

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

The potential for adverse systems interactions arises from hidden dependencies between systems. A systematic approach for systems interaction analysis has been proposed for demonstration at four of the eleven plants in the first group of SEP-phase III in conjunction with NREP. Hardware/procedural fixes to reduce the potential for any adverse interactions identified will be implemented if required under existing NRC regulations.

AFFECTED PLANTS[a]

BWR:	Operating = N/A	Planned = N/A
PWR:	Operating = N/A	Planned = N/A

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	4700
OCCUPATIONAL DOSES:	
SIR Implementation =	520
SIR Operation/Maintenance =	39
Total of Above =	560
Accident-Avoidance =	29

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	3.9
SIR Operation/Maintenance =	0.43
Total of Above =	4.4
Accident-Avoidance =	2.4
NRC COSTS:	
SIR Development =	0.01
SIR Implementation Support =	0.34
SIR Operation/Maintenance Review =	0.043
Total of Above =	0.40

(a) Four backfit LWRs are assumed to be affected. There is no breakdown into BWRs and PWRs in this analysis.

SYSTEMS INTERACTION

TMI ACTION PLAN TASK II.C.3
(INCORPORATING UNRESOLVED SAFETY ISSUE A-17)

As defined in NUREG-0660 (US NRC 1980), TMI Action Plan (TAP) Task II.C.3 on Systems Interaction incorporates the earlier Unresolved Safety Issue (USI) A-17 on Systems Interactions in Nuclear Power Plants (US NRC/NRR 1978). Estimates of the priority measures for Systems Interaction are given for the issue as integrated in TAP Task II.C.3 with the understanding that these include the contributions from USI A-17.

The objective of a systems interaction analysis is to provide assurance that the independent functioning of safety systems is not jeopardized by preconditions that cause faults to be dependent. The NRC systems interaction program was initiated because the design, analysis, installation, operation, and maintenance of systems are frequently the responsibilities of teams of engineers with functional specialties. Experience at operating plants has led to questions of whether the work of these specialists is sufficiently integrated to minimize the potential for adverse interactions among systems. Some adverse events that occurred in the past might have been prevented if the teams had assured the necessary independence of safety systems under all conditions of operation.

The concern over systems interactions was first documented explicitly by the ACRS in November 1974 when they requested that the NRC give "...attention to the evaluation of ... potentially undesirable interactions between systems..." from a multi-disciplinary point of view. In May 1978, USI A-17 (Systems Interactions in Nuclear Power Plants) was defined. The resulting program initially developed a methodology and applied it to an analysis of the adequacy of the NRC Standard Review Plan by analyzing Watts Bar 1 with the intent of evaluating the method (Boyd 1979). In July 1978, the ACRS recommended that, as a major part of the systems interaction program, attention be given to a review of Indian Point 3 for "systems interactions that might lead to significant degradation of safety." In addition, a special limited-scope systems interaction analysis (seismically-induced events) was performed at Diablo Canyon Units 1 and 2 (PGE 1980).

In April 1980, the Systems Interaction Branch was formed to broaden the systems interaction program beyond considerations of just one plant. This included the performance of TAP Task II.C.3 (Systems Interaction), incorporating USI A-17. The Indian Point effort was included as a plant-specific part of II.C.3. The preliminary plan for a systems interaction study was developed (Lim, May 1981), and NRC review was completed in September 1981. The final plan was finished in January 1982.

In January 1981, surveys of potential methodologies for systems interaction analysis were completed by three separate laboratories (Cybulskis 1981; Lim, January 1981; Buslik 1981). Subsequently, the Systems Interaction Branch began to develop initial regulatory guidance to provide a general approach and two illustrative procedures. Industry input was provided from April until August 1981 through the AIF Subcommittee on Systems Interactions.

In May 1981, the Systems Interaction Branch was dissolved and the program was assigned to the Reliability and Risk Assessment Branch. Development of initial regulatory guidance has continued up through the issuance of the most recent draft version in January 1982 (US NRC/NRR 1982).

1.0 SAFETY ISSUE DESCRIPTION

Current NRC regulatory guidance for conducting a systems interaction analysis at an LWR advocates the use of a systematic procedure to identify and evaluate intersystems dependencies that will fail any one of four safety-systems criteria. A safety-systems criterion is failed where a precondition exists that jeopardizes the independent functioning of safety systems. The safety systems are those that are relied upon to:

1. maintain the primary coolant inventory
2. transfer decay heat from the reactor to the ultimate heat sink,
3. render and keep the entire core subcritical, and
4. keep the Engineered Safety Features unimpaired, including those systems for the control of radioactive material.

Although the safety-systems criteria center around systems that are typically safety-grade, the actions of safety-grade systems caused by adverse influences from nonsafety-grade systems are expected to be a major part of a systems interaction analysis.

A systems interaction analysis of a plant will employ analytical methods, visual inspection, experience feedback, and experiments to identify dependencies. The systems interaction program is a vehicle to aid development of a future regulatory requirement that explicit systems interaction analyses be conducted at LWRs. Development of analysis methods has evolved to the point where pilot plant studies based on the initial guidance are deemed appropriate. The program currently proposes that system interaction pilot analyses be conducted at four of the eleven plants selected for the first group of phase III of the Systematic Evaluation Program (SEP). Since each of these plants will be performing a National Reliability Evaluation Program (NREP) study (see Issue II.C.2), the systems interaction analyses at the four pilot plants will be performed in conjunction with the NREP studies. This should streamline much of the preliminary work needed in a systems interaction analysis, thereby reducing the level of effort which would be required if the analysis were done separately.

The immediate goal of the systems interaction program is to formulate a conclusion as to the importance of systems interactions to plant safety and the effectiveness of the proposed analytical approach in identifying and evaluating these. The results of the pilot analyses will provide a measure of the benefit to be obtained from a systems interaction analysis and the cost involved, so as to support the decision whether or not to proceed with the requirement for explicit systems interaction analyses at LWRs. Subsequent NRC action will depend on this conclusion. Ultimately, systems interaction analyses could be required at all LWRs, either as separate studies or incorporated into risk analyses.

For the purposes of estimating priority measures, the resolution of the Systems Interaction issue is assumed to be two-fold:

1. Identification and evaluation of systems interactions at four of the eleven plants in the first group of phase III of the SEP using the interim regulatory guidance.

2. Reduction of the potential for any adverse systems interactions uncovered at these plants as deemed necessary under existing NRC regulations. This may involve both hardware and procedural fixes.

Hardware/procedural fixes may or may not be implemented at all four plants depending upon compliance with existing regulations. Furthermore, the fixes will most likely differ from plant to plant, resulting in varying risk reductions, doses, and costs. However, for the purposes of estimating the risk, dose, and cost associated with resolution of Issue II.C.3/A-17, it is assumed that each of the four plants implements some "typical" fix for which the risk reductions, doses, and costs are equivalent at each plant

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose due to SIR are estimated in this section. Analysis results are summarized in Tables 1 and 2 respectively. Due to deviations from the standardized procedures of PNL-4297 (Andrews 1983), attachments are provided for both tables.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

    Systems Interaction (II.C.3/A-17)

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

    As discussed in Issue II.C.2, eleven backfit LWRs in group one of the SEP-phase III are assumed to perform NREP studies. Of these plants,

TABLE 1. (contd)

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

four are assumed to conduct systems interaction analyses in conjunction with their NREP studies. These four plants are further assumed to implement hardware/procedural fixes as a result of their systems interaction analyses to reduce the potential for adverse interactions. Such fixes will presumably be implemented in 1985. Therefore, for public risk reduction estimation:

$$N = 4 \text{ backfit LWRs with } \bar{T} = 23.9 \text{ yr}$$

3. Plants Selected for Analysis:

A hypothetical backfit LWR with a core-melt frequency of 1.5E-4/py (of which 10% is attributable to systems interactions) is assumed to be representative of the four affected backfit LWRs.[a]

4-7. Steps Related to Affected Parameters, Accident Sequences and Release Categories and Their Base-Case Values:

The base-case, affected core-melt frequency is estimated directly in the next step. Thus, these steps are omitted.

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

$\bar{F}$ = 1.5E-4/py[a]

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

W = 495 man-rem/py[a]

10-12. Steps Related to Affected Parameters, Accident Sequences and Release Categories and Their Adjusted-Case Values:

The adjusted-case, affected core-melt frequency is estimated directly in the next step. Thus, these steps are omitted.

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

$\bar{F}^*$ = 1.35E-4/py[a]

(a) See Attachment 1.

TABLE 1. (contd)

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

$W^* = 446$ man-rem/py[a]

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

$\Delta\bar{F} = 1.5E-5$/py

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

$\Delta W = 49.5$ man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
4700	1.4E+6	0

ATTACHMENT 1

Because the Systems Interaction issue is a broad one perceived to affect plant safety as a whole, estimation of the public risk reduction associated with its resolution necessitates some deviation from the standardized procedure. An attempt is made to estimate the public risk reduction by considering the impacts on the public risk from four example system interactions. These are:

1. Browns Ferry fire on March 22, 1975
2. DC power failure due to common-cause battery failure
3. BWR scram failure due to slow loss of control air pressure
4. Rancho Seco instrumentation power loss on March 20, 1978.

Browns Ferry Fire on March 22, 1975

The events of this fire are well known. For detail, the reader is referred to the report issued by the NRC Office of Inspection and Enforcement (US NRC/IE 1975). Of interest here are the analyses performed in Appendix XI of WASH-1400 (US NRC 1975) and in NUREG/CR-2497 (Minarick 1982).

WASH-1400 estimates the frequency of core-melt due to a Browns-Ferry-type fire in the cable spreading room to be 1E-5/py (plant-year). This amounts to 20% of the estimated overall core-melt frequency (5E-5/py). In effect, the base-case core-melt frequency would be re-evaluated as 6E-5/py to include this fire contribution. Reducing the potential for this systems interaction lowers this frequency to 5E-5/py (the original WASH-1400 estimate) in the adjusted case. This corresponds to a reduction in core-melt frequency of 1E-5/py, or 17%. Since the public risk is proportional to the core-melt frequency (when an average dose factor is defined), the public risk reduction due to decreasing the potential for this systems interaction would also be 17%, using the WASH-1400 results.

NUREG/CR-2497 estimates the frequency of severe core damage due to the Browns Ferry fire to be 9.2E-4/py. This amounts to 20% of the overall frequency of severe core damage estimated in NUREG/CR-2497 (.0045/py). Thus, reducing the potential for this systems interaction lowers the base-case frequency of severe core damage by 20%. Assuming both the core-melt frequency and public risk to be proportional to the frequency of severe core damage, one obtains 20% reductions in both of these from decreasing the potential for this systems interaction. This agrees well with the results from WASH-1400 (a 17% reduction).

DC Power Failure Due to Common-Cause Battery Failure

A systems interaction between two DC station batteries has been envisioned which could lead to loss of both redundant DC power supply trains (Eisenhut 1978). A common-cause failure of both batteries is postulated as the initiating event. Subsequent failures are then assumed which eventually lead to a core melt. The frequency of this accident sequence is estimated to be 4E-7/py (Buslik 1981 [Appendix A]). This sequence is found to be similar in consequence to WASH-1400 PWR transient sequence TMLB', which has an estimated frequency of 3.0E-6/py.

In effect, the base-case frequency of sequence TMLB' would be re-evaluated as 3.4E-6/py to include this contribution for DC power failure. Reducing the potential for this systems interaction lowers this frequency to 3.0E-6/py (the original WASH-1400 estimate) in the adjusted case. This is a reduction of 4E-7/py, or 12%. If one assumes that this reduction is typical for all accident sequences following decrease in the potential for most systems interactions, then the resulting reduction in core-melt frequency will also be 12%. Since the public risk is proportional to the core-melt frequency, a 12% reduction in public risk due to decreasing the potential for most systems interactions is conceivable.

<u>BWR Scram Failure Due to Slow Loss of Control Air Pressure</u>

As a result of the investigation into the partial scram failure at Browns Ferry 3 on June 28, 1980 (Rubin 1980), a potential systems interaction between the control air and control rod scram systems was uncovered (Michelson 1980). A slow loss of control air pressure could result in the scram inlet and exhaust valves slowly drifting open before enough air pressure is lost to initiate a full scram. This slow opening could result in some water inleakage into the scram discharge volumes prior to scram, possibly preventing full control rod insertion if this inleakage is excessive. In fact, a possible precursor of such an event occurred at Browns Ferry 1 and 2 on August 18, 1978. The two units' control rods were observed to have drifted inward prior to a manual scram upon a massive loss of control air pressure.

The fault tree for failure of the reactor protection system developed in WASH-1400 (Figure II.6-16) is shown as Figure 1. As originally drawn, this tree does not account for the potential systems interaction discussed above. Furthermore, it does not identify failure of the scram discharge volume vent valves to remain open as a possible cause of inadequate drainage of the scram discharge volumes. This was identified as a drainage failure mechanism for one of the scram discharge volumes at Browns Ferry 3 (Rubin 1980). Thus, to reflect these additional failure mechanisms, this fault tree is modified as follows:

1. To each OR gate for "Water Enters Header," an additional input event is added entitled "Slow Loss of Control Air Pressure to Scram Inlet and Exhaust Valves Allows Excessive Inleakage to Scram Discharge Volume" (designated as event "A").

2. In place of one of the basic failures (diamonds) entitled "Trip Header Drain Line Blocked" (say drain line B) is substituted an OR gate entitled "Trip Header B Fails to Drain Adequately." As inputs to this OR gate, the following events are provided:

   i) "Trip Drain Header Line 'B' Blocked" (designated as 3PPF001P)

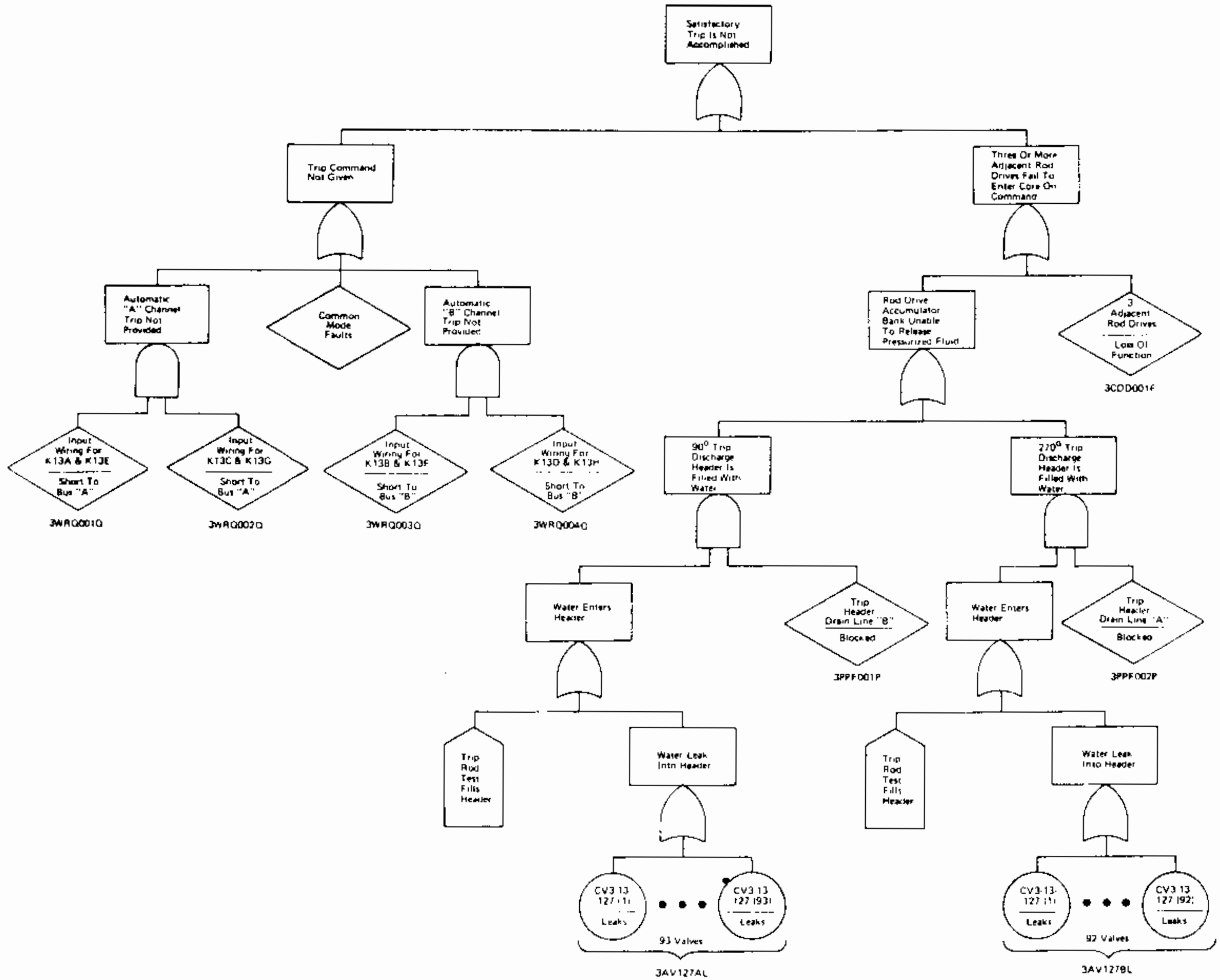

FIGURE 1. Reactor Protection System Reduced Fault Tree

ii) "Vent Valve for Scram Discharge Volume B Fails to Remain Open" (designated as "V").

The unavailabilities used in the original fault tree are:

Event	Unavailability
3WRQ001Q, 3WRQ002Q, 3WRQ003Q, 3WRQ004Q	3.6E-6
3COD001F	5.8E-6
3PPF001P, 3PPF002P	1.3E-7
3AV127AL, 3AV127BL	0.12
"Trip Test Fills Header"	N/A
"Common Mode Faults"	1.9E-6

Two new events (A and V) have been added. Their unavailabilities are estimated as follows:

1. Event A. As of August 1981, slow loss of control air pressure to the scram inlet and exhaust valves was assumed to have occurred once in 175 BWR-years of experience (Denton 1981), at Browns Ferry 1 and 2 on August 18, 1978. Since then, about one year of additional experience has been accumulated at each of the 19 BWRs susceptible to this systems interaction. No additional failures have occurred. Thus, the failure rate for event A is estimated to be:

$$\lambda(A) = 1/(175 \text{ py} + 19 \text{ py}) = .0052/\text{py}$$

The exposure time assumed for components in the control rod scram system is given in WASH-1400 as 4,300 hrs (Table II.6-9). Thus, the unavailability for event A becomes:

$$q(A) = (.0052/\text{py})(4{,}300 \text{ ph})\left(\frac{1 \text{ py}}{8{,}760 \text{ ph}}\right)$$
$$= .0025 \qquad (\text{ph} = \text{plant-hour})$$

2. Event V. The vent valve is of the type designed as "air-fluid operated" in Table III.4-1 of WASH-1400. The rate of failure to remain open in given there as:

$$\lambda(V) = 3E\text{-}7/hr$$

For the 4,300-hr exposure time, the unavailability becomes:

$$q(V) = (3E\text{-}7)(4{,}300\ hr) = .0013$$

When these unavailabilities are combined with those from the original assessment via the modified fault tree, the probability of reactor protection systems failure becomes 1.7E-4 (base case). The contribution to this probability from the minimal cut sets containing event A is 3.3E-6, or 1.9%. Thus, reducing the potential for this systems interaction lowers this probability by 1.9% in the adjusted case. Since several dominant accident sequences for the WASH-1400 BWR involve failure of the reactor protection systems (e.g., $S_2C\text{-}\gamma$ and $TC\text{-}\alpha$, Table 5-3), their frequencies will presumably decrease by 1.9% from the base to the adjusted case if this systems interaction potential is reduced. Assuming this reduction to be typical for all accident sequences following decrease in the potential for most systems interactions, one also obtains a core-melt frequency reduction of 1.9%. Since the public risk is proportional to the core-melt frequency, a 1.9% reduction in public risk due to decreasing the potential for most systems interactions is conceivable.

Rancho Seco Instrumentation Power Loss on March 20, 1978

A systems interaction between the non-nuclear instrumentation (NNI) and integrated control (IC) systems occurred at Rancho Seco on March 20, 1978. With the reactor at 70% power, an operator, while changing bulbs, dropped one into an open backlighted push button assembly. The bulb created a short-to-ground in one portion of the NNI, which caused protection circuits and devices to actuate and, as a result, removed all AC power to one channel of the NNI. Approximately 2/3 of all NNI signals were affected by this power loss resulting in erroneous information being transmitted to the control room and to the IC system.

The IC system reduced feedwater flow to zero in response to these faulty signals. The auxiliary feed pumps started; but, due to the other erroneous information, the auxiliary feed pump valves remained closed. During the nine-minute period following trip, the steam generators boiled dry. After nine-minutes the drifting signals reached the setpoint, and the auxiliary feed pump valves opened, admitting feed flow to the steam generators. Auxiliary feed flow was continued until the power could be restored to the disabled channel of the NNI. This event resulted in an overcooling of reactor coolant and exceeded reactor cooldown rates.

The frequency of severe core damage due to this systems interaction has been estimated at 5.8E-4/py in NUREG/CR-2497 (Minarick 1982). This amounts to 13% of the overall frequency of severe core damage estimated in NUREG/CR-2497 (.0045/py). Thus, reducing the potential for this systems interaction lowers the base-case frequency of severe core damage by 13%. Assuming both the core-melt frequency and public risk to be proportional to the frequency of severe core damage, one obtains 13% reductions in both of these from decreasing the potential for this systems interaction.

Estimated Percent Reduction in Public Risk

If these four systems interactions are typical of ones existing at nuclear plants, then, given the previous assumptions, one could estimate the potential reduction in public risk arising from the decrease in the likelihood for most of them to lie in a range of ~1% to 20%. Some argument could be made that the potential for systems interactions on the order of the Browns Ferry fire has been diminished by intensive programs such as seismic and fire reviews. However, one cannot assure that another Browns Ferry-type systems interaction will not occur.

The average reduction in public risk for the four examples given is (note that [20% + 17%]/2 = 18.5% is used for the Browns Ferry fire):

$$\frac{1}{4}\,(18.5\% + 12\% + 1.9\% + 13\%) = 11\%$$

Since this is around the midpoint of the estimated range for the potential reduction due to decrease in the likelihood for most systems interactions (~1% to 20%), a value of 10% is assumed as the difference between the base and adjusted-case, affected public risks (and affected core-melt frequencies) associated with resolution of this issue. Since these estimates can be applied directly to the affected public risk and core-melt frequency, there is no need to consider individual parameters or accident sequences. Thus, Steps 4-7 and 10-12 in the Public Risk Reduction Work Sheet are skipped.

Affected Core-Melt Frequency and Public Risk

Resolution of this issue is assumed to affect four of the eleven plants in the first group of the SEP-phase III. To estimate the core-melt frequency and public risk reductions due to SIR, the core-melt frequency at a typical plant in this group is approximated as follows.

Risk/reliability studies have been completed on 15 plants to date. The core-melt frequencies at these plants are as follows:

Program	Plant	Core-Melt Frequency (1/py)
Reactor Safety Study (RSS)	Surry 1	6.0E-5
	Peach Bottom 2	2.9E-5
RSS Methodology Applications Program (RSSMAP)	Oconee 3	8.2E-5
	Grand Gulf 1	3.7E-5

Program	Plant	Core-Melt Frequency (1/py)
	Calvert Cliffs 2	~4E-4[a]
	Sequoyah 1	~6E-5
Interim Reliability Evaluation Program (IREP)	Crystal River 3	3.7E-4
	Arkansas Nuclear One 1	5.0E-5
	Browns Ferry 1	2.0E-4
Utility	Indian Point 2	4.7E-4
	Indian Point 3	1.9E-4
	Zion 1	6.7E-5
	Zion 2	6.7E-5
	Limerick 1	1.5E-5
	Limerick 2	1.5E-5

(a) Takes credit for AFWS upgrade (Hatch 1982).

The average core-melt frequency is ~1.4E-4/py. Assuming that detailed systems interaction studies have generally not been included in risk/reliability analyses performed to date, the base-case core-melt frequency at a typical LWR in the first group of the SEP-phase III is taken to be 10% higher than this value, or 1.5E-4/py. SIR (hardware/procedural fixes implemented as a result of the systems interaction analyses) is assumed to reduce this frequency by 10%, or 1.5E-5/py.

To obtain the base and adjusted-case, affected public risks, the overall risk is written as follows:

$$W_o = \bar{F}_o R_o$$

where W_o = overall risk

$\bar{F}_o$ = overall core-melt frequency

R_o = average dose factor.

Denoting the number of plants as N and their average remaining lives as $\bar{T}$, the average dose factor for an LWR can be estimated as follows:

$$(R_o)_{LWR} = \frac{(N\bar{T}W_o)_{PWR} + (N\bar{T}W_o)_{BWR}}{(N\bar{T}\bar{F}_o)_{PWR} + (N\bar{T}\bar{F}_o)_{BWR}}$$

Based on Appendices A-D of PNL-4297,

$$(R_o)_{LWR} = 3.3E+6 \text{ man-rem}$$

where $N = 90$ (PWR) and 44 (BWR)

$\bar{T} = 28.8$ yr (PWR) and 27.4 yr (BWR)

$W_o = 207$ man-rem/py (PWR) and 250 man-rem/py (BWR)

$\bar{F}_o = 8.2E-5$/py (PWR) and 3.7E-5/py (BWR)

Thus, for this issue,

$$W(\text{base case}) = (1.5E-4/py)(3.3E+6 \text{ man-rem}) = 495 \text{ man-rem/py}$$

$$W^*(\text{adjusted case}) = (1.35E-4/py)(3.3E+6 \text{ man-rem}) = 446 \text{ man-rem/py.}$$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Systems Interaction (II.C.3/A-17)

2. Affected Plants (N):

   Four backfit LWRs (see Step 2 of Table 1). No occupational dose from accidents will be avoided unless a plant implements hardware/procedural fixes. Likewise, occupational dose will be accumulated from SIR implementation and operation/maintenance only if hardware fixes are imposed.

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

   $\bar{T}$ = 23.9 yr (See Step 2 of Table 1).

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

   $\Delta(\bar{F}D_R)$ = (1.5E-5/py)(19,900 man-rem) = 0.30 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
29	1700	0

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

   Labor inside containment = 2.7 man-yr/plant[a]

   Labor outside containment = 2.7 man-yr/plant[a]

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

   Exposure rates of 25 mR/hr and 2.5 mR/hr are assumed for work in radiation zones inside and outside containment, respectively.

$$D = [(2.7 \text{ man-yr/plant})(.025 \text{ R/hr}) + (2.7 \text{ man-yr/plant})(.0025 \text{ R/hr})](40 \text{ man-hr/man-wk})(44 \text{ man-wk/man-yr})$$

$$= 131 \text{ man-rem/plant}^{(a)}$$

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

   ND = 4(131) = 524 man-rem

(a) See Attachment 2.

TABLE 2. (contd)

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

Step 9 of Table 3 indicates that 1 man-wk/py will presumably be expended in operation/maintenance of a hardware/procedural fix. Where work in radiation zones is necessary, worker productivity is roughly 75%. Thus, SIR operation/maintenance labor in radiation zones is assumed to amount to (0.75)(1 man-wk/py) = 0.75 man-wk/py. Assuming this to be divided equally inside and outside containment (as for implementation labor), the following estimates result:

Labor inside containment = (0.5)(0.75 man-wk/py)
= 0.375 man-wk/py

Labor outside containment = (0.5)(0.75 man-wk/py)
= 0.375 man-wk/py

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

Again assuming exposure rates of 25 mR/hr and 2.5 mR/hr inside and outside containment, respectively,

$$D_O = [(0.375 \text{ man-wk/py})(.025 \text{ R/hr}) + (0.375 \text{ man-wk/py})(.0025 \text{ R/hr})] (40 \text{ man-hr/man-wk}) = 0.41 \text{ man-rem/py}$$

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

$$N\bar{T}D_O = 4(23.9)(0.41) = 39 \text{ man-rem/py}$$

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
560	1700	190

ATTACHMENT 2

Implementation of the resolution for the Systems Interaction issue will presumably involve utility labor in radiation zones consisting of a plant walk-through and any hardware fixes necessitated under existing NRC regulations to reduce the potential for systems interactions. This walk-through will follow a detailed review of plant drawings for potential systems interaction locations. Thus, it is envisioned that the walk-through will be sufficiently directed so as not to include the entire plant.

Such a walk-through was conducted at Diablo Canyon for seismically-induced systems interactions. Although the pilot studies will cover the full range of systems interactions, the proposed approach is designed to minimize the time spent examining each type. Thus, it is assumed that the utility labor to conduct the walk-through will be approximately equivalent to that expended per plant in conducting the walk-through at Diablo Canyon. From personal communication with a consultant on the Diablo Canyon study, a value of 3 man-yr/plant is estimated.

Additional hardware/procedural fixes necessitated under existing NRC regulations are envisioned over the simple fixes performed at Diablo Canyon due to the broader scope of the proposed pilot reviews. The industry cost associated with implementing these fixes is estimated at $4.8E+5/plant (see Attachment 3). This translates into 4.8 man-yr/plant using the standardized utility labor rate of $1.0E+5/man-yr. Assuming half of this labor is hardware-related, necessitating exposure in radiation zones, an estimate of 2.4 man-yr/plant is obtained. Combining this with the 3 man-yr/plant for the walk-through yields an estimate of 5.4 man-yr/plant of utility labor in radiation zones to implement resolution of the Systems Interaction Issue.

At Diablo Canyon, most of the labor involved in the walk-through and simple hardware fixes took place inside containment (~90%). For the pilot studies, non-safety/support systems are expected to be a focal point for systems interaction searches. Thus, a smaller percentage of labor will be expended inside containment during the walk-throughs and hardware fixes. It is assumed that 50% of the labor will be inside containment and 50% outside containment. For a total labor effort of 5.4 man-yr/plant, this translates into 2.7 man-yr/plant in each area.

Outside of containment, the radiation dose rate will average ~2.5 mR/hr. Within containment, the dose rate will be higher even during shutdwon when the walk-through and hardware fixes are assumed to take place. A value of 25 mR/hr is assumed.

3.0 SAFETY ISSUE COSTS

The industry and NRC costs due to SIR are estimated this section. Analysis results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Systems Interaction (II.C.3/A-17)

2. Affected Plants (N):

   Four backfit LWRs (see Step 2 of Table 1). No occupational dose from accidents will be avoided unless a plant implements hardware/procedural fixes. Likewise, occupational dose will be accumulated from SIR implementation and operation/maintenance only if hardware fixes are imposed.

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

   $\bar{T}$ = 23.9 yr (See Step 2 of Table 1)

Industry Costs (Steps 4 through 12)

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

   $\Delta(\bar{F}A)$ = (1.5E-5/py)($1.65E+9) = $2.5E+4/py

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

Best Estimate	Upper Bound	Lower Bound
$2.4E+6	$1.4E+8	0

6. Per-Plant Industry Resources for SIR Implementation:

   Costs are estimated directly in next step.

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

   As discussed in Section 1.0, SIR involves two parts--systems interaction analysis (in conjunction with an NREP study) and implementation of fixes--at each affected plant. The costs of each part are estimated as follows:

TABLE 3. (contd)

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

Systems interaction analysis = \$5.0E+5/plant
(in conjunction with NREP)

Hardware/procedural fixes = \$4.8E+5/plant
(see Attachment 3)

I = \$9.8E+5/plant

(Note--as discussed in Attachment 3, the cost of an independent systems interaction analysis would be much higher-- ~\$2.4E+6/plant based on the Diablo Canyon experience--than one done in conjunction with NREP.)

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

NI = 4(\$9.8E+5) = \$3.92E+6

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

Updating an existing systems interaction study is assumed to require 1 man-wk/py. Operation/maintenance of a hardware/procedural fix is also taken to require 1 man-wk/py (for procedural fixes, operation/maintenance may take the form of reviews and updates). Therefore, the labor required for SIR operation/maintenance amounts to 2 man-wk/py.

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

I_0 = 2(\$2270) = \$4540/py

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

$N\bar{T}I_0$ = 4(23.9)(\$4540) = \$4.3E+5

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$4.4E+6	\$6.3E+6	\$2.4E+6

TABLE 3. (contd)

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Guidelines for performing systems interaction analyses are essentially complete. A small amount of NRC staff labor, say 6 man-wk, is presumed necessary to integrate these guidelines with those for NREP.

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

C_D = (6 man-wk)($2270/man-wk) = $1.4E+4

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

To support SIR implementation, the following amounts of NRC labor are assumed:

1. Monitor and review systems interaction study (in conjunction with NREP) = 0.6 man-yr/plant (taken to be 20% of labor required to monitor and review NREP study, 3 man-yr/plant, as given in issue II.C.2).

2. Monitor and review hardware/procedural fixes = 0.25 man-yr/plant (taken to be 50% of labor required to monitor and review fixes from NREP studies, 0.5 man-yr/plant, as given in issue II.C.2)

Therefore, the labor required to support SIR implementation amounts to 0.85 man-yr/plant.

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

C = (0.85)($1.0E+5) = $8.5E+4/plant

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

NC = 4($8.5E+4) = $3.4E+5

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

To review SIR operation/maintenance, the following amounts of NRC labor are assumed:

1. Review systems interaction analysis updates = 0.1 man-wk/py

2. Inspect/review hardware/procedural fixes = 0.1 man-wk/py

Therefore, the labor required to review SIR operation/maintenance amounts to 0.2 man-wk/py.

TABLE 3. (contd)

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

$$C_o = (0.2)(\$2270) = \$454/py$$

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

$$N\bar{T}C_o = 4(23.9)(\$454) = \$4.3E+4$$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$4.0E+5	\$5.7E+5	\$2.3E+5

ATTACHMENT 3

An independent systems interaction analysis has been conducted at the Diablo Canyon site for seismically-induced initiators. The total cost of the program for both Diablo Canyon units has been reported as \$4E+6 (Killpack 1981). This included the following elements:

1. Identifying safety-related equipment responsible for performing the plant safety functions. System schematics, technical specifications, and operating procedures were reviewed.

2. Establishing criteria for postulating and evaluating systems interactions, such as assumed displacements for pipes under seismic loads.

3. Postulating the potential interactions between the critical safety-related equipment and any other equipment (safety or non-safety) given seismic loads. Plant drawings were reviewed to determine the locations for possible interactions. Walk-throughs were conducted to verify the drawings and search for possible interactions not indicated by the drawings.

4. Technically evaluating the interactions, including whether they could realistically occur and what, if any, detrimental effects they could impose.

5. Implementing some simple, minor fixes where practical to reduce interaction potential (e.g., restraining a free-swinging chain which could have struck a safety-related pipe).

The two Diablo Canyon units are similar, and the systems interaction analysis process used for each was comparable. The cost required to analyze the second unit should have been less than that of the first due to these similarities. Thus, the cost for the first plant reviewed would have been greater than half the total cost, i.e., >\$2E+6. Assuming that the cost for the second unit was 2/3 that for the first, the cost for the systems interaction program at the first unit becomes \$2.4E+6. This is taken to be the cost per plant of conducting an <u>independent</u> systems interaction review.

As indicated in element five above, this cost included implementation of minor fixes to reduce the potential for seismically-induced systems interactions. Presumably, the cost associated with these fixes was a small part of the total, say 10% (\$2.4E+5/plant). Under the current scope, the pilot systems interaction studies will cover the full range of interactions, not only those due to seismic initiation. However, the advocated systematic approach is designed to minimize the time spent examining each interaction type. Therefore, it is assumed that the industry cost per pilot plant to implement hardware/procedural fixes will be only twice that expended at Diablo Canyon, or \$4.8E+5/plant.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: II.C.4, Reliability Engineering

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

The role of probabilistic risk analyses and reliability engineering in reactor design, operation, and maintenance needs to be integrated into the licensing process. This task would result in the promulgation of a new regulatory guide, defining acceptable reliability engineering programs for all reactors. Hardware/procedural fixes could result.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 3.9E+5

OCCUPATIONAL DOSES:

SIR Implementation =	1.0E+3
SIR Operation/Maintenance =	3.2E+3
Total of Above =	4.2E+3
Accident-Avoidance =	2.4E+3

COST RESULTS ($$10^6$)

INDUSTRY COSTS:

SIR Implementation =	370
SIR Operation/Maintenance =	360
Total of Above =	730
Accident-Avoidance =	200

NRC COSTS:

SIR Development =	.024
SIR Implementation Support =	8.3
SIR Operation/Maintenance Review =	16
Total of Above =	25

RELIABILITY ENGINEERING SAFETY
ISSUE II.C.4

1.0 INTRODUCTION

As defined in NUREG-0060 TMI Action Plan (US NRC 1980), Task II.C proposes to integrate a systems-oriented approach to reactor safety review using the techniques of risk assessment and reliability engineering. The elements of this task are shown below:

TASK II.C: Reliability Engineering and Risk Assessment

II.C.1 Interim Reliability Evaluation Program (IREP)

II.C.2 Continuation of IREP (NREP - National Reliability Evaluation Program)

II.C.3 Systems Interaction

II.C.4 Reliability Engineering

The IREP and NREP programs in particular will study a finite number of existing plants using risk assessment techniques. The goal is to identify and particularly high-risk accident sequences and to verify any identified safety weakness through alterations in design procedures, and technical specifications. This safety issue (II.C.4) is seen as the final goal of II.C to fully integrate reliability engineering and risk assessment into the licensing, construction, and operation of all remaining light water reactors (LWRs).

As stated in the TMI Action Plant (NUREG-0660) the original scope of II.C.4 is as follows:

> "Specifications will be developed by NRR for acceptable reliability assurance programs to be implemented by operating license holders, construction permit holders, and future construction permit applicants. The role of applicant-supplied probabilistic safety or reliability analysis in future safety analysis reports will be defined in this program. Ultimately, reliability assurance program requirements will be promulgated by SD in a new regulatory guide."

PROPOSED RESOLUTION

The NREP studies may eventually extend to all operating power plants. If this is the case, the dominant accident sequences would be identified and corrected under the NREP Program (Issue II.C.2). The primary benefit of II.C.4 would then presumably come from more subtle, general improvement in plant design and hence plant safety. A probable measure of plant risk reduction to a general reliability program could then be a percentage reduction in total plant risk (with high-risk sequences already removed under NREP). This would reflect the general over-all improvement in plant design, construction, operations and maintenance.

The current NRC plans for NREP are to initially study only 11 plants in addition to the 16 studied under IREP and other risk/reliability assessment programs (e.g., Reactor Safety Study Methodology Applications Program). This leaves 107 reactors with no scheduled risk assessments. However, the II.C program will eventually apply to all plants. It is assumed II.C.4 will account for the rest.

Issue II.C.4 is seen as the culmination of a reliability/risk analysis program which will bring all reactors up to a common standard. As such, it is uncertain that full IREP/NREP-type studies will be required to identify 'high-risk' sequences as experience in this field is gained, but risk studies for each plant of some sort will still be required to implement the program.

The SIR for Issue II.C.4 is assumed to consist of four parts:

1. Performance of risk/reliability studies at all plants currently without or not scheduled for such studies.

2. Issuance of a regulatory guide outlining standards for reliability engineering based on the results of all risk/reliability studies.

3. Implementation of a reliability engineering program at all plants based on this regulatory guide (this will most likely involve minor hardware/procedural fixes, with some small amount of risk reduction achieved).

4. Implementation of major hardware/procedural fixes at some plants deemed to have core-melt frequencies which are "too high" (see Attachment 1 to Public Risk Reduction Work Sheet).

AFFECTED PLANTS

The regulatory guide will apply to all reactors. The additional reactors requiring risk/reliability studies are assumed to number 134 - (16 + 11) = 107. As with the resolution of NREP discussed in II.C.2, the implementation of major hardware/procedural fixes is assumed to be enacted only at a fraction of the 107 reactor where the core-melt frequency is determined to be "too high." As developed in Attachment 1, one-third of the 107 reactors are assumed to exhibit core-melt frequencies deemed "too high."

Costs will thus be incurred at all reactors for implemenation of a relibility engineering program, with some small amount of risk reduction achieved (see Attachment 1). Additional costs will be incurred at 107 reactors for risk studies, with public risk reduction realized at 36 reactors where major fixes are implemented as a result of their studies.

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. Note that the analysis is the conducted for a representative LWR, rather than a PWR or BWR, due to the data base employed in analyzing this issue.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Reliability Engineering (II.C.4)

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

   All 134 LWRs are assumed to implement reliability engineering programs (with minor hardware/procedural fixes) as a result of the regulatory guide. Of 107 LWRs performing risk studies beyond NRC, 36 are assumed to implement major hardware/procedural fixes as a result of these studies. All hardware/procedural fixes are assumed to be implemented in 1988, three years beyond that assumed in Issue I.C.2 for NREP. Thus, the number of affected plants and their average remaining lives are as follows.

	N	$\bar{T}$ (yr)
PWR	90	27.4
BWR	44	26.0
LWR	134	26.9

3. Plants Selected for Analysis:

   A hypothetical "industry-average" LWR is chosen as the representative plant. It is assumed to have a core-melt frequency of 1.3E-4/py just prior to implementation of the major and minor hardware/procedural fixes as part of SIR.[a]

(a) See Attachment 1.

TABLE 1. (contd)

4-7. Steps Related to Affected Parameters, Accident Sequences and Release Categories and Their Base-Case Values:

The base-case, affected core-melt frequency is estimated directly in the next step. Thus, these steps are omitted.

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

$\bar{F}$ = 1.3E-4/py[(a)]

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

W = 429 man-rem/py[(a)]

10-12. Steps Related to Affected Parameters, Accident Sequence and Release Categories and Their Adjusted-Case Values:

The adjusted-case, affected core-melt frequency is estimated directly in the next step. Thus, these steps are omitted.

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

F* = 9.7E-5/py[(a)]

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

W* = 320 man-rem/py[(a)]

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

$\Delta\bar{F}$ = 3.3E-5/py

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

ΔW = 109 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
3.9E+5	4.6E+7	0

(a) See Attachment 1.

ATTACHMENT 1

The method for determining the impact of extending risk/reliability studies for all reactors is essentially that used for evaluating Issue II.C.2. For the 10% plants currently without or not scheduled for risk/reliability studies, it is assumed that 36 have core-melt frequencies judged to be "too high." As stated in Attachment 1 of II.C.2.

Of the 15 risk/reliability studies currently available for LWRs, five plants have been assessed as having core-melt frequencies in excess of 1E-4/py: Indian Point Units 2 and 3 (IP2 and IP3), Crystal River 3 (CR3), Calvert Cliffs 2(CC2), and Browns Ferry 1 (BF1). The table below lists the core-melt frequencies, frequencies of the most dominant accident sequences, and percent contributions of the core-melt frequencies from these most dominant sequences for these five LWRs:

Plant	Core-Melt Frequency (1/py)	Dominant Seq. Freq. (1/py)	Dominant % Contribution
IP2	4.7E-4	1.4E-4	30
IP3	1.9E-4	8.2E-5	43
CR3	3.7E-4	1.7E-4	46
CC2[a]	4.0E-4	9.6E-5	25
BF1	2.0E-4	9.7E-5	49

(a) The values given here assume AFWS upgrades scheduled for 1982 are implemented.

For these five plants, the average core-melt frequency is 3.3E-4/py and the average contribution of the most dominant sequence is 39%.

For the 36 plants impacted it is assumed that the fix implemented will reduce the frequency of the most dominant sequence at each by 90%, giving an adjusted-case core-melt frequency of:

$$(3.3E-4)[1 - (0.90)(0.39)] = 2.1E-4/py$$

To determine the effectiveness of the implementation minor hardware/procedural fixes resulting from a reliability engineering program, it is assumed that a 5% reduction in the average overall core-melt frequency can be achieved. An industry average at the point in the time where this issue will be implemented is required. The following is taken from issue II.C.3, Systems Interaction:

Risk/reliability studies have been completed on 15 plants to date. The core-melt frequencies at these plants are as follows:

Program	Plant	Core-Melt Frequency (1/py)
Reactor Safety Study (RSS)	Surry 1	6.0E-5
	Peach Bottom 2	2.9E-5
RSS Methodology Applications Program (RSSMAP)	Oconee 3	8.2E-5
	Grand Gulf 1	3.7E-5
	Calvert Cliffs 2	~4E-4[a]
	Sequoyah 1	~6E-5
Interim Reliability Evaluation Program (IREP)	Crystal River 3	3.7E-4
	Arkansas Nuclear One 1	5.0E-5
	Browns Ferry 1	2.0E-4
Utility	Indian Point 2	4.7E-4
	Indian Point 3	1.9E-4
	Zion 1	6.7E-5
	Zion 2	6.7E-5
	Limerick 1	1.5E-5
	Limerick 2	1.5E-5

(a) Takes credit for AFWS upgrade.

The average core-melt frequency for the 15 plants is ~1.4E-4/py.

As developed above, the 5 plants with core-melt frequencies deem "too high" have an average of 3.3E-4/py. The remaining 10 have an average of 4.8E-5/py, derived from the above list. The base-case core-melt frequency for this issue is calculated for a point in time when 27 plants have had risk/reliability studies completed (1988, three years beyond that assumed in II.C.2), with one-third (9) implementing fixes to reduce to reduce the most dominant sequence frequency at each by 90% to yield a core-melt frequency of 2.1E-4/py. The remaining 18 are assumed to have a core-melt frequency of 4.8E-5/py.

Of the remaining 107 plants, again one-third (36) are assumed to have "too high" core-melt frequencies of 3.3E-4/py, requiring fixes on dominant sequences the remaining 71 plants are but at 4.8E-5/py. The base-case core-melt frequency is than put at

$$\bar{F} = \frac{(18 + 71)(4.8E\text{-}5/py) + 9(2.1E\text{-}4/py) + 36(3.3E\text{-}4/py)}{134}$$

$$= 1.3E\text{-}4/py$$

To calculate the adjusted-case core-melt frequency, it is assumed that the plants with core-melt frequencies judged "too high" will again be reduced to 2.1E-4/py. In addition, it is assumed that the implementation of minor hardware/procedural fixes resulting from an industry-wide reliability engineering program will lower the overall core-melt frequency by 5% after the major fixes from dominant sequences have been implemented.[a] The adjusted-case core-melt frequency is the

$$\bar{F} = (0.95)\ \frac{(18 + 71)(4.8E\text{-}5/py) + (9 + 36)(2.1E\text{-}4/py)}{134}$$

$$= 9.7E\text{-}5/py$$

The dose factor $(R_o)_{LWR}$ necessary to calculate overall risk is estimated from the values given in Appendices A-D of PNL-4297 (Andrews 1983) as

$$(R_o)_{LWR} = \frac{(N\bar{T}W_o)_{PWR} + (N\bar{T}W_o)_{BWR}}{(N\bar{T}\bar{F}_o)_{PWR} + (N\bar{T}\bar{F}_o)_{BWR}}$$

where

N = 90 (PWR) and 44 (BWR)

$\bar{T}$ = 28.8 yr (PWR) and 27.4 yr (BWR)

W_o = 207 man-rem/py (PWR) and 250 man-rem/py (BWR)

$\bar{F}_o$ = 8.2E-5/py (PWR) and 3.7E-5/py (BWR)

giving $(R_o)_{LWR}$ = 3.3E+6 man-rem.

Thus, for this issue the base- and adjusted-case public risks are:

$$W = (1.3E\text{-}4/py)(3.3E{+}6 \text{ man-rem}) = 429 \text{ man-rem/py}$$

$$W^* = (9.7E\text{-}5/py)(3.3E{+}6 \text{ man-rem}) = 320 \text{ man-rem/py.}$$

(a) Note that if NREP is eventually extended to all plants, II.C.4 could no longer take credit for fixes to dominant sequences, and this issue would require re-evaluation. Only the 5% reduction would remain.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Reliability Engineering (II.C.4)

2. Affected Plants (N):

   All 134 LWRs are assumed to implement a reliability engineering program (and minor hardware/procedural fixes) as a result of the regulatory guide. 36 plants are assumed to require major hardware/procedural fixes to reduce the frequencies to most dominant sequences. Since implementation of these fixes will not occur until 1988, the affected plants are divided into three groups:

   1) 71 backfit plants
   2) 52 forward-fit plants operational by 1988
   3) 11 forward-fit plants still not operational by 1988.

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

	$\bar{T}$ (yr)
71 backfit LWRs	26.9
52 forward-fit LWRs operational by 1988	26.4
11 remaining forward-fit LWRs	30
All 134 LWRs	26.9

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

$$\Delta(\bar{FD}_R) = (19{,}900 \text{ man-rem})(3.3E\text{-}5/\text{py}) = 6.6E\text{-}1 \text{ man-rem/py}$$

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.4E+3	5.6E+4	0

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

   As per Issue II.C.2, it is assumed that major hardware/procedural fixes impacting dominant sequence cost \$2.0E+6/plant with 20% of the cost representing staff labor in a radiation zone. At \$2.0E+5/man-yr, this represents 4 man-yr/plant. However, some plants may require fixes more procedural oriented, presumably involving little radiation work. It is assumed that 3/4 or 27 of 36 plants are so affected.

For implementation of the reliability engineering program (via minor hardware/procedural fixes), NUREG-0660 estimates 10 man-years/plant. Because this involves implementation in the design and construction phase in addition to operation, work in radiation zones would represent a smaller fraction than above. A 10% value is assumed here, giving 1 man-yr/plant for all 123 plants operational by 1988.

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

An average exposure rate of 2.5 mR/hr is assumed.

D = (.0025 R/hr)(40 man-hr/man-wk)(44 man-wk/man-yr)
(4 man-yr/plant) = 17.6 man-rem/plant (27 plants implementing major fixes)
(1 man-yr/plant) = 4.4 man-rem/plant (123 plants implementing minor fixes)

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

ND = (17.6 man-rem/plant)(27 plants) + (4.4 man-rem/plant)(123 plants)
= 1020 man-rem

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

For operation and maintenance, NUREG-0660 puts utility time at 1 man-yr/py. Again, 20% will be assumed to be in radiation zone, or 0.2 man-yr/py. Note that this applies to all 134 plants.

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

A radiation exposure rate of 2.5 mR/hr at 1760 hr/man-yr is again assumed.

TABLE 2. (contd)

$$D_O = (.0025\ R/hr)(40\ man\text{-}hr/man\text{-}wk)(0.2\ man\text{-}yr/py)$$
$$= 0.88\ man\text{-}rem/py$$

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

$$N\bar{T}O_O = (134\ plants)(26.9\ yrs)(0.88\ man\text{-}rem/py)$$
$$= 3170\ man\text{-}rem$$

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
4200	1.3E+4	1400

3.0 SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. Results are summarized in Table 3. Note that the costs of performing risk/reliability studies beyond currently existing or scheduled risk/reliability assessments are incurred by all additional 107 LWRs. However, costs for implementing major hardware/procedural fixes associated with dominant accident sequences are required at only 36 plants. Costs associated with implementation of minor hardware/procedural fixes are incurred at all LWRs.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Reliability Engineering (II.C.4)

2. Affected Plants (N):

   All 134 LWRs are affected as follows:

   - All 134 LWRs will implement minor hardware/procedural fixes as part of their reliability engineering programs
   - 107 LWRs will perform risk/reliability studies as part of the SIR for II.C.4
   - 36 LWRs will implement major hardware/procedural fixes to reduce the frequency of the most dominant accident sequences (not--to simplify calculations, these 36 LWRs are assumed to all be operational by 1988).

   For implementation of these fixes in 1988, the affected plants are again divided into three groups (see Step 2 of Table 2):

   1) 71 backfit LWRs
   2) 52 forward-fit LWRs (by 1988)
   3) 11 forward-fit LWRs (after 1988)

TABLE 3. (contd)

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

	$\bar{T}$ (yr)
71 backfit LWRs	26.9
52 forward-fit LWRs (by 1988)	26.4
11 forward-fit LWRs (after 1988)	30
All 134 LWRs	26.9

Industry Costs (Steps 4 through 12):

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

$$\Delta(\bar{FA}) = (\$1.65E+9)(2.2E-5/py)$$
$$= \$5.4E+4/py$$

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

Best Estimate	Upper Bound	Lower Bound
\$2.0E+8	\$4.6E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

Plant user resources are estimated in NUREG-0660 at 10 man-years/plant for implementation of the reliability engineering program. Costs for implementing fixes to dominant accident sequences are estimated directly in the next step.

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

As per Issue II.C.2, costs for risk/reliability studies are put at \$1.5E+6 based on previous IREP studies. This will be required in some form for all 107 palnts currently without or not scheduled for risk/reliability studies, but it is uncertain if the same level of detail will be required after the experience gained in the NREP program. The \$1.5E+6 estimate is used here. For those plants found to require a major fix to reduce the frequency of some dominant accident sequence (36 plants of 107), it is assumed that the average cost of the fix is \$2.0E+6. This

TABLE 3. (contd)

figure is common to Issues II.B.6 and II.C.2. In addition, these latter 36 plants would resumably require a class III license amendment fee of \$4000.

Affected Plants	I
134 LWRs (implementation of reliability engineering program via minor fixes)	(10 man-yr/plant)(\$1.0E+5/man-yr) = \$1.0E+6/plant
107 LWRs (risk/reliability studies)	\$1.5E+6/plant
36 LWRs (implementation of major fixes)	\$2.0E+6/plant + \$4000/plant = \$2.0E+6/plant

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

$$NI = (134 \text{ plants})(\$1E+6/\text{plant}) + (107 \text{ plants})(\$1.5E+6/\text{plant}) + (36 \text{ plants})(\$2.0E+6/\text{plant})$$

$$= \$3.7E+8$$

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

NUREG-0660 estimates sustaining labor at 1 man-yr/py. This applied to all 14 plants.

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

$$I_O = (\$1E+5/\text{man-yr})(1 \text{ man-yr/py}) = \$1E+5/\text{py}$$

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

$$(N\bar{T}I_O) = (134)(26.9 \text{ yrs})(\$1E+5/\text{py}) = \$3.6E+8$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$7.3E+8	\$9.9E+8	\$4.7E+8

TABLE 3. (contd)

NRC Costs (Steps 13 through 21):

13. NRC Resources for SIR Development:

The risk/reliability studies required for this issue resolution are assumed to be developed on the experience of the IREP/NREP programs. Guidelines for the studies should be fully developed by the previous program. NUREG-0660 estimates 2.4 man-yr required for development of the program culminating in the issuance of regulatory guide.

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

$$C_D = (2.4 \text{ man-yr})(\$1.0E+5/\text{man-yr})$$
$$= \$2.4E+5$$

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

Issue II.C.2 assumes 3 man-yr/plant is required to monitor NREP studies, plus 0.5 man-yr/plant to review those plant performing major fixes for dominant accident sequences. It is assumed that, based on the experience of NREP, the NRC will be able to reduce its monitoring of risk studies to 0.5 man-yr/plant for the 107 plants beyond NREP. It is assumed that 0.5 man-yr/plant will still be required in conjunction with the 36 plants assumed to perform major fixes from dominant sequences frequency reduction.

Review of the general reliability provisions of the engineering program is assumed to add one man-month/plant of NRC labor for all 134 LWRs.

Affected Plants	Labor (man-yrs/plant)
134 LWRs (reliability engineering programs)	0.083
107 LWRs (risk/reliability studies)	0.5
36 LWRs (dominant sequence fixes)	0.5

TABLE 3. (contd)

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

Affected Plants	($/plant)
134 LWRs (reliability engineering programs)	8.3E+3
107 LWRs (risk/reliability studies)	5.0E+4
36 LWRs (dominant sequence fires)	5.0E+4

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

NC = (134 plants)($8.3E+3/plant) + (107 plants)($5.0E+4/plant)
+ (36 plants)($5.0E+4/plant)

= $8.3E+6

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

Two man-weeks/py are estimated, primarily assoicated with review of the reliability engineering program.

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

Co = (2 man-weeks/py)(1 man-yr/44 man-weeks)($1E+5/man-yr)
= $4.5E+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)(26.9 yr)($4.5 E+3/py)
= $1.6E+7

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$2.5E+7	$3.4E+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: III.D.3.1(1), Radiation Protection Plans

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

Improvements can be made in radiation protection programs by better defining criteria and responsibility for such programs. The primary result would be be better monitoring and control of occupational dose received individually and collectively. A radiation protection plan would be a concise statement of plant radiation protection policy and program, addressing the elements of a strong, self-improving program as described in NUREG-0761. The plan would be a guiding document for implementing procedures which currently exist, at least in part, at licensed plants.

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 =	-3.0E+5
Total of Above =	-3.0E+5
Accident-Avoidance =	510

COST RESULTS ($$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	17
SIR Operation/Maintenance =	760
Total of Above =	780
Accident-Avoidance =	21
NRC COSTS:	
SIR Development =	0
SIR Implementation Support =	0.061
SIR Operation/Maintenance Review =	5.2
Total of Above =	5.2

RADIATION PROTECTION PLANS

TMI ACTION PLAN ITEM III.D.3.1(1)

1.0 SAFETY ISSUE DESCRIPTION

Improvements in nuclear power plant worker radiation protection programs can be made by better defining the criteria and responsibility for such programs. In-depth appraisals of health physics programs at all operating nuclear power plants were performed in 1980 and 1981. These appraisals, summarized in NUREG-0855 (US NRC 1982), indicated that certain generic deficiencies existed at many plants due in part to lack of specific performance criteria and/or assigned responsibility for programs. The establishment of a radiation protection plan as a guiding document for implementing procedures has been proposed as a method for formalizing commitment to specific performance criteria contained in Regulatory Guides and NRC Standard Review Plan Chapter 12 (NUREG-0800). Proposed guidance and acceptance criteria for radiation plans have been published in draft form as NUREG-0761 (US NRC 1981).

As currently envisioned, radiation protection plans would tie together specific implementing procedures, many of which currently exist at licensed plants. Additional procedures may be required at many plants to fully implement the plan; however, extensive revision of procedures should not generally be required. Administrative and technical manpower would be required to develop the plan, revise and write procedures as necessary, and possibly install some additional equipment (such as additional survey equipment). Installation of such equipment should not require any significant work in radiation areas. The benefit of radiation protection plans would be in two primary areas: reduction of individual and collective dose due to improved planning and controls for work in radiation areas, and improved confidence in results of radiation protection programs.

The assessment of this safety issue resolution has been performed by consensus opinions of four PNL health physicists who were extensively involved in the Health Physics Appraisal Program. These personnel possess expertise from both industry and regulatory sides of the issue. Estimates of routine cost and probable man-rem reductions have been discussed and agreed upon. For core-melt accident recovery and refurbishing, the panel has assumed man-rem savings comparable on a percentage basis to those for routine operations. The cost impact of these man-rem savings was then estimated by a PNL Energy Systems expert involved in estimating accident recovery costs.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with radiation protection plans are described in the following two subsections.

2.1 PUBLIC RISK REDUCTION

The development of radiation protection plans has no impact on public safety. The pathways to public risk reduction are either a mitigation of consequences to the public, given an accident, or the reduction in accident frequency. Radiation protection plans affect neither.

2.2 OCCUPATIDNAL DOSE

With regard to the occupational dose reduction due to accident-avoidance, no change in accident frequency is expected to occur due to this SIR. However, a small change in occupational accident-recovery dose is expected. Radiation protection plans are primarily oriented toward routine plant operation. However, in the event of a major core-melt accident, specialized procedures would be developed. Having the upgraded radiation protection plan for normal operation in place is expected to result in improved specialized accident-recovery procedures. The resulting reduction in occupational dose for plant recovery from a core-melt accident is estimated to be approximately 2000 man-rems, (10% of the 20,000 man-rem total given in Appendix D of NUREG/CR 2800 [Andrews et al. 1983]).

The implementation of radiation protection plans would be primarily an administrative effort. Therefore, there is no occupational exposure associated with implementation. The establishment of radiation protection plans is estimated to result in a reduction of occupational dose during operation. This reduction would be due to improved controls on personnel dose and an improved ALARA (As Low As Reasonably Achievable) Program. A reasonable average estimate of the occupational dose reduction from establishing radiation protection plans is 10%. Based on a typical plant collective occupational dose of 800 man-rem/py, a 10% reduction would be 80 man-rem/py. Results of the analysis for occupational dose are summarized in Table 1.

TABLE 1. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Radiation Protection Plans [III.D.3.1(1)]

2. Affected Plants (N):

   44 BWRs (24 backfit, 20 forward-fit)

   90 PWRs (47 backfit, 43 forward-fit)

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

	$\bar{T}$ (yr)
BWR:	27.4
PWR:	28.8

TABLE 1. (contd)

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

For this analysis, the per-plant occupational dose reduction due to accident-avoidance is defined as follows:

$$\Delta(\bar{F}D_R) = \bar{F}(\Delta D_R)$$

where $\bar{F}$ = total core-melt frequency (8.2E-5/py and 3.7E-5/py for Oconee and Grand Gulf, respectively, as the representative PWR and BWR).

ΔD_R = (0.10)(20,000 man-rem) = 2000 man-rem

Therefore,

$$\Delta(\bar{F}D_R)_{PWR} = (8.2E\text{-}5/py)(2000\ \text{man-rem}) = 0.164\ \text{man-rem/py}$$

$$\Delta(\bar{F}D_R)_{BWR} = (3.7E\text{-}5/py)(2000\ \text{man-rem}) = 0.074\ \text{man-rem/py}$$

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper[a]	Lower
510	3.1E+4	0

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

Since implementation of radiation protection plans is seen as mainly administrative, no significant work in radiation zones should result.

$$D = 0$$

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

Dose estimated directly in next step.

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

D_o = -80 man-rem/py (Negative sign indicates dose reduction.)

This assumes a 10% reduction in an average plant collective dose of 800 man-rem/py.

(a) Calculated using D_R = 2.0E+4 man-rem.

TABLE 1. (contd)

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

$$N\bar{T}D_O = [90(28.8) + 44(27.4)](-80) = -3.0E+5 \text{ man-rem}$$

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
-3.0E+5	-1.0E+5	-9.0E+5

(Negative sign indicates dose reduction.)

3.0 SAFETY ISSUE COSTS

With regard to the industry cost savings due to accident avoidance, there would be no change in the accident frequency for radiation protection plans. However, the cost associated with the accident is expected to be reduced due to the reduction in occupational exposure and, therefore, manpower required. It is estimated that a 10% reduction in occupational dose during initial and recovery phases of an accident would result from implementing radiation protection plans. This would result in approximately a 5% savings in overall accident-consequence costs (Murphy 1982).

It is estimated that 35 man-weeks of labor and $50,000 of equipment would be required per plant to implement the radiation protection plans. In order to operate under the new radiation protection plans, it is felt most plants would have to add personnel. It is estimated that one professional and one technical staff member would be needed over the remaining lives of the plants.

At present the NRC does not plan an extensive review of radiation protection plan submittals. A nominal amount of labor will be required for receipt, acknowledgement, and cursory review at the time submittals are made. This would represent the NRC effort for implemenation. This labor is estimated to be 0.2 man-wk/plant (1 day per plant).

The evaluation of plans will be performed as part of the routine NRC/IE inspection program. The increase in NRC labor for periodic review of radiation protection plans is estimated at approximately 3 man-days/py. This estimate assumes one day added time for inspection preparation (including radiation protection plan review) and two additional days per year for in-plant inspection.

Results of the analysis for industry and NRC costs are summarized in Table 2.

TABLE 2. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Radiation Protection Plans [III.D.3.1(1)]

2. Affected Plants (N):

   44 BWRs (24 backfit, 20 forward-fit)

   90 PWRs (47 backfit, 43 forward-fit)

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

	$\bar{T}$ (yr)
BWR:	27.4
PWR:	28.8

Industry Costs (Steps 4 through 12)

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

   For this analysis, the per-plant industry cost savings due to accident avoidance is defined as follows:

$$\Delta(\bar{F}A) = \bar{F}(\Delta A)$$

where $\bar{F}$ = 8.2E-5/py (representative PWR--Oconee) and 3.7E-5/py (representative BWR--Grand Gulf)

ΔA = (0.05)($1.65E+9) = $8.3E+7

Therefore,

$\Delta(\bar{F}A)_{PWR}$ = (8.2E-5/py)($8.3E+7) = $6800/py

$\Delta(\bar{F}A)_{BWR}$ = (3.7E-5/py)($8.3E+7) = $3100/py

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

Best Estimate	Upper Bound[a]	Lower Bound
$2.1E+7	$2.5E+9	0

(a) Calculated using A = $1.65E+9.

TABLE 2. (contd)

6. Per-Plant Industry Resources for SIR Implementation:

Labor = 35 man-wk/plant

Equipment = survey instruments, radiation protection equipment, dosimetry equipment, and calibration equipment (cost estimated directly in next step).

Additional down-time = none

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

Labor = 35($2270) = $7.9E+4/plant

Equipment = $5.0E+4/plant

I = $1.29E+5/plant

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

NI = 134($1.29E+5) = $1.7E+7

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

Two additional staff members will be needed over the remaining lives of the plants, corresponding to additional labor of 2 man-yr/py.

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

I_o = 2($1.0E+5) = $2.0E+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)] ($2.0E+5) = $7.6E+8

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$7.8E+8	$1.2E+9	$4.0E+8

NRC Costs (Steps 13 through 21)

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

No additional NRC resources are foreseen for SIR development.

$C_D = 0$

TABLE 2. (contd)

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

    0.2 man-wk/plant

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

    C = 0.2($2270) = $454/plant

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

    NC = 134($454) = $6.1E+4

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

    3 man-day/py = 0.6 man-wk/py

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

    C_o = (0.6)($2270) = $1400/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)]($1400) = $5.2E+6

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$5.2E+6	$7.8E+6	$2.7E+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.

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U.S. NRC. March 1981. Radiation Protection Plans for Nuclear Power Reactor Licensees, NUREG-0761, Nuclear Regulatory Commission, Washington, D.C.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: IV.E.5, Safety Decision Making - Assess Currently Operating Reactors

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

This issue proposes extension of the current SEP efforts to 11 additional plants. The assumed resolution involves two parts: 1) engineering studies to identify and evaluate modifications to the plants and 2) implementation of modifications in plants not affected by the NREP issue (II.C.2).

AFFECTED PLANTS	BWR:(a)	Operating = NA	Planned = 0
	PWR:(a)	Operating = NA	Planned = 0

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	1.3E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	70
SIR Operation/Maintenance =	42
Total of Above =	112
Accident-Avoidance =	80

COST RESULTS ($$\$10^6$$)

INDUSTRY COSTS:	
SIR Implementation =	92
SIR Operation/Maintenance =	20
Total of Above =	112
Accident-Avoidance =	6.6
NRC COSTS:	
SIR Development =	13
SIR Implementation Support =	3.5
SIR Operation/Maintenance Review =	0.075
Total of Above =	17

(a) Eleven backfit LWRs are assumed to be affected. There is no breakdown into BWRs and PWRs in this analysis.

SAFETY DECISION MAKING-
ASSESS CURRENTLY OPERATING REACTORS

ISSUE IV.E.5

As part of developing plans for an integrated program of safety decision making, NRR, in consultation with other appropriate offices, will develop a plan for approval by the Commission for the systematic assessment of the safety of all operating reactors. Development of such a plan will take into account the Systematic Evaluation Program (SEP), the ACRS comments on the program, the Interim Reliability Evaluation Program (IREP) plan, and ongoing TMI lessons learned activities. This value/impact assessment of Item IV.E.5 deals with the work under the SEP. Value/impact assessment of IREP and National Reliability Evaluation Program (NREP) is presented in Items II.C.1 and II.C.2, respectively.

1.0 SAFETY ISSUE DESCRIPTION

SEP is now reviewing the ten oldest plants against current licensing review safety criteria, including Standard Review Plans, to provide the basis for integrated and balanced backfit decisions. This review is nearly complete and therefore is not part of this assessment. The next SEP phase involves evaluation of eleven additional plants. In this next phase, probabilistic risk assessment (PRA) evaluations will be coordinated with the deterministic review method (review against current licensing safety criteria). The PRA will be done as part of NREP which is TMI Action Plan Item II.C.2.

As safety-related problems are identified for each plant, resolutions are developed using procedural and administrative changes, possible credit for non-safety systems where justified, and hardware backfits as necessary to reduce risk levels. The process used to decide appropriate corrective actions employs the judgment of a team of NRC staff familiar with each plant.

SAFETY ISSUE RESOLUTION

The assumed SIR of IV.E.5 is that plant-specific studies will be performed to identify safety problems due to inconsistencies with current review practices and major risk contributors. Plant modifications will be performed to reduce risk levels. Risk and cost estimates included in this analysis are limited to the 11 plants in the next phase of the SEP. This issue is also assumed to be performed in conjunction with NREP.

NREP (II.C.2) is a program to perform risk assessments of all plants without existing risk assessments. The intent is to identify dominant core-melt accident sequences and take possible corrective actions to reduce core-melt likelihood to "acceptable" levels. Plants with core-melt frequencies greater than 1.0E-4 per plant year were assumed, in Issue II.C.2, to be candidates for modifications. Based on the past IREP studies, four of the eleven Phase III SEP plants are expected to exceed the core-melt limit. All costs and risk reductions for these four plants are attributed to Issue II.C.2. No significant additional reductions are anticipated in these plants due to this issue.

The approach followed in this issue is to assume that plant and procedure modifications would conservatively result in a 50% reduction in core melt frequency. This is supported by an analysis of dominant accident sequences in existing risk assessments for plants with core-melt frequencies below 1.0E-4/py:

Plant	Contribution of Dominant Accident Sequence	Core-Melt Frequency (1/py)
Oconee 3	13	8.2E-5
Grand Gulf	32	3.7E-5
Sequoyah 1	36	6.0E-5
Surry	10	6.0E-5
Peach Bottom	34	2.9E-5
Arkansas Nuclear One 1	20	5.0E-5
Zion 1/2	37	5.0E-5
Limerick	40	1.5E-5
Average	28	4.8E-5

The average dominant accident contribution for these studies is 28%. An additional 22% was added to consider deterministic evaluations that will be performed under the SEP. A review of additional plants considered in issue II.C.2 (NREP) also indicated that contributions from dominant accident sequences can be as high as 49%.

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. Note that the analysis is conducted for a representative LWR, rather than for a representative PWR and BWR, a consequence of the data base employed for this issue analysis (see Attachment 1).

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

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

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

   Seven of eleven backfit LWRs assumed to perform SEP studies are further assumed to implement hardware/procedural fixes as a result of these studies. Such fixes are assumed to be implemented in 1985. Therefore, for public risk reduction estimation:

   $$N = 7 \text{ backfit LWRs with } \bar{T} = 23.9 \text{ yr}$$

3. Plants Selected for Analysis:

   A hypothetical backfit LWR with a core-melt frequency of 4.8E-5 is assumed to be representative of the seven affected backfit LWRs.[a]

4-7. Steps Related to Affected Parameters, Accident Sequences and Release Categories and Their Base-Case Values:

   The base-case, affected core-melt frequency is estimated directly in the next step. Thus, these steps are omitted.

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

   $\bar{F}$ = 4.8E-5/py[a]

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

   W = 158 man-rem/py[a]

10-12. Steps Related to Affected Parameters, Accident Sequences and Release Categories and Their Base-Case Values

   The adjusted-case, affected core-melt frequency is estimated directly in the next step. Thus, these steps are omitted.

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

   $\bar{F}$* = 2.4E-5/py[a]

(a) See Attachment 1.

TABLE 1. (contd)

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

$W^* = 79$ man-rem/py[a]

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

$\Delta\bar{F} = 2.4E-5/py$

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

$\Delta W = 79$ man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.3E+4	7.9E+5	0

(a) See Attachment 1.

ATTACHMENT 1

Issue II.C.2 provides an analysis of currently available risk assessments to predict the number SEP plants that will be affected by NREP. Four of the 11 plants are expected to exceed 1E-4/py core-melt frequencies and were assumed modified under NREP. The remaining seven plants are assumed modified under this issue. Based on previous risk assessments and judgment as to the effect of deterministic reviews, the seven plants are predicted to have an average core-melt frequency deduction of 2.4E-5/py. Under these assumptions, the adjusted-case, affected core-melt frequency becomes:

$$\bar{F}^* = (4.8E\text{-}5/py)-(2.4E\text{-}5) = 2.4E\text{-}5$$

To obtain the base and adjusted-case, affected public risks, the overall risk is written as follows:

$$W_o = \bar{F}_o R_o$$

where W_o = overall risk

$\bar{F}_o$ = overall core-melt frequency

R_o = average dose factor.

Denoting the number of plants as N and their average remaining lives as $\bar{T}$, the average dose factor for an LWR can be estimated as follows:

$$(R_o)_{LWR} = \frac{(N\bar{T}W_o)_{PWR} + (N\bar{T}W_o)_{BWR}}{(N\bar{T}\bar{F}_o)_{PWR} + (N\bar{T}\bar{F}_o)_{BWR}}$$

Based on Appendices A-D of PNL-4297 (Andrews 1983),

$(R_o)_{LWR}$ = 3.3E+6 man-rem

where N = 90(PWR) and 44(BWR)

$\bar{T}$ = 28.8 yr (PWR) and 27.4 yr (BWR)

W_o = 207 man-rem/py (PWR) and 250 man-rem/py (BWR)

$\bar{F}_o$ = 8.2E-5/py(PWR) and 3.7E-5/py(BWR)

Thus, for this issue,

W = (4.8E-5)(3.3E+6 man-rem)
= 158 man-rem/py

W* = (2.4E-5/py)(3.3E+6 man-rem)
= 79 man-rem/py

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

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

2. Affected Plants (N):

   Seven backfit LWRs (see Step 2 of Table 1).

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

   $\bar{T}$ = 23.9 yr (see Step 2 of Table 1).

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

   $\Delta(F\bar{D}R)$ = (19,900 man-rem)(2.4E-5/py) = 0.48 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
80	960	0

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

   It is assumed that 4000 man-hr/plant of labor in radiation zones will be required to implement the plant modification associated with issue resolution.

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

   An average exposure rate of 2.5 mR/hr is assumed to apply to the implementation labor.

   D = (0.0025 R/hr)(4000 man-hr/plant) = 10 man-rem/plant

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

   ND = (7 plants)(10 man-rem/plant) = 70 man-rem

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

   It is assumed that 100 man-hr/py will be required for operation and maintenance.

TABLE 2. (contd)

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

Again, a 2.5 mR/hr radiation field is assumed.

D_0 = (0.0025 R/hr)(100 man-hr/py) = 0.25 man-rem/py

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

$N\bar{T}D_0$ = (7 plants)(23.9 yr)(0.25 man-rem/py) = 42 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
112	336	37

3.0 SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. Results are summarized in Table 3. Note that the cost of performing an SEP study will be incurred at all eleven backfit LWRs. However, the cost of implementing, operating, and maintaining a hardware/procedural fix will be incurred only at seven of these plants.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

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

2. Affected Plants (N):

For SIR implementation, operation, and maintenance, all eleven backfit LWRs in Phase III of the SEP are presumed to be affected. For industry cost savings due to accident avoidance, only those seven plants imposing hardware/procedural fixes will be affected.

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

$\bar{T}$ = 23.9 yr (see Step 2 of Table 1)

TABLE 3. (contd)

Industry Costs (Steps 4 through 12):

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

$$\Delta(\bar{F}A) = (\$1.65E+9)(2.4E-5/py) = \$4.0E+4/py$$

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
\$6.6E+6	\$8.0E+7	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):

Based on SEP studies completed to date, the following costs have been incurred:

- \$2M for engineering studies to identify areas of plant modifications
- \$2M - \$50M to design and install modifications.

Plans for the future SEP plants include a reduction in the number of items considered in the review. Based on conversations with NRC staff, this is estimated to reduce implementation costs to an average of \$10M/plant. For the purposes of this issue analysis, the per-plant industry implementation cost is assumed to be the following:

I = \$2M (engineering) + \$10M (design [including capital cost of equipment] and installation) = \$12M/plant with backfits

I = \$2M (for engineering)/non backfit plant.

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

$$NI = (11 \text{ plants})(\$2.0E+6/\text{plant}) + (7 \text{ plants})(\$10.0E+6/\text{plant})$$
$$= \$9.2E+7$$

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

Costs estimated directly in next step.

TABLE 3. (contd)

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

It is assumed that maintenance of modifications would cost 1% of their initial cost each year or 120,000/py for seven plants.

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

$N\bar{T}I_O$ = (7 plants)(120,000/py)(23.9 yr) = \$2.01E+7

12. Total Industry Cost (S_I):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
\$1.12E+8	\$1.6E+8	\$6.5E+7

NRC Costs (Steps 13 through 21):

13. NRC Resources for SIR Development:

7 man-yr/plant

\$500K/plant

(see Attachment 2)

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

C_D = (11 plants)([7 man-yr/plant][\$100K/man-yr] + \$500K/plant)
= \$1.3E+7

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

3 man-yr/plant

\$200K/plant

(see Attachment 2)

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

C = (3 man-yr/plant)(\$100K/man-yr) + \$200K/plant = \$500K/plant

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

NC = (7 affected plants)(\$500K/plant) = \$3.5E+6

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

NRC will review and inspect plant modifications. The assumed labor is 0.2 man-wk/py.

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

(0.2 man-wk/py)($2270/man-wk) = $450/py

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

$7.5E+4

21. Total NRC Cost (S_N):

Best Estimate	Error Bounds Upper	Lower
$1.7E+7	$2.4E+7	$1.0E+7

ATTACHMENT 2

NRC resources for past SEP studies have totaled 10 man-yr and $700K per plant. For the purposes of this study, these resources were divided as follows between SIR development and implementation support:

SIR Development -
7 man-yr/plant
$500K other resources

SIR Implementation -
3 man-yr/plant
$200K other resources.

This division was assumed as an indication of the detailed plant-specific studies that are required prior to decisions on plant modifications. Development costs apply to all 11 plants. Implementation costs apply to the seven affected plants.

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This is the second 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. This report contains results of issue-specific analyses for 15 issues. Each issue was considered within the constraints of available information as of September 1982 and two staff-weeks of labor. The results will be referenced, as one consideration in setting priorities for reactor safety issues, in an NRC prioritization report to be published at a future date.

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Cost Analysis	Passive Mechanical Failures
Prioritization	Control Systems
Reactor Coolant Pump	Systems Interation
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