Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development

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U.S. Nuclear Regulatory Commission

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Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development

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ABSTRACT

This is the sixth in a series of reports to document the development and use of a methodology developed by the Pacific Northwest Laboratory (PNL) to calculate, for prioritization purposes, the risk, dose, and cost impacts of implementing potential resolutions to reactor safety issues (see NUREG/CR-2800, Andrews, et al., 1983). This report contains the results of issue-specific analyses for 34 generic issues. Each issue was considered within the constraints of available information at the time the issues were examined and approximately 2 staff-weeks of labor. The results are referenced as one consideration in NUREG-0933, *A Prioritization of Generic Safety Issues* (Emrit, et al., 1983).

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 the resolutions to reactor safety issues and apply that methodology to issues of interest to the NRC. Results of this project will be used by the NRC to support, in part, decisions on allocating resources to resolve specific issues. Prioritization decisions by the NRC are documented in NUREG-0933, A Prioritization of Generic Safety Issues.

This is the sixth in a series of reports from the PSI project. The first report, the initial NUREG/CR-2800, contains a description of the methodology and three example issue analyses. The second report (Supplement 1) contains results of 15 additional issues. The third report (Supplement 2) contains results of analyses for 31 additional issues. The fourth report (Supplement 3) consists of two parts. Each part describes the results of a research effort that was undertaken in the human factors area. The first part, entitled "Estimating the Public Risk Reduction Affected by Human Factors Improvement," documents efforts to determine if currently used methods for assessing human-factors effects can be improved. The second part, entitled "Prioritization of the U.S. Nuclear Regulatory Commission Human Factors Program," summarizes the results of risk and cost analyses conducted by the Pacific Northwest Laboratory in support of efforts for the Human Factors Program Plan. The fifth report (Supplement 4) contains the results of the analyses for 23 additional issues.

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

NUREG/CR-2800 (PNL-4297)

18	Steam Line Break with Consequential Small LOCA
B-56	Diesel Generator Reliability
I.A.2.2	Training and Qualifications of Operations Personnel

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

23	Reactor Coolant Pump Seal Failures
B-6	Loads, Load Combinations, Stress Limits
B-10	Behavior of BWR Mark III Containments
B-26	Structural Integrity of Containment Penetrations
B-55	Improved Reliability of Target Rock Safety Relief Valves
B-58	Passive Mechanical Failures
C-8	Main Steam Line Leakage Control Systems
I.A.2.7	Accreditation of Training Institutions
I.C.1(4)	Confirmatory Analysis of Selected Transients
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities
II.C.2	Continuation of Interim Reliability Evaluation Program
II.C.3	Systems Interaction

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

II.C.4	Reliability Engineering
III.D.3.1	Radiation Protection Plans
IV.E.5	Safety Decision Making--Assess Currently Operating Reactors

NUREG/CR-2800 (PNL-4297) - SUPPLEMENT 2

15	Radiation Effects on Reactor Vessel Support Structures
A-18	Pipe Rupture Design Criteria
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage
C-11	Assessment of Failure and Reliability of Pumps and Valves
D-1	Advisability of a Seismic Scram--High Trip Level
I.A.2.6.(1-3,5)	Long-Term Upgrading of Training and Qualifications (Simulators)
I.A.2.6(4)	Long-Term Upgrading of Training and Qualifications (Simulators)
I.A.2.6.(6)	Long-Term Upgrading of Training and Qualifications (Nuclear Power Fundamentals for Operator Training)
I.A.3.3	Requirements for Operator Fitness
I.A.3.4	Licensing of Additional Operations Personnel
I.A.4.2	Long-Term Training Simulator Upgrade
I.B.1.1(5-7)	Management for Operations: Organization and Management of Long-Term Improvements
I.C.9	Long-Term Program Plan for Upgrading Procedures
I.D.3	Safety System Status Monitoring
I.D.4	Control Room Design Standard
I.D.5(3-5)	Control Room Design: Improved Control Room Instrumentation Research
I.F.2	Detailed QA Criteria for Design, Construction and Operation
II.B.5(1,2)/	Research on Phenomena Associated with Core Degradation and
II.B.8	Fuel Melting: Behavior of Severely Damaged Fuel, Behavior of Core Melt; Severely Damaged Core Rulemaking
II.D.2	Research on Relief and Safety Valve Test Requirements
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients
II.E.6.1	In-Situ Testing of Valves
II.F.5	Instrumentation and Controls: Classification of Instrumentation, Control, and Electrical Equipment
II.J.3.1/	Organization and Staffing to Oversee Design and Construction
II.J.3.2	Issue Regulatory Guide
II.J.4.1	Revise Deficiency Reporting Requirements
III.A.1.3	Maintain Supplies of Thyroid Blocking Agent (Potassium Iodide)
III.A.3.4	Nuclear Data Link
III.D.1.4	Radwaste System Design Feature to Aid in Accident Recovery and Decontamination
III.D.2.1	Radiological Monitoring of Effluents
III.D.2.2	Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis
III.D.2.5	Offsite Dose Calculation Manual
III.D.3.2	Worker Radiation Protection Improvement: Health Physics Improvements

NUREG/CR-2800 (PNL-4297) - SUPPLEMENT 3

I.A.1.4	Operator and Staffing: Long-Term Upgrades
I.A.2.2	Training and Qualifications of Operating
I.A.2.4	NRR Participation in Training
I.A.2.6(4)	Long-Term Upgrading of Training and Qualifications: Training Workshops
I.A.2.6(6)	Long-Term Upgrading of Training and Qualifications: Nuclear Power Fundamentals for Operator Training
I.A.2.6 (1,2,3,5)	Long-Term Upgrading of Training and Qualifications: Simulators
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NUREG/CR-2800 (PNL-4297) - SUPPLEMENT 4

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45	Inoperability of Instruments due to Extreme Cold
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57	Effects of Fire Protection System Actuation Interaction on Safety Related Equipment/Systems
58	Inadvertent Containment Flooding
59	Technical Safety Requirements for Plant Shutdown When Equipment Required for Safe Shutdown is Degraded or Inoperable
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61	SRV Line Break Inside BWR Wetwell Airspace of Mark I and Mark II Containments
63	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis
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The results contained in this report have been prepared by a number of issue authors over the last twelve years (February 1983 through February 1995). The following table lists the release date and author for each prioritization analysis in this report:

<u>Issue Number</u>	<u>Date Released</u>	<u>Author(s)^(a)</u>
15	November 1988	P.M. Daling
24	December 1990	P.M. Daling
43	October 1987	M.S. Harris
55	February 1983	W. A. Bickford R.H.V. Gallucci
76	January 1991	P.M. Daling
78	April 1992	P.M. Daling
81	October 1991	P.M. Daling
89	April 1992	P.M. Daling R.H.V. Gallucci
95	January 1989	M.S. Harris
100	February 1991	P.M. Daling
106	1988	T. B. Powers
107	June 1988	B.W. Smith
110	January 1992	P.M. Daling
115	1986	
117	February 1985	P.M. Daling
120	June 1988	B.W. Smith
123	February 1991	P.M. Daling
129	September 1986	W.L. Ferrell ^(b) W.B. Andrews ^(b) A.S. Tabatabai
132	August 1991	P.M. Daling
137	October 1987	P.M. Daling J.V. Dovey
138	December 1988	P.M. Daling
142	May 1989	P.M. Daling

<u>Issue Number</u>	<u>Date Released</u>	<u>Author(s)^(a)</u>
143	February 1995	W.C. Geckler ^(c) T.J. McIntyre ^(c) S.R. Rao ^(c) P.M. Daling
144	June 1991	P.M. Daling
147	May 1990	P.M. Daling S. Veitenheimer
148	August 1991	P.M. Daling
149	July 1992	P.M. Daling R.H.V. Gallucci
150	September 1990	P.M. Daling
151	September 1990	P.M. Daling E.A. Banaghan
152	March 1992	R.H.V. Gallucci
154	May 1991	P.M. Daling
158	August 1993	P.M. Daling H.K. Phan
160	March 1993	P.M. Daling H.K. Phan T.M. Mitts
167	February 1994	

- (a) Unless identified otherwise at the time the issue was prepared author was employed by PNL.
(b) Science Applications International Corporation.
(c) Pickard, Lowe, and Garrick.

1.0 INTRODUCTION

This report documents the use of a methodology developed by the Pacific Northwest Laboratory¹ to provide the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) with information to use in prioritizing 34 safety issues related to nuclear power plants. Estimates in this report, along with other subjective factors, were used by the NRC to rank safety issues for further investigation or possible implementation. The safety issue ranking decisions made by NRC are documented in NUREG-0933 (Emrit, et al., 1983).

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

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

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

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

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

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

¹

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workers. Person-rem is chosen as the risk/dose measure for simplicity and for convenient relationship with most safety effects. Models used to calculate person-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 safety issue resolution (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:

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

where N = number of reactors affected by the SIR
 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 (reactor-years)⁻¹ and public consequences per accident in person-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:

$$\begin{aligned}
 G &= \text{occupational dose increase due to implementation and O/M of the SIR} \\
 &= N(TD_0 + D) \text{ in man-rem}
 \end{aligned}$$

where N = number of reactors affected by the SIR
 T = average remaining operating life of reactors affected (yrs)
 D_0 = annual incremental dose increase due to O/M of the SIR (person-rem/reactor-year)
 D = incremental dose increase due to implementation of the SIR (person-rem/reactor).

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

$$\begin{aligned}
 \Delta U &= \text{change, due to the SIR, in the accident-frequency-weighted occupational dose from cleanup and repair of a reactor following an accident (person-rem)} \\
 &= NT \Delta(FD_R)
 \end{aligned}$$

where N = number of reactors affected by the SIR
 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 (reactor-years)⁻¹ and occupational dose due to cleanup and repair of the reactor following an accident (person-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 in 1982 dollars.

1.3.1 NRC Costs

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

$$\begin{aligned} (SN)_{\text{Total}} &= \text{Future cost to the NRC for SIR development, support of SIR implementation, and review of SIR O/M } (\$10^6) \\ &= C_D + N(TC_0 + C) \end{aligned}$$

where

N	=	number of plants affected by the SIR
T	=	average remaining operating life of reactors affected (years)
C _D	=	future NRC costs for SIR development (\$10 ⁶)
C ₀	=	annual incremental NRC costs for review of SIR O/M (\$10 ⁶ /reactor-year)
C	=	incremental NRC costs for support of SIR implementation (\$10 ⁶ /reactor).

1.3.2 Industry Costs

Industry costs are defined as follows:

$$\begin{aligned} SI &= \text{future costs to the industry for SIR implementation and O/M } (\$10^6) \\ &= N(TI_0 + I) \end{aligned}$$

where

N	=	number of reactors affected
T	=	average remaining operating life of reactors affected (years)

I_o = annual incremental industry costs for SIR O/M
 (\$10⁶/reactor-year)
 I = incremental industry costs for SIR implementation
 (\$10⁶/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
 (\$10⁶)
 = $NT \Delta(FA)$

where N = number of reactors affected
 T = average remaining operating life of reactors affected (years)
 $\Delta(FA)$ = change due to the SIR, in the product of estimated time
 frequency of affected accidents in (reactor-years) and cost of
 cleanup, repair and replacement power following an accident
 (\$10⁶).

1.4 REFERENCES (for SECTION 1.0)

Andrews, W. B., R. H. V. Gallucci, S. W. Heaberlin, W. E. Bickford, G. J. Konzek, D. L. Streng, R. I. Smith, and S. A. Weakley. 1983. *Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development*. NUREG/CR-1800, PNL-4297. Pacific Northwest Laboratory, Richland, Washington.

Emrit, R., W. Minners, H. VanderMolen, R. Colmar, D. Thatcher, J. Pittman, W. Milstead, R. Riggs, G. Sege, P. Matthews, and L. Riani. 1983. *A Prioritization of Generic Safety Issues*. NUREG-0933. U.S. Nuclear Regulatory Commission, Washington, D.C.

2.0 ISSUE ANALYSES

Thirty-four issue analyses are described in this section. Nearly all are similar in format and contain the following components:

- Safety Issue Summary Work Sheet - Results are summarized for the issue.
- Section 1.0, Issue Description - The safety issue resolution (SIR) and affected plants are described.
- Section 2.0, Safety Issue Risk - and Dose Analysis of public risk reduction and the occupational dose resulting from the SIR is presented. Results are summarized in the Public Risk Reduction Work Sheet and the Occupational Dose Work Sheet, respectively.
- Section 3.0, Safety Issue Costs - Analysis of industry and NRC costs attributable to the SIR is presented. Results are summarized in the Safety Issue Cost Work Sheet.
- Attachment to Sections - Attachments are often included immediately following the section, or table to which they apply.
- References - Included at the end of each issue.

ISSUE SUMMARY WORK SHEET

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

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: This issue was previously reviewed in NUREG-0933 (NRC 1983) and it was concluded that the occupational dose increase associated with resolving the issue far outweighed the public risk reduction. However, recent data suggests that the potential embrittlement of reactor vessel support structures (RVSSs) as a result of neutron irradiation damage may be significantly greater than previously anticipated. The proposed resolution to this issue involves installation and operation of local heaters to maintain the RVSSs well above the range of brittle fracture temperatures. At planned plants, use of nonsusceptible structural steel is the proposed resolution.

AFFECTED PLANTS: BWR: Operating = 1 Planned = 0
PWR: Operating = 26 Planned = 34

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 2.3E+4

OCCUPATIONAL DOSES:

SIR Implementation =	1420
SIR Operation/Maintenance =	3640
Total of Above =	5100
Accident Avoidance =	330

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	6.2
SIR Operation/Maintenance =	14
Total of Above =	20
Accident Avoidance =	28

NRC COSTS:

SIR Development =	0.54
SIR Implementation =	0.61
SIR Operation/Maintenance Review =	0.70
Total of Above =	1.9

RADIATION EFFECTS ON REACTOR VESSEL SUPPORT STRUCTURES

ISSUE 15 (Revised)

1.0 SAFETY ISSUE DESCRIPTION

This issue addresses the potential problem of radiation embrittlement of reactor vessel support structures (RVSSs). This issue was previously addressed in NUREG-0933 (NRC 1983) and it was concluded that the occupational doses associated with resolving the issue far outweighed the potential decrease in public risks. As a result, the issue was assigned a low priority until additional data on the problem became available.

Neutron damage of structural materials causes embrittlement that may increase the potential for propagation of flaws that might exist in the materials. The potential for brittle fracture of these materials is typically measured in terms of the material's nil ductility transition temperature (NDTT), which is the highest temperature at which the material would not be susceptible to failure by brittle fracture. As long the operating environment in which the materials are used has a higher temperature than the material's NDTT, no failure by brittle fracture would be expected. Many materials, when subjected to neutron irradiation, experience an upward shift in the NDTT; i.e., they become more susceptible to brittle fracture at the operating temperatures of interest. This effect is accounted for in the design and fabrication of RVSSs. However, recent research at the Oak Ridge National Laboratory (ORNL; Cheverton et al., 1988a, Cheverton et al., 1988b) indicates that the upward shift in NDTT with increased exposure to neutron irradiation has been underestimated. The loss in fracture toughness may result in failure of the RVSSs and consequent movement of the reactor vessel, given the occurrence of a transient stress or shock such as would be experienced in an earthquake.

Cheverton et al. (1988b) surveyed RVSS designs at LWRs and categorized each plant into one of five categories or types of RVSSs. These five categories were: 1) skirt, 2) long-column, 3) shield-tank, 4) short-column, and 5) suspension. Skirt type supports are located away from the core with a large volume of intervening metal and water. Radiation embrittlement of skirt type RVSSs is not anticipated. Long-column type supports are located in a zone of potentially high neutron fluence and are thus susceptible to radiation damage. Similarly, shield-tank supports are also located in a potentially high radiation damage zone. Short-column type supports include several subcategories that are located in various regions relative to the reactor core. Thus, they appear to have a wide variability in susceptibility to radiation damage. Many plants with this type of support have special designs for heat dissipation, including natural convection, forced convection, and water/cooling-coil designs. The fifth category, suspension supports, are employed at only one plant. Although the supports are located in a region of potentially high irradiation damage, the temperature is high enough to preclude brittle fracture. As a result, plants employing the long-column, shield-tank, short-column, and suspension type supports are assumed to be susceptible to irradiation damage.

The number of potentially susceptible plants was determined based on the results of the survey performed by Cheverton et al. (1988b) and are summarized below:

Plant Type	RVSS Type	Number of Affected Plants	
		Operating	Under Construction
PWR	Short-column	29	27
	Long-column	8	1
	Shield-tank	<u>6</u>	<u>6</u>
	TOTAL:	43	34
BWR	Suspension	1	0

Plant status, either operating or under construction, was determined based on information in NUREG/CR-2800, Appendix D (Andrews et al., 1983). The number of potentially affected plants listed above (78) is substantially larger than the number of plants examined in the previous analysis of this issue (27).

PROPOSED RESOLUTION

The proposed resolution is the same as that proposed in the previous analysis of this issue. This involves application of local heaters and insulation for the RVSSs to maintain operating temperatures well above the NDTT of the potentially embrittled support. This resolution would only involve those plants that employ long-column and shield-tank supports. Short-column and suspension supports are in a higher-temperature environment than the others and thus heaters are not necessary to maintain the temperatures above the NDTT. However, minor design and equipment changes would be needed to control the amount of heat dissipation applied to the short-column and suspension supports to ensure the NDTT of the structural materials do not exceed the environmental temperature. The proposed resolution for plants under construction is to use structural steel that is not susceptible to radiation-induced shifts in NDTT.

2.0 SAFETY ISSUE RISK AND DOSE

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

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Radiation Effects on Reactor Vessel Support Structures (15)

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

The number and average remaining lives of affected plants are derived in Attachment 1. The results are:

	$\frac{N}{\text{PWR}}$	$\frac{\bar{T} \text{ (yr)}}{7}$
PWR	78	7
BWR	1	7

3. Plants Selected for Analysis:

Representative PWR - Oconee 3
Representative BWR - Grand Gulf 1

4. Parameters Affected by SIR:

Oconee: SS_1, SS_2, SS_3
Grand Gulf: SS

These are seismically-induced LOCA initiators analogous to S_1, S_2, S_3 , and S (see Attachment 1).

5. Base-Case Values for Affected Parameters:

$S_1 = S_2 = S_3 = S = 3.5E-4/\text{py}$ (see Attachment 1).

6. Affected Accident Sequences and Base-Case Frequencies:

Oconee

SS_3H-	τ (PWR-3)	=	$1.4E-6/\text{py}$
	β (PWR-5)	=	$2.0E-8/\text{py}$
	ϵ (PWR-7)	=	$1.4E-6/\text{py}$
$SS_1D -$	α (PWR-1)	=	$2.4E-7/\text{py}$
	τ (PWR-3)	=	$4.8E-6/\text{py}$
	β (PWR-5)	=	$1.8E-7/\text{py}$
	ϵ (PWR-7)	=	$1.9E-5/\text{py}$

SS ₃ FH -	τ (PWR-2)	=	6.0E-7/py
	β (PWR-4)	=	8.8E-9/py
	ϵ (PWR-6)	=	6.0E-7/py
SS ₂ FH -	α (PWR-1)	=	1.1E-8/py
	β (PWR-4)	=	8.0E-9/py
	ϵ (PWR-6)	=	8.8E-7/py
	α (PWR-1)	=	1.8E-8/py
	τ (PWR-3)	=	3.6E-7/py
SS ₂ D -	β (PWR-5)	=	1.3E-8/py
	ϵ (PWR-7)	=	1.4E-6/py
SS ₃ D -	τ (PWR-3)	=	1.9E-7/py
	β (PWR-5)	=	2.7E-9/py
	ϵ (PWR-7)	=	1.9E-7/py

Grand Gulf

S ₁ -	α (BWR-1)	=	1.2E-8/py
	δ (BWR-2)	=	1.2E-6/py

These frequencies were derived by substituting the frequency of the seismically-induced initiators into the minimal cut sets given in Andrews et al. (1983).

7. Affected Release Categories and Base-Case Frequencies:

<u>Oconee</u>	<u>Grand Gulf</u>
PWR-1 = 2.7E-7/py	BWR-1 = 1.2E-8/py
PWR-2 = 6.0E-7/py	BWR-2 = 1.2E-6/py
PWR-3 = 6.8E-6/py	
PWR-4 = 1.7E-8/py	
PWR-5 = 2.2E-7/py	
PWR-6 = 1.5E-6/py	
PWR-7 = 2.2E-5/py	

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

$$\bar{F}(\text{PWR}) = 3.1\text{E-}5/\text{py} \quad \bar{F}(\text{BWR}) = 1.2\text{E-}6/\text{py}$$

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

$$W(\text{PWR}) = 41.6 \text{ man-rem/py} \quad W(\text{BWR}) = 8.6 \text{ man-rem/py}$$

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

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

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

$$\Delta\bar{F}(\text{PWR}) = 3.1\text{E-}5/\text{py} \qquad \Delta\bar{F}(\text{BWR}) = 1.2\text{E-}6/\text{py}$$

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

$$\Delta W(\text{PWR}) = 41.6 \text{ man-rem/py} \quad \Delta W(\text{BWR}) = 8.6 \text{ man-rem/py}$$

17. Total Public Risk Reduction, (ΔW)Total:

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
2.3E+4	6.8E+4	0

ATTACHMENT 1 (To Table 1)

The numbers of affected plants, as discussed previously, were determined from a recent survey of plant designs (Cheverton et al., 1988b). This report also provided the basis for estimates of the length of time a plant could potentially operate in a vulnerable condition; i.e., with embrittled reactor vessel supports. Cheverton et al. investigated the radiation embrittlement of RVSS materials from two operating LWRs and provided data on the change in NDTT over time. These NDTT data were plotted as a function of reactor operating years. Next, the arithmetic averages of the plotted NDTTs were computed and the averages were plotted as a function of reactor operating years. Then, a range of RVSS operating temperatures were overlaid on the plot of average NDTT vs. operating time. These curves are shown in Figure 1. The operating time at which the "average" curve intersects the operating temperature curve is the approximate time at which the RVSS material has become susceptible to brittle fracture. This occurs at 23 years after the reactor has begun full-power operation. Therefore, the potential susceptibility of the RVSS to brittle fracture exists for 7 years at the end of a reactor's lifetime, assuming an average operating lifetime of 30 years.

The assumed accident scenario is occurrence of a seismic event of sufficient magnitude to cause fracture of an embrittled RVSS, subsequent movement of the reactor vessel, and a corresponding LOCA as attached piping ruptures. The analogous accident sequences are those involving LOCA initiators S_1 , S_2 , and S_3 for Oconee (different initiator frequencies for three pipe diameters) and S for Grand Gulf. These are the corresponding LOCA initiators for pipe ruptures. However, this issue is concerned with only seismically-induced pipe ruptures, which were not addressed in the original Oconee and Grand Gulf studies. As a result, seismically-induced LOCAs are defined here and incorporated into the base case.

The base-case frequencies of seismically-induced LOCA initiators SS_1 , SS_2 , SS_3 , and SS are assumed to be equal; i.e., the conditional probabilities of fracturing different sizes of pipe given an earthquake are assumed to be equal. Their base-case frequencies are estimated as follows:

$$f(SS_1) = f(SS_2) = f(SS_3) = f(S) = f(PGA \geq 0.2g) * p(NDTT) * p(PR)$$

where: $f(PGA \geq 0.2g)$ = frequency of a seismic event with peak ground acceleration greater than or equal to 0.2g; annual frequency = $7E-4/yr$ (NRC 1975).

$p(NDTT)$ = conditional probability that a RVSS is susceptible to radiation damage and fails as a result of reactor vessel movement (this value is derived below)

$p(PR)$ = conditional probability of pipe rupture given movement of the reactor vessel (assumed to be accounted for in estimate of $p(NDTT)$; effectively 1.0 for pipes of all diameters).

The conditional probability of failure of an embrittled RVSS as a result of a seismic event [$p(NDTT)$] is a function of the NDTT at the time the seismic

event occurs, the number and size of preexisting flaws in the support material, and the safety factor built into the design of the supports and selection of the material. As discussed above, the RVSS materials at some plants may exceed operating temperatures during the last seven years of reactor operation. Assuming that this occurs, the safety factor built into the RVSS may not exceed 1.0, whereas using previous predictions of radiation damage this safety factor may be as much as 20. Using a correlation between safety factor and failure probability presented by Hsieh and Okrent (1976), the conditional probability of failure leading to reactor core damage for a safety factor of 1 is 0.5. Using this value, the frequency of seismically-induced LOCAs is:

$$\begin{aligned} f(SS_1) = f(SS_2) = f(SS_3) = f(S) &= 7E-4/py * 0.5 * 1.0 \\ &= 3.5E-4/py \end{aligned}$$

FIGURE 1: Relationship Between NDTT and Reactor Operating Years

Attachment 2¹ (To Table 1)

Long Column Supports

Operating Plants (10)

Arkansas Nuclear One - 2
Prarie Island - 1,2
San Onofre - 2,3
St. Lucie - 1,2
Palo Verde - 1,2,3

Under Construction (1)

Washington Nuclear - 3

Shield Tank Supports

Operating Plants (8)

North Anna - 1,2
Beaver Valley - 1,2
Scurry - 1,2
Maine Yankee
Millstone - 3

Under Construction (0)

Short Column Supports

Operating Plants (45)

Calvert Cliffs - 1,2
Cook - 1,2
Davis Besse
Farley - 1,2
Fort Calhoun
Ginna
Haddam Neck
Indian Point - 2,3
Kewaunee
McGuire - 1,2
Millstone - 2
Catawba - 1,2
Palisades
Point Beach - 1,2
Robinson - 2
Salem - 1,2
Sequoyah - 1,2
Trojan
Turkey Point - 3,4
Zion - 1,2
Harris - 1
Byron 1,2
Calloway
Diablo Canyon 1,2
San Onofre 1
Summer 1
Waterford 3
Wolf Creek 1
Yankee Rowe
Vogtle
Braidwood - 1,2

Under Construction (13)

Vogtle - 2
Bellefonte - 1,2
Comanche Peak - 1,2
South Texas Project - 1,2
Seabrook - 1,2²
Harris^(b)
Watts Bar - 1,2
Washington Nuclear - 1

¹ Plant status given as of June, 1988 (American Nuclear Society, 1988). Type of RVSS given by Cheverton et al. (1988).

² Seabrook 2 and Harris 2 have been cancelled. Included in totals.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Radiation Effects on Reactor Vessel Support Structures (15)

2. Affected Plants (N):

Affected backfit plants are assumed to implement the resolution after about ten years of reactor operation. It is further assumed that only plants with long-column and shield-tank type supports will install and operate local heaters and insulation on their RVSSs. The plants with suspension and short-column type supports are assumed to implement measures to control or limit cooling of the RVSSs. Affected forward-fit plants will implement the SIR before fuel is loaded into the core. Therefore, no occupational dose will be incurred during SIR implementation, operation, and maintenance at forward-fit plants. However, occupational dose reduction due to accident avoidance will be realized at these forward-fit plants, as well as at backfit plants, over the last 7 years of reactor operation.

	<u>N</u>
PWRs: Backfit	
Heaters	14
Cooling Control	29
Forward-fit	34 (accident-avoidance dose only)
BWRs Backfit (cooling)	1

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

	<u>$\bar{T}$ (yr)</u>
PWR	7
BWR	7

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

$$\begin{aligned}\Delta(\bar{F}D_r)_{PWR} &= (19900 \text{ man-rem})(3.1E-5/\text{py}) = 0.62 \text{ man-rem/py} \\ \Delta(\bar{F}D_r)_{BWR} &= (19900 \text{ man-rem})(1.2E-6/\text{py}) = 0.024 \text{ man-rem/py}\end{aligned}$$

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

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
330	2000	0

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

Labor in radiation zones to install and test the RVSS heaters and insulation at backfit plants was estimated to require 21 man-wk/plant (including a 50% contingency factor). This estimate applies to only those plants with long-column and shield-tank type supports. It was estimated

that 10% of the above labor estimate (2 man-wks/plant) would be required for implementation of cooling control to the supports. This estimate applies to the plants with short-column and suspension type supports.

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

Radiation fields of 100 millirem/hr are assumed to exist in the vicinity of the reactor vessel.

$$\begin{aligned} \text{(Heaters) } D &= (21 \text{ man-wk/plant})(40 \text{ man-hr/man-wk})(0.1 \text{ rem/hr}) \\ &= 84 \text{ man-rem/plant} \\ \text{(Cooling) } D &= (2 \text{ man-wk/plant})(40 \text{ man-hr/man-wk})(0.1 \text{ rem/hr}) \\ &= 8 \text{ man-rem/plant} \end{aligned}$$

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

$$\begin{aligned} ND &= 14 \text{ plants}(84 \text{ man-rem/plant}) + 30 \text{ plants}(8 \text{ man-rem/plant}) \\ &= 1420 \text{ man-rem} \end{aligned}$$

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

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

$$\begin{aligned} \text{Routine operation and maintenance} &= 2 \text{ man-wk/py} \\ \text{Periodic heater replacement} & \\ \text{(each heater replaced once)} & \\ \text{Installation} & 8 \text{ man-wks} \\ \text{Testing} & 4 \text{ man-wks} \\ \text{Contingency (@50\%)} & 6 \text{ man-wks} \\ & 18 \text{ man-wks} \div 7 \text{ yrs} \\ \text{TOTAL} &= \frac{3 \text{ man-wks/py}}{5 \text{ man-wks/py}} \end{aligned}$$

This estimate applies to backfit plants with long-column and shield-tank type supports. Utility labor for operation and maintenance at plants that employ short-column and suspension type supports is assumed to consist only of the routine operation and maintenance portion of the above estimate, or 2 man-wk/py. Forward-fit plants are assumed to use nonsusceptible materials in their RVSSs, and no additional labor is foreseen beyond that for routine inspection and maintenance.

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

As with Step 7, a radiation field of 100 mrem/hr is assumed.

$$\begin{aligned} \text{(Heaters) } D_o &= (5 \text{ man-wk/py})(40 \text{ man-hr/man-wk})(0.1 \text{ rem/hr}) \\ &= 20 \text{ man-rem/py} \\ \text{(Cooling) } D_o &= (2 \text{ man-wk/py})(40 \text{ man-hr/man-wk})(0.1 \text{ rem/hr}) \\ &= 8 \text{ man-rem/py} \end{aligned}$$

11. Total Occupational Dose Increase for SIR Operation/Maintenance, (NTD_o):

$$\begin{aligned} (\text{NTD}_o) &= (14)(7 \text{ yrs})(20 \text{ man-rem/py}) + (30)(7 \text{ yrs})(8 \text{ man-rem/py}) \\ &= 3640 \text{ man-rem} \end{aligned}$$

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
5100	1.5E+4	1700

3.0 SAFETY ISSUE COSTS

The industry and NRC costs for development, implementation, operation, and maintenance of the proposed resolution to this issue are estimated in this section. At operating plants, SIR consists of controlling the temperature of the RVSSs, either through application of local heaters and insulation or through controlling cooling systems that are already in place, to ensure that the temperatures do not exceed the materials NDTT after irradiation embrittlement. At planned plants, the use of nonsusceptible materials is the proposed resolution. Since this can be accommodated during the design and construction stages of a plant, no additional costs are foreseen beyond those normally incurred during design and construction.

Table 3 presents the estimates of industry and NRC costs.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Radiation Effects on Reactor Vessel Support Structures (15)

2. Affected Plants (N):

See Step 2 of the Occupational Dose Worksheet, Table 2.

	<u>N</u>
PWRs: Backfit	
Heaters	14
Cooling Control	29
Forward-fit	34 (accident-avoidance cost only)
BWRs Backfit (cooling)	1

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

See step 3 of the Occupational Dose Worksheet, Table 2.

	<u>$\bar{T}$ (yr)</u>
PWR	7
BWR	7

Industry Costs (Steps 4 through 12):

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

$$\begin{aligned}\Delta(\bar{F}A)_{\text{PWR}} &= (\$1.65\text{E}+9)(3.1\text{E}-5/\text{py}) = \$5.1\text{E}+4/\text{py} \\ \Delta(\bar{F}A)_{\text{BWR}} &= (\$1.65\text{E}+9)(1.2\text{E}-6/\text{py}) = \$2.0\text{E}+3/\text{py}\end{aligned}$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$2.8E+7	\$1.7E+8	0

6. Per-Plant Industry Resources for SIR Implementation:

This step estimates the equipment and labor requirements necessary to implement the SIR at the affected plants. For plants with long-column and shield-tank type supports, it is assumed that heaters will be attached to four reactor vessel support columns and that mounting hardware, metal-sheathed heating cables, switchgear, transformers, and a power controller will be installed. It is also assumed that the equipment will be installed during scheduled reactor outages. Therefore, no additional replacement power costs would be necessary. It is further assumed that access to the reactor cavity is possible for heater installation.

For plants with short-column and suspension type supports, it is assumed that equipment and labor requirements are 10% of that estimated for application of local heaters and insulation.

Labor: (Heaters) - 105 man-wk/plant
(Cooling) - 11 man-wk/plant

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

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

<u>Heaters</u>	
Equipment cost (50% contingency)	= \$5.2E+4/plant
Labor cost = (105 man-wk/plant)(\$2270/man-wk)	= \$2.4E+5/plant
Class V License Amendment Fee	= <u>\$2.6E+4/plant</u>
	I = \$3.2E+5/plant
<u>Cooling</u>	
Equipment cost (50% contingency)	= \$5.2E+3/plant
Labor cost = (11 man-wk/plant)(\$2270/man-wk)	= \$2.5E+4/plant
Class V License Amendment Fee	= <u>\$2.6E+4/plant</u>
	I = \$5.6E+4/plant

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

$$\begin{aligned} NI &= (14 \text{ plants})(\$3.2E+5) + (30 \text{ plants})(\$5.6E+4) \\ &= \$6.2E+6 \end{aligned}$$

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

Routine operation and maintenance (estimated)
 Operation = 1 man-wk/py
 Maintenance = 1 man-wk/py
 Subtotal = 2 man-wk/py

Periodic heater replacement:

Heater replacement is assumed to occur after five years of heater operation. It is estimated that 30 man-wk/plant, including a 50% contingency factor, will be required to retrain staff, conduct tests, fabricate, and install replacement heater elements (see next step for equipment costs)

Annual labor/plant:

$$\text{Total} = 2 + (30 \text{ man-wks/plant} \div 5 \text{ yrs}) = 8 \text{ man-wks/py}$$

This estimate applies to plants with long-column and shield-tank type supports. For plants with short-column and suspension type supports, the annual labor requirements are assumed to be 50% of the routine operation and maintenance labor given above plus 25% of the replacement labor estimate for a total of 3 man-wks/py.

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

Heaters

$$\text{Labor cost} = (8 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$1.8\text{E}+4/\text{py}$$

$$\text{Material cost} = (\text{heater cost including 50\% contingency} \div 5 \text{ yrs}) = \$2.8\text{E}+3/\text{py}$$

$$\begin{aligned} \text{Power costs for operating 300 kW heaters} \\ \text{with 80\% availability at a unit cost of} \\ \$0.05/\text{kW-hr} = (365 \text{ days/py})(24 \text{ hr/day})(0.8) \\ (300 \text{ kW})(\$0.05/\text{kW-hr}) \end{aligned} \quad \begin{aligned} &= \$1.1\text{E}+5/\text{py} \\ I_o &= \$1.3\text{E}+5/\text{py} \end{aligned}$$

Cooling Control

$$\text{Labor cost} = (3 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$6.8\text{E}+3/\text{py}$$

$$\text{Material cost} = (10\% \text{ of above estimate}) = \$2.8\text{E}+2/\text{py}$$

$$\begin{aligned} \text{Power costs} &= 0 \\ I_o &= \$7.1\text{E}+3/\text{py} \end{aligned}$$

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

$$\begin{aligned} \overline{NTI}_o &= (14)(7\text{yrs})(\$1.3\text{E}+5/\text{py}) + (30)(7 \text{ yrs})(\$7.1\text{E}+3/\text{py}) \\ &= \$1.4\text{E}+7 \end{aligned}$$

12. Total Industry Cost (S_i):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$2.0E+7	\$2.8E+7	\$1.2E+7

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC staff labor = 16 man-wk
Contractor support (cost estimated directly in next step)

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

$$\begin{aligned} \text{Labor} &= (16 \text{ man-wk})(\$2270/\text{man-wk}) = \$3.6\text{E}+4 \\ \text{Contractor support} &= \$5.0\text{E}+5 \\ C_D &= \$5.4\text{E}+5 \end{aligned}$$

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

NRC labor is estimated to be about 15% of industry labor or about 15 man-wks/plant for plants with long-column and shield-tank type supports and 2 man-wks/plant for plants with short-column and suspension type supports.

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

$$\begin{aligned} \text{Heaters: } C &= (15 \text{ man-wks/plant})(\$2270/\text{man-wk}) = \$3.4\text{E}+4/\text{plant} \\ \text{Cooling: } C &= (2 \text{ man-wks/plant})(\$2270/\text{man-wk}) = \$4.5\text{E}+3/\text{plant} \end{aligned}$$

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

$$\begin{aligned} NC &= (14 \text{ plants})(\$3.4\text{E}+4/\text{plant}) + (30 \text{ plants})(\$4.5\text{E}+3/\text{plant}) \\ &= \$6.1\text{E}+5 \end{aligned}$$

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

Assumed to be 1 man-wk/py for all affected plants.

19. Per-Plant NRC Cost for Review of SIR Operation and Maintenance (C_0):

$$C_0 = (1.0 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$2270/\text{py}$$

20. Total NRC Cost for Review of SIR Operation and Maintenance ($N\bar{C}_0$):

$$N\bar{C}_0 = (14 + 30 \text{ plants})(7 \text{ yrs})(\$2270/\text{py}) = \$7.0\text{E}+5$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.9E+6	\$2.4E+6	\$1.4E+6

ATTACHMENT 2

Industry Resources for SIR Implementation

For reactor vessel supports of the long-column and shield-tank types that may require improvement in reliability, it is assumed that electric heaters could be installed at backfit plants and operated at a temperature well above the NDTT for a radiation-embrittled support. The equipment required and costs are tabulated below:

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

<u>Labor, by task</u>	<u>Man-wk/plant</u>
Develop design changes and drawings	8
Analyze, document, and process through NRC approval	30
Procure equipment specifications, purchase, and inspect	3
Plan implementation effort	2
Train staff	2
Change plant procedures	3
Fabricate heater assemblies	8
Complete installation:	
Mockup *	2
Installation *	8
Conduct final tests *	<u>4</u>
Labor subtotal	70
Contingency (@ 50%)	<u>35</u>
TOTAL	105

* Activities performed in radiation zones.

Cooling capabilities are currently provided for reactor vessel supports of the short-column and suspension types. Analyses and minor design changes to remove or limit the amount of cooling are assumed to be required at these plants to prevent the support structures from operating at temperatures below their NDTT. Equipment and labor requirements are assumed to be approximately 10% of that estimated for installation of local heaters.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 24, Automatic Emergency Core Cooling Switch to Recirculation

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: The emergency core cooling system (ECCS) operation has two different phases: (1) injection phase; and (2) recirculation phase. The first phase involves initial cooling of the reactor core and replenishment of the primary coolant following a Loss of Coolant Accident (LOCA). The second phase provides long-term cooling during the accident recovery period. Switchover from injection to recirculation phases involves realignment of several valves. Switchover may be accomplished by manual operations to realign the valve, fully automatic realignment of the valves, or automatic realignment of some valves followed by manual completion of the process (semi-automatic switchover). Each option is vulnerable in varying degrees to human errors, hardware failures, and common cause failures. The public risk, occupational doses, and costs for implementing automatic and semi-automatic switchover options relative to the base-case manual switchover option are evaluated.

AFFECTED PLANTS: Only operating and planned PWRs are affected by this issue.

RISK/DOSE RESULTS (man-rem):

	<u>Automatic Switchover</u>	<u>Semi-Automatic Switchover</u>
PUBLIC RISK REDUCTION =	1.2E+4	1.1E+4
OCCUPATIONAL DOSES:		
SIR Implementation =	45	45
SIR Operation/Maintenance =	52	52
Total of Above =	97	97
Accident Avoidance =	93	77

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	9.9	9.9
SIR Operation/Maintenance =	5.6	5.9
Total of Above =	16	16
Accident Avoidance =	7.7	6.4

NRC COSTS:

SIR Development =	0.13	0.13
SIR Implementation =	1.6	1.6
SIR Operation/Maintenance Review =	2.8	2.8
Total of Above =	4.5	4.5

AUTOMATIC EMERGENCY CORE COOLING SWITCHOVER TO RECIRCULATION

ISSUE 24

1.0 SAFETY ISSUE DESCRIPTION

The emergency core cooling system (ECCS) operation has two different phases: (1) injection phase; and (2) recirculation phase. The first phase involves initial cooling of the reactor core and replenishment of the primary coolant following a Loss of Coolant Accident (LOCA). In this phase, the ECCS pumps typically take suction from the Refueling Water Storage Tank (RFWS). The second phase provides long-term cooling during the accident recovery period. The ECCS is realigned in the recirculation phase to take suction off the containment sump.

Switchover from injection to recirculation phases involves realignment of several valves. Switchover may be accomplished by manual operations to realign the valves, fully automatic realignment of the valves, or automatic realignment of some valves followed by manual completion of the process (semi-automatic switchover). Each option is vulnerable in varying degrees to human errors, hardware failures, and common cause failures. The public risk, occupational doses, and costs for implementing automatic and semi-automatic switchover options relative to the base-case manual switchover option are evaluated in this analysis.

PROPOSED RESOLUTION

The effects of two potential proposed resolution to this issue are examined. The first is to implement a fully-automatic switchover system which would monitor the water level in the RFWS Tank and upon exhausting this supply of water would automatically realign the ECCS to take suction of the containment sump. The second option is to implement semi-automatic switchover systems which would involve automatic alignment of some valves and manual completion of the switchover process.

AFFECTED PLANTS

Resolution of this issue is assumed to affect only PWR nuclear power plants, including those operating and under construction. BWRs are not affected by this issue.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this issue are estimated in this section and summarized in Tables 1 and 2. Attachment 1 describes the basis for the risk and dose changes that are estimated in the Tables.

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

Automatic Emergency Core Cooling System Switch to Recirculation (24).

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

All PWRs but no BWRs are affected.

	$\frac{N}{90}$	$\frac{\bar{T}}{28.8}$
PWRs		

3. Plants Selected for Analysis:

Representative PWR: Oconee 3

4. Parameters Affected by SIR:

New parameters RSF(n) were introduced into the Oconee PRA as described in Attachment 1. For the modified base-case, the affected parameters are RSF1 and RSF2.

5. Base-Case Values for Affected Parameters:

RSF1 = 1.3E-3/Demand

RSF2 = 1.0E-4/Demand

6. Affected Accident Sequences and Base-Case Frequencies:

Based on the Oconee 3 analysis (see Attachment 1):

T_2 MQFH	γ	(PWR-2) = 1.3E-6/RY
	β	(PWR-4) = 1.9E-8/RY
	ϵ	(PWR-6) = 1.3E-6/RY

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

S_2 FH	α	(PWR-1) = 9.7E-10/RY
	β	(PWR-4) = 7.1E-10/RY
	ϵ	(PWR-6) = 7.8E-8/RY
S_3 FH	γ	(PWR-2) = 2.1E-7/RY
	β	(PWR-4) = 3.0E-9/RY
	ϵ	(PWR-6) = 2.1E-7/RY

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 9.7E-10/RY
PWR-2 = 1.5E-6/RY
PWR-4 = 2.3E-8/RY
PWR-6 = 1.6E-6/RY

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

Oconee 3 $\bar{F}$ = 3.1E-6/RY

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

Oconee 3 W = 7.5 person-rem/RY

10. Adjusted-Case Values for Affected Parameters:

See Attachment 1 for derivation of the following values.

Automatic Switchover

RSF3 = 1.0E-4 RSF4 = 1.6E-4

Semi-Automatic Switchover

RSF5 = 1.0E-4 RSF6 = 8.7E-4 RSF7 = 2.1E-4

11. Affected Accident Sequences and Adjusted-Case Frequencies:

For Oconee 3:

Automatic Switchover

T_2MQFH	γ	(PWR-2)	=	3.4E-7/RY
	β	(PWR-4)	=	5.0E-9/RY
	ϵ	(PWR-6)	=	3.4E-7/RY
S_2FH	α	(PWR-1)	=	1.3E-9/RY
	β	(PWR-4)	=	9.2E-10/RY
	ϵ	(PWR-6)	=	1.0E-7/RY
S_3FH	γ	(PWR-2)	=	2.5E-7/RY
	β	(PWR-4)	=	3.7E-9/RY
	ϵ	(PWR-6)	=	2.5E-7/RY

Semi-Automatic Switchover

T_2MQFH	γ	(PWR-2)	=	3.4E-7/RY
	β	(PWR-4)	=	4.9E-9/RY
	ϵ	(PWR-6)	=	3.4E-7/RY
S_2FH	α	(PWR-1)	=	4.1E-9/RY
	β	(PWR-4)	=	3.0E-9/RY
	ϵ	(PWR-6)	=	3.3E-7/RY

$$\begin{array}{lll}
 S_3FH & \gamma & (PWR-2) = 2.8E-7/RY \\
 & \beta & (PWR-4) = 4.1E-9/RY \\
 & \epsilon & (PWR-6) = 2.8E-7/RY
 \end{array}$$

12. Affected Release Categories and Adjusted-Case Frequencies:

<u>Automatic Switchover</u>	<u>Semi-Automatic Switchover</u>
PWR-1 = 1.3E-9/RY	PWR-1 = 4.1E-9/RY
PWR-2 = 5.9E-7/RY	PWR-2 = 6.2E-7/RY
PWR-4 = 4.6E-9/RY	PWR-3 = 1.2E-8/RY
PWR-6 = 6.9E-7/RY	PWR-4 = 9.5E-7/RY

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

Automatic Switchover: $\bar{F}^* = 1.3E-6/RY$

Semi-Automatic Switchover: $\bar{F}^* = 1.6E-6/RY$

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

Automatic Switchover: $W^* = 3.0 \text{ person-rem/RY}$

Semi-Automatic Switchover: $W^* = 3.2 \text{ person-rem/RY}$

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

Automatic Switchover: $\Delta\bar{F} (PWR) = 1.8E-6/RY$

Semi-Automatic Switchover: $\Delta\bar{F} (PWR) = 1.5E-6/RY$

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

Automatic Switchover: $\Delta W (PWR) = 4.5 \text{ person-rem/RY}$

Semi-Automatic Switchover: $\Delta W (PWR) = 4.3 \text{ person-rem/RY}$

17. Total Public Risk Reduction, (ΔW)_{Total}:

	Best Estimate (person-rem)	Error Bounds (person-rem)	
		<u>Upper</u>	<u>Lower</u>
Automatic Switchover	1.2E+4	5.8E+5	0
Semi-Automatic Switchover	1.1E+4	5.8E+5	0

Assuming all PWRs are affected (expected not to be so).

ATTACHMENT 1

Analysis of the public risk reduction associated with failure of the ECCS switchover procedure from injection to recirculation (RSF) is based on NRC-developed estimates of recirculation system unavailability corresponding to manual, automated, and semi-automated switchover options (Thadani 1981). The estimates were taken from Table 4 of this reference. The values in Thadani (1981) were then updated to account for advances in human error probability analysis techniques. This was done using human error rate estimates given in NUREG/CR-4639 (Gertman et al., 1988). The human error probability given by Gertman for a control room operator failing to switch over the core spray system from injection to recirculation modes was $1.3\text{E-}3/\text{demand}$. This value was assumed to more accurately reflect the human error probability for a manual switchover procedure than that given in Thadani (1981). To update the remaining human error probabilities given by Thadani (1981), the ratio of the Thadani estimate to the updated probability for the manual switchover was maintained. The switchover failure probabilities are as follows:

<u>Recirculation System Unavailability*</u>			
<u>Initiator</u>	<u>Manual</u>	<u>Auto</u>	<u>Semi-auto</u>
Large LOCA (A)	$1.3\text{E-}3$	$1.0\text{E-}4$	$1.0\text{E-}4$
Small LOCA (S1)	$3.9\text{E-}4$	$1.6\text{E-}4$	$8.7\text{E-}4$
Small/small LOCA (S2)	$3.9\text{E-}4$	$1.6\text{E-}4$	$2.1\text{E-}4$

* When a range of values is given, the high end of the range is assumed for conservatism.

The unavailabilities in the above table include credit for manual recovery from the errors in the manual mode and human error propensity in recovering from spurious activation in the automatic mode. These values are used in this analysis as the probabilities of element RSF as follows:

Manual System

RSF1 = $1.3\text{E-}3$ RSF2 = $3.9\text{E-}4$

Automatic System

RSF3 = $1.0\text{E-}4$ RSF4 = $1.6\text{E-}4$

Semi-automatic System

RSF5 = $1.0\text{E-}4$ RSF6 = $8.7\text{E-}4$ RSF7 = $2.1\text{E-}4$

AFFECTED ACCIDENT SEQUENCES

For a PWR, the dominant accident sequences involve failure of the spray recirculation system (F) and failure of the emergency coolant recirculation system (H). The dominant accident sequences containing events F and H for Ocone 3 (Andrews et al., 1983) are T2MQFH, S2FH, and S3FH. The corresponding minimal cut sets containing a common-cause failure of the operator to open containment sump suction valves at the start of recirculation (WXCM) are T2•M•P₁•Q•WXCM, S2•WXCM, and S3•WXCM. In this analysis RSF(n) replaces WXCM in the designated cut sets to establish the modified base-case frequencies as follows:

Modified Base-Case

<u>Accid. Seq.</u>	<u>Cont. Fail. Mode</u>	<u>Mode Prob.</u>	<u>Rel. Cat.</u>	<u>Rel. Cat. Freq. (RY⁻¹)</u>
T2MQFH	γ	.5	2	1.3E-6
	β	.0073	4	1.9E-8
	ϵ	.5	6	1.3E-6
S2FH	α	.01	1	9.7E-10
	β	.0073	4	7.1E-8
	ϵ	.8	6	7.8E-8
S3FH	γ	.5	2	2.1E-7
	β	.0073	4	3.0E-9
	ϵ	.5	6	2.1E-7

The modified base-case frequencies of release category were then multiplied by public dose factor (Andrews et al. 1983, Appendix D).

<u>Base-Case Frequency (RY⁻¹)</u>	<u>Dose Factor (person-rem)</u>	<u>Risk (person-rem/RY)</u>
PWR-1 = 9.7E-10	5.4E+6	0.005
PWR-2 = 1.5E-6	4.8E+6	7.2
PWR-4 = 2.3E-8	2.7E+6	0.06
PWR-6 = <u>1.6E-6</u>	1.5E+5	<u>0.2</u>
3.1E-6		7.5

ADJUSTED-CASE AFFECTED PUBLIC RISK

Two adjusted-cases are examined for this issue: (1) fully automatic actuation; and (2) semi-automatic actuation. The adjusted-case core-melt frequencies were calculated by substituting the appropriate values for RSF(n) into the minimal cut sets containing event WXCM that were given in Andrews et al. (1983). This accounts for the potential risk reduction associated with more reliable actuation systems (i.e., systems that are less prone to human error) for the recirculation mode. However, it is known that automatic and semi-automatic actuation systems are more prone to spurious actuations than are manual systems.

Spurious actuation of the recirculation mode may result in switchover of ECCS and containment spray pump suction to a dry containment sump. This would result in pump damage and potentially adverse safety consequences if these pumps are rendered unavailable in a real demand situation. Therefore, the risk reduction for installing actuation systems that are less prone to human error are offset somewhat by an increase in risks due to spurious actuations of switchover to the recirculation mode.

A new accident sequence was postulated to model the deleterious effects of spurious actuations. This sequence consists of the initiating events ($T_2 \bullet M \bullet P_1 \bullet Q$, S_2 , and S_3), followed by successful actuation of the injection mode, spurious actuation of the switchover to recirculation (SA), and then failure to recover by manually opening the containment sump suction valves. It was assumed that the probability of successful injection was 1. The event involving failure to recover by manually opening containment sump suction valves is identical to event WXCM (probability = 0.003). The probability of spurious actuation was estimated as follows.

A search of Licensee Event Reports (LERs) was performed to develop the basis for the probability of spurious actuation of the switchover to recirculation. LERs involving the key words "spurious actuation" were reviewed for the years 1987 to 1990 and categorized as to the affected system. The important systems for this analysis are the emergency control room air recirculation system, ESF systems, emergency diesel generators, and the Reactor Core Isolation Cooling (RCIC) system. The total number of LERs in the years 1987 to 1990 involving spurious actuation of these systems are shown below:

LER Data on Spurious Actuations

Year	CRAC	ESF	EDG	RCIC	Other	Total	Scrams
1987	6	6	0	1	41	54	13
1988	14	3	1	2	31	51	12
1989	9	4	1	2	35	51	15
1990	2	2	0	3	13	20	10
	31	15	2	8	120	176	50
AVERAGE	7.75	3.75	0.5	2	30	44	12.5
No. of RY	440	440	440	440	440	440	440
Prob./yr	1.76E-02	8.52E-03	1.14E-03	4.55E-03	6.82E-02	1.00E-01	2.84E-02

The probability of spurious actuation for the fully-automatic system was assumed to be represented by the value for the ESF actuation systems, $8.5E-3/\text{RY}$. For the semi-automatic switchover, spurious actuations are somewhat less frequent. The value for spurious actuations of RCIC systems was used, $4.55E-3/\text{RY}$. These failure rate values were converted to failures per demand assuming that 2 real demands/RY are required. This results in failure probabilities for the automatic switchover case of $4.3E-3/\text{demand}$ and for the semi-automatic switchover case of $2.3E-3/\text{demand}$. The core-melt frequency increases, which were added to the adjusted-case accident sequence frequencies, for the two cases are as follows:

Automatic Switchover

$$\begin{aligned}
 T_2 \bullet M \bullet P_1 \bullet Q \bullet SA \bullet WXCM &= (1.5E-3/RY)(4.3E-3)(0.003) = 1.9E-8 \\
 S_2 \bullet SA \bullet WXCM &= (4E-4/RY)(4.3E-3)(0.003) = 5.2E-9 \\
 S_3 \bullet SA \bullet WXCM &= (0.0013/RY)(4.3E-3)(0.003) = 1.7E-8
 \end{aligned}$$

Semi-Automatic Switchover

$$\begin{aligned}
 T_2 \bullet M \bullet P_1 \bullet Q \bullet SA \bullet WXCM &= (1.5E-3/RY)(2.3E-3)(0.003) = 1.0E-8 \\
 S_2 \bullet SA \bullet WXCM &= (4E-4/RY)(2.3E-3)(0.003) = 2.8E-9 \\
 S_3 \bullet SA \bullet WXCM &= (0.0013/RY)(2.3E-3)(0.003) = 9.0E-9
 \end{aligned}$$

The adjusted-case accident sequences and their frequencies are as follows.

Automatic Switchover

<u>Accid. Seq.</u>	<u>Cont. Fail. Mode</u>	<u>Mode Prob.</u>	<u>Rel. Cat.</u>	<u>Rel. Cat. Freq. (RY⁻¹)</u>
T2MQFH	γ	.5	2	3.4E-7
	β	.0073	4	5.0E-9
	ϵ	.5	6	3.4E-7
<u>Accid. Seq.</u>	<u>Cont. Fail. Mode</u>	<u>Mode Prob.</u>	<u>Rel. Cat.</u>	<u>Rel. Cat. Freq. (RY⁻¹)</u>
S2FH	α	.01	1	1.3E-9
	β	.0073	4	9.2E-10
	ϵ	.8	6	1.0E-7
S3FH	γ	.5	2	2.5E-7
	β	.0073	4	3.7E-9
	ϵ	.5	6	2.5E-7

Semi-Automatic Switchover

<u>Accid. Seq.</u>	<u>Cont. Fail. Mode</u>	<u>Mode Prob.</u>	<u>Rel. Cat.</u>	<u>Rel. Cat. Freq. (RY⁻¹)</u>
T2MQFH	γ	.5	2	3.4E-7
	β	.0073	4	4.9E-9
	ϵ	.5	6	3.4E-7
S2FH	α	.01	1	4.1E-9
	β	.0073	4	3.0E-9
	ϵ	.8	6	3.3E-7
S3FH	γ	.5	2	2.8E-7
	β	.0073	4	4.1E-9
	ϵ	.5	6	2.8E-7

The final step is to multiply the frequencies of each release category by the dose factors given by Andrews et al. (1983) and then sum the resulting values to obtain the adjusted-case risk estimates. These calculations are shown below:

Automatic Switchover

<u>Base-Case Frequency (RY⁻¹)</u>	<u>Dose Factor (person-rem)</u>	<u>Risk (person-rem/RY)</u>
PWR-1 = 1.3E-9	5.4E+6	0.007
PWR-2 = 5.9E-7	4.8E+6	2.84
PWR-4 = 4.6E-9	2.7E+6	0.01
PWR-6 = <u>6.9E-7</u>	1.5E+5	<u>0.1</u>
1.3E-6		3.0

Semi-Automatic Switchover

<u>Base-Case Frequency (RY⁻¹)</u>	<u>Dose Factor (person-rem)</u>	<u>Risk (person-rem/RY)</u>
PWR-1 = 4.1E-9	5.4E+6	0.02
PWR-2 = 6.2E-7	4.8E+6	3.00
PWR-4 = 1.2E-8	2.7E+6	0.03
PWR-6 = <u>9.5E-7</u>	1.5E+5	<u>0.1</u>
1.3E-6		3.2

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Automatic Emergency Core Cooling Switch to Recirculation (24)

2. Affected Plants (N):

All PWRs are assumed to be affected.

	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	47	43	90

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

	<u>$\bar{T}$</u>		
	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	27.7	30	28.8

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

Fully-Automatic Actuation

$$PWR = (19,860 \text{ person-rem})(1.8E-6/R\bar{Y}) = 3.6E-2 \text{ person-rem/R}\bar{Y}$$

Semi-Automatic Actuation

$$PWR = (19,860 \text{ person-rem})(1.5E-6/R\bar{Y}) = 3.0E-2 \text{ person-rem/R}\bar{Y}$$

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

	<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
		<u>Upper</u>	<u>Lower</u>
<u>Fully Automatic</u>	9.3E+1	9.6E+2	0
<u>Semi-Automatic</u>	7.7E+1	9.6E+2	0

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

The SIR implementation involves hardware fixes in the control room and procedure changes. As discussed in Section 2.0 (See Table 3), implementation is estimated to require approximately 5 man-days/plant of utility labor in radiation zones to install, test, and calibrate new controllers/logic modules for the automatic and semi-automatic actuation of recirculation.

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

The dose rate is assumed to be 2.5 mrem/hr, which is representative of the dose rate outside containment during reactor shutdowns (Andrews et al., 1983). The following estimate is assumed to be applicable to both potential SIRs (i.e., fully-automatic and semi-automatic actuation).

$$D = (5 \text{ person-days/plant})(8 \text{ hr/d})(2.5 \text{ mrem/hr}) = 0.5 \text{ person-rem/plant}$$

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

$$ND = (0.5 \text{ person-rem/plant})(90 \text{ plants}) = 45 \text{ person-rem}$$

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

Additional utility labor in radiation zones for SIR operation and maintenance is anticipated to be minimal. However, because maintenance and testing procedures will be more complex for the automatic and semi-automatic actuation systems, an additional 1 person-day/yr of utility labor in radiation zones is included to accomplish this function.

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

$$D_o = (1 \text{ person-day/Ry})(8 \text{ hr/d})(2.5 \text{ mrem/hr}) = 0.02 \text{ person-rem/Ry}$$

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

Operation/maintenance doses are incurred at all PWRs.

$$N\bar{T}D_o = (90 \text{ plants})(28.8 \text{ yr})(0.02 \text{ man-rem/Ry}) = 5.2E+1 \text{ man-rem}$$

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
9.7E+1	2.9E+2	3.2E+1

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Automatic Emergency Core Cooling Switch to Recirculation (24)

2. Affected Plants (N):

All PWRs are assumed to be affected.

	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	47	43	90

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

	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	27.7	30	28.8

Industry Costs (Steps 4 through 12):

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

Automatic Switchover

$$\Delta\bar{F}A \text{ (PWR)} = (\$1.65E+9)(1.8E-6/\text{RY}) = \$ 2970/\text{RY}$$

Semi-Automatic Switchover

$$\Delta\bar{F}A \text{ (PWR)} = (\$1.65E+9)(1.3E-6/\text{RY}) = \$ 2475/\text{RY}$$

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

	<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
<u>Automatic Switchover</u>	\$7.7E+6	\$8.0E+7	0
<u>Semi-Automatic Switchover</u>	\$6.4E+6	\$8.0E+7	0

6. Per-Plant Industry Resources for SIR Implementation:

SIR implementation costs are itemized as follows:

Design/QA	2 man-months/plant *
Procure new controller/logic module	\$ 5,000 *
Install/calibrate/test	1 man-wk *
Safety Analysis	2 man-months *
Technical Specification Change	16 man-wks **
Revise Operating Procedures	\$ 3,900 **
Revise Maintenance Procedures	\$ 3,900 **
<u>Training</u>	2 man-months *

* Estimated for this analysis.

** Taken from NUREG/CR-4627 (Sciacca 1989).

The cost estimates for implementation of the automatic switchover system is assumed to be approximately the same as the costs for implementation of the semi-automatic system.

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

$$I \text{ (PWR)} = (41 \text{ man-wk/plant})(\$2270/\text{man-wk}) + \$13,800 = \$ 1.1\text{E}+5/\text{plant}$$

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

$$NI = (90 \text{ PWRs})(\$1.1\text{E}+5/\text{plant}) = \$ 9.9\text{E}+6$$

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

Additional labor requirements for operation and maintenance of the proposed SIR are estimated to be minimal. An additional 1 man-wk/Ry is estimated to be required.

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

$$I_o = (\$2270/\text{man-wk})(1 \text{ man-wk/Ry}) = \$ 2270/\text{Ry}$$

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

$$\begin{aligned} N\bar{I}_o &= (90 \text{ plants})(28.8 \text{ yr})(\$2270/\text{Ry}) \\ &= \$ 5.9\text{E}+6 \end{aligned}$$

12. Total Industry Cost (S_I):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$ 1.6E+7	\$ 2.2E+7	\$ 1.0E+7

NRC Cost (Steps 13 through 21)

13-20. Steps Related to NRC Costs for SIR Development, Support of SIR Implementation, and Review of SIR Operation and Maintenance

13. NRC Resources for SIR Development:

NRC resources for SIR development are estimated to include a 1 man-yr contractor study to research potential design changes and prepare a regulatory analysis for the NRC. Resources will also be required for a NRC Project Manager at about 10% of the contractor labor estimate.

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

$$\begin{aligned} C_D &= (1.0 \text{ man-yr})(\$1.0\text{E}+5/\text{man-yr}) + (0.1 \text{ man-yr})(\$1.0\text{E}+5/\text{man-yr}) \\ &= \$ 1.3\text{E}+5 \end{aligned}$$

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

It is estimated that approximately 8 man-wk/plant of NRC labor are required to review and evaluate the licensee's design, safety analyses, and QA documentation for the hardware modifications, technical specification changes, and procedure changes.

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

$$C = (8 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$ 1.8\text{E}+4/\text{plant}$$

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

$$\begin{aligned} NC &= (90 \text{ plants})(\$ 1.8\text{E}+4/\text{plant}) \\ &= \$ 1.6\text{E}+6 \end{aligned}$$

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

NRC labor for this activity is anticipated to be minimal. It is estimated that 0.5 man-wk/Ry will be sufficient for NRC review of each technical specification and procedure change and the component data developed by the licensees.

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

$$C_o = (0.5 \text{ man-wk/Ry})(\$2270/\text{man-wk}) = \$ 1.1\text{E}+3/\text{Ry}$$

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

$$\begin{aligned} \bar{N}TC_o &= (90 \text{ plants})(28.8 \text{ yr})(\$1.1\text{E}+3/\text{Ry}) \\ &= \$ 2.8\text{E}+6 \end{aligned}$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$ 4.5E+6	\$ 6.1E+6	\$ 2.9E+6

REFERENCES (For Issue 24)

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.

Sciacca, F. 1989. *Generic Cost Estimates*. NUREG/CR-4627, Rev. 2. Science and Engineering Associates, Inc., Albuquerque, New Mexico.

Gertman, D. I. et al. 1988. *Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR)*. NUREG/CR-4639, EGG-2458. Idaho National Engineering Laboratory, Idaho Falls, Idaho.

Thadani, A. 1981. Memorandum for T. Novak and J. Olshinski, "Comparative Risk Assessment of the ECCS Functional Switchover Options." April 1, 1981. U. S. Nuclear Regulatory Commission, Washington D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 43, Reliability of Air Systems (Revised)

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: The principal concern of this issue is the potential safety implications associated with air system problems at U.S. light water reactors (LWRs). The risk of severe accidents at U.S. plants may have been underestimated in the past because it has previously been assumed that air system failures cannot disable safety systems. Recent operating experience indicates dozens of occurrences in which, contrary to previous assumptions, a safety-related system has failed because of air system failure or degradation.

AFFECTED PLANTS:

BWR: Operating = 34	Planned = 10
PWR: Operating = 70	Planned = 20

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION =	8.6E+03
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OCCUPATIONAL DOSES:

SIR Implementation =	10.4
SIR Operation/Maintenance =	593.4
Total of Above =	603.8
Accident Avoidance =	52.7

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	5.5
SIR Operation/Maintenance =	18.0
Total of Above =	23.5
Accident Avoidance =	4.4

NRC COSTS:

SIR Development =	0.3
SIR Implementation =	2.6
SIR Operation/Maintenance Review =	3.7
Total of Above =	6.6

RELIABILITY OF AIR SYSTEMS

ISSUE 43 (revised)¹

1.0 SAFETY ISSUE DESCRIPTION

The principal concern of this issue is the potential safety implications associated with air system problems at U.S. light water reactors (LWRs). The risk of severe accidents at U.S. plants may have been underestimated in the past because it has previously been assumed that air system failures cannot disable safety systems.

Many U.S. light water reactors (LWRs) rely upon air systems to actuate or control safety-related equipment during normal operation. However, at most LWRs the air systems themselves are not classified as safety systems. As a result, plant accident analyses typically assume that safety-related equipment dependent upon air systems will either "fail safe" upon loss of air or perform its intended function with the assistance of safety grade backup accumulators. A recent case study by the NRC's Office for Analysis & Evaluation of Operational Data (AEOD) has documented more than two dozen incidents in which, contrary to licensing assumptions, a safety-related system failed because of air system degradation or failure (Ornstein 1987).

The root cause of most of these failures is traceable to design and/or management deficiencies. The design and operating problems found appear to reflect a lack of sufficient regulatory requirements and review, and the view by many applicants and licensees that air systems are not highly important to assuring plant safety.

At many plants, the priority given for repairing air system components is low, and as a result, component or system redundancy is frequently lost. For example, the licensee for H. B. Robinson 2 has concluded that significant gains in air system availability (and overall plant availability) could be achieved by assigning a high priority to air system maintenance and repair operations (EDS Nuclear Inc. 1982). Many plants operate with high moisture content in the air system, and routinely drain out water from the air lines. There have been many cases where malfunctioning air dryers have been bypassed for long periods of time. In addition, plant personnel are generally unaware of the potential for simultaneous or common mode failures of redundant safety-related equipment which can result from degraded air systems.

As a result of such problems and due to a concern for the continued reliability of air systems support to safety-related systems, NRC has issued Information Notice No. 87-28: Air Systems Problems at U.S. Light Water Reactors

¹

The scope of this generic issue has been revised to reflect a broader interest in the performance of air systems at U.S. nuclear power plants. The principal concern of the original generic issue 43, Contamination of Instrument Air Lines, was limited to consideration of the safety significance of material contamination of instrument air lines. The principal concern of the revised Issue 43 includes that of the original Issue 43 but is expanded to encompass any factors that can result in a reduction to the overall reliability of air systems. Accordingly, the revised issue title is "Reliability of Air Systems".

(NRC 1987). The purpose of the Information Notice is to alert all nuclear power plant facilities holding an operating license or a construction permit to the potentially significant problems pertaining to air systems at light water reactors.

PROPOSED RESOLUTION

For this analysis, this proposed safety issue resolution (SIR) is assumed to involve the following actions:

1. Licensees should evaluate their air system(s) to ensure that the air quality is consistent with the equipment specifications and that it is periodically monitored and tested.
2. Anticipated transients and system recovery procedures and related training for loss of air systems should be reviewed for adequacy and revised as necessary.
3. Plant staff should be trained regarding the importance of air systems.
4. The adequacy of safety-grade backup air accumulators for safety related equipment should be verified.
5. All operating plants should be required to perform gradual loss of instrument air system pressure tests.

AFFECTED PLANTS

The resolution of this safety issue is assumed to affect all operating and planned LWRs.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with the issue resolution 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. This is a consequence of the limited quantitative treatment of air systems in conventional PRAs. The Oconee-3 PRA (Sugnet et al., 1984) risk equations for air systems have been assumed to be representative of all LWRs.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Reliability of Air Systems, 43 (revised)

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

All 134 LWRs are assumed to be affected ($\bar{T}$ = 24.6 yr)

3. Plants Selected for Analysis:

A hypothetical LWR is assumed to be representative of all LWRs. (The safety analysis is conducted for Oconee-3, and the results are applied to the hypothetical LWR, as discussed in Attachment 1.)

4. Parameters affected by SIR:

T6, Loss of Instrument Air

5. Base-Case Values for Affected Parameters:

T6 base-case frequency = 0.17/py

6. Affected Accident Sequences and Base-Case Frequencies:

T6*REIA2/6*EFTDST6H*REEF122/6*UTHPIH	1.4E-06/py
T6*REIA2/6*EFM17*REEF122/6*UTHPIH	5.2E-07/py
T6*EFUSTF*UTHPIH	6.8E-07/py
T5SUBF*RESUBAIR2*REEF122/6*EFTDST5H*UTHPIH	<u>6.0E-07/py</u>
	3.2E-06/py

7. Affected Release Categories and Base-Case Frequencies:

Average dose factor for an LWR is estimated directly (see Attachment 1).

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

$$\bar{F} = 3.2\text{E-}06/\text{py}$$

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

$$W = 10.6 \text{ man-rem/py}$$

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

$$T6 = 0.12/\text{py}$$

11. Affected Accident Sequences and Adjusted-Case Frequencies:

T6*REIA2/6*EFTDST6H*REEF122/6*UTHPIH	9.9E-07/py
T6*REIA2/6*EFM17*REEF122/6*UTHPIH	3.7E-07/py
T6*EFUSTF*UTHPIH	4.8E-07/py
T5SUBF*RESUBAIR2*REEF122/6*EFTDST5H*UTHPIH	<u>6.0E-07/py</u>
	2.4E-06/py

12. Affected Release Categories and Adjusted-Case Frequencies:

Release categories assumed not affected by SIR (see Attachment 1).

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

$$\bar{F}^* = 2.4\text{E-}06$$

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

$$W^* = 7.9/\text{py}$$

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

$$\Delta\bar{F} = 8.0\text{E-}07/\text{py}$$

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

$$\Delta W = 2.6 \text{ man-rem/py}$$

17. Total Public Risk Reduction, (ΔW)Total:

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
8.6E+03	2.5E+04	0

ATTACHMENT 1 (to Table 1)

Resolution of Issue 43 (revised) uses the Oconee-3 Probabilistic Risk Assessment (Sugnet, et al., 1984) as a basis for evaluation due to its detailed assessment of the potential for safety system failures due to air system malfunctions involvement of air systems in accident sequences). Estimates of the affected public risk and the affected core-melt frequency reductions presented in this analysis are based on the results of that study.

It is assumed that the reductions in affected public risk and core-melt frequency can be estimated as reductions in air systems contribution to the total Oconee-3 risk. The dominant sequence type involves the following scenario: A loss of instrument air (T6) occurs, as an initiating event, as a result of a loss of offsite power, or as a result of system faults after a reactor trip. Main feedwater is not available because of the loss of instrument air, and the special operation of the emergency feedwater system after a loss of instrument air also fails (only the steam-driven pump is available, and it requires special actions). Feedwater is not recovered, and HPI cooling fails to be initiated.

Another sequence type involves a loss of offsite power through substation failure, T5SUBF, as an initiating event. Instrument air is not recovered within 2 hours after the loss of offsite power, feedwater is not recovered, and HPI cooling fails to be initiated. Since this sequence type involves a failure to recover an air system, it is included here for completeness. However, it is conservatively assumed that the safety issue resolution (SIR) will have a negligible affect on the frequency of this cut set. Therefore, although it is included in consideration, the adjusted-case frequency for the T5SUBF-initiated cut set is equivalent to its base-case frequency.

The dominant cut sets involving air systems malfunctions, along with their original, base-case frequencies are shown below:

<u>Cut Set</u>	<u>Base-Case Event frequency /probability</u>	<u>Base-Case Cut-set frequency</u>
T6	0.17/yr	1.4E-06/py
* REIA2/6	0.055	
* EFTDST6H	0.15	
* REEF122/6	0.1	
* UTHPIH	0.01	
T6	0.17/yr	5.2E-07/py
* REIA2/6	0.055	
* EFM17	0.056	
* REEF122/6	0.1	
* UTHPIH	0.01	
T6	0.17/yr	6.8E-07/py
* EFUSTF	4.0E-04	
* UTHPIH	1.0E-02	

<u>Cut Set</u>	<u>Base-Case Event frequency /probability</u>	<u>Base-Case Cut-set frequency</u>
T5SUBF	0.13/yr	6.0E-07/py
* RESUBAIR2	0.031	
* REEF122/6	0.1	
* EFTDST5H	0.15	
* UTHPIH	0.01	
		<hr/>
		3.2E-06/py

The events in these cut sets are defined as follows:

<u>Event</u>	<u>Description</u>
T6	Loss of instrument air
REIA2/6	Failure of operator to recover instrument air in 2 to 6 hours after the initiating event
EFTDST6H	Operator's failure to provide suction to the steam-driven emergency-feedwater pump after the upper surge tank is depleted, given loss of instrument air; the operator is required to perform remote manual actions, including a position change on a locked valve
REEF122/6	Failure to obtain feedwater from the SSF after the failure of emergency feedwater 2 to 6 hours after the initiating event
UTHPIH	Operator's failure to initiate HPI cooling
EFM17	Failure of the steam-driven emergency-feedwater pump through local hardware faults and human errors
EFUSTF	Failure of the emergency-feedwater pump because of insufficient inventory in the upper surge tank at the start of the sequence
T5SUBF	Loss of offsite power through substation failure
RESUBAIR2	Failure to recover instrument air within 2 hours after the loss of offsite power
EFTDST5H	Analogous to EFTDST6H for the loss of offsite power

Therefore, the frequency of core damage (3.2E-06/py) due to air system problems at Oconee-3 is used as the base-case, affected core-melt frequency for the hypothetical LWR.

In order to evaluate the adjusted case core-melt frequency, it is assumed that T6 initiating event frequency will be reduced approximately 30% as a result

of SIR implementation. This assumption is based on an evaluation of the potential effects of the proposed resolution on each of several factors that contribute to the overall frequency of loss of instrument air, T6. It is believed that this assumption is conservative (some plants may realize greater improvements). One conservatism of this estimate is that no credit was given for potential improvements in operator recovery actions, which could in fact reduce the consequences of air systems-related problems. An upper bound that assumes a 90% improvement in instrument air reliability is evaluated to show the potential effects on plants which may have greater improvement in air systems performance.

The adjusted-case, affected core-melt frequency for the hypothetical LWR is calculated by replacing the base-case T6 event frequency of 0.17/py by the adjusted-case T6 event frequency of 0.12/py. This results in an adjusted-case, affected core-melt frequency of 2.4E-06/py.

To obtain the base and adjusted-case, affected public risks, the overall risk is written as follows:

$$\bar{W}_o = F_o R_o$$

where W_o = overall risk

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:

$$(R_o)_{LWR} = \frac{(\bar{T}W_o)_{PWR} + (\bar{T}W_o)_{BWR}}{(\bar{T}F_o)_{PWR} + (\bar{T}F_o)_{BWR}}$$

Based on Appendices A-D of NUREG/CR-2800 (Andrews, et al., 1983).

$$(R_o)_{LWR} = 3.3E+06 \text{ man-rem}$$

where N = 90 (PWR) and 44 (BWR)

$\bar{T}$ = 24.2 yr (PWR) and 22.8 yr (BWR)

W_o = 207 man-rem/py (PWR) and 250 man-rem/py (BWR)

F_o = 8.2E-05/py (PWR) and 3.7E-05/py (BWR)

Thus, for this issue,

$$\begin{aligned} W &= (3.2E-06/py)(3.3E+06 \text{ man-rem}) \\ &= 10.6 \text{ man-rem/py} \end{aligned}$$

$$\begin{aligned} W^* &= (2.4E-06/py)(3.3E+06 \text{ man-rem}) \\ &= 7.9 \text{ man-rem/py} \end{aligned}$$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Reliability of Air Systems, Generic Issue 43 (revised)

2. Affected Plants (N):

All 134 plants (104 operating and 30 planned) are assumed to be affected. There are no significant differences between plant types for occupational doses of SIR implementation; thus, there is no need to distinguish between PWRs and BWRs to estimate occupational doses.

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

Operating $\bar{T}$ = 23.1 yr

Planned $\bar{T}$ = 30 yr

$\bar{T}$ = 24.6 yr for all plants

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

$$\Delta(\bar{F}D_r) = (19,860 \text{ man-rem})(8.0E-07/\text{py}) = 1.6E-02/\text{py}$$

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

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
52.7	153.2	0

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

For the 104 operating plants, it is assumed that each will require additional labor time in order to evaluate their air system(s) to ensure that the air quality is consistent with the equipment specifications. This effort is expected to require 40 man-hours per plant in a radiation zone. For planned plants, no additional work in radiation zones is foreseen since evaluation of their systems can be done in a radiation-free environment (i.e., prior to initial fuel loading).

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

It is assumed that the areas of the air system which are of principle concern are located outside reactor containment. The dose rate is assumed to be 2.5 mR/hr.

$$D = (0.0025 \text{ R/hr})(40 \text{ man-hr/plant}) = 0.1 \text{ man-rem/plant}$$

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

$$ND = (104 \text{ plants})(0.1 \text{ man-rem/plant}) = 10.4 \text{ man-rem}$$

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

It is assumed that for all LWRs, operation requires 8 man-hr/plant, each month, to periodically monitor and test the air system(s) to ensure appropriate air quality and system operation. This results in a labor requirement of 96 man-hr/py. Using a 75% utilization factor, this labor requirement becomes 72 man-hr/py in a radiation zone.

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

Again, a dose rate of 2.5 mR/hr is assumed.

$$D_o = (0.0025 \text{ R/hr})(72 \text{ man-hr/py}) = 0.18 \text{ man-rem/py}$$

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

$$\bar{N}D_o = (134 \text{ plants})(24.6 \text{ yr})(0.18 \text{ man-rem/py}) = 593.4 \text{ man-rem}$$

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
604	1812	201

3.0 SAFETY ISSUE COSTS

The industry and NRC costs associated with resolution of this safety issue 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:

Reliability of Air Systems, Generic Issue 43 (revised)

2. Affected Plants (N):

All 134 plants (104 operating and 30 planned) are assumed to be affected.

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

Operating $\bar{T}$ = 23.1 yr

Planned $\bar{T}$ = 30 yr

$\bar{T}$ = 24.6 yr for all plants

Industry Costs (Steps 4 through 12):

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

$$\Delta(\bar{F}A) = (8.0E-07/\text{py})(\$1.65E+09) = \$1320/\text{py}$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$4.35E+06	\$1.27E+07	0

6. Per-Plant Industry Resources for SIR Implementation:

For the 104 operating plants, the assumed resource requirements² are as follows:

Labor = 2.4 man-weeks/plant to evaluate air system(s)
4.3 man-weeks/plant to review recovery procedures
0.6 man-weeks/plant on average to revise recovery procedures
2.8 man-weeks/plant for additional staff training
3.3 man-weeks/plant to verify adequacy of back-up accumulators
7.9 man-weeks/plant for planning and executing tests
21.3 man-weeks/plant for SIR Implementation

² Based on solicitation of expert opinion.

Equipment = additional testing equipment, moisture indicators,
contamination sensors

For planned plants, no additional labor, down-time or equipment is needed because it is assumed that air systems will be properly evaluated at the pre-operational testing phase.

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

$$\begin{aligned}\text{Labor} &= (21.3 \text{ man-wk/plant})(\$2270/\text{man-wk}) \\ &= \$4.8\text{E}+04/\text{plant}\end{aligned}$$

$$\text{Equipment} = \$5.0\text{E}+03/\text{plant}$$

$$\text{Total} = \$5.3\text{E}+04/\text{plant}$$

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

$$\text{NI} = (104 \text{ plants})(\$5.3\text{E}+04/\text{plant}) = \$5.5\text{E}+06$$

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

$$\text{Labor} = 8 \text{ man-hr/mo for periodic monitoring and testing}$$

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

$$\begin{aligned}\text{I}_o &= (96 \text{ man-hour/py})(\$2270/\text{man-wk})(1 \text{ man-wk}/40 \text{ man-hr}) \\ &= \$5.45\text{E}+03/\text{py}\end{aligned}$$

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

$$\text{NTI}_o = (134 \text{ plants})(\$5.45\text{E}+03/\text{py})(24.6 \text{ yr}) = \$18.0\text{E}+06$$

12. Total Industry Cost (S_I):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
18.0E+06	27.4E+06	8.6E+06

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development

Estimates included directly in next step.

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

Continuing Trend Analysis	=	\$5.0E+04
NUREG	=	2.0E+05
IE Bulletin	=	1.7E+04
C _o	=	\$2.7E+05

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

Assumed to consist of two parts:

1. Review and approval of license's evaluations = 2 man-months
2. Onsite Inspections = 1 man-month

Total = 3 man-months
or 0.25 man-yr/plant

This is assumed to apply only to the 104 operating plants since these activities will presumably be incorporated initially into the normal review and inspection accorded plants prior to operation.

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

$$C = (0.25 \text{ man-yr/plant})(\$1.0\text{E}+05/\text{man-yr}) = \$2.5\text{E}+04$$

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

$$NC = (104 \text{ plants})(\$2.5\text{E}+04) = \$2.6\text{E}+06$$

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

0.5 man-week/py

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

$$C_o = (0.5 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$1135/\text{py}$$

20. Total NRC Cost for Review of SIR Operation and Maintenance (NTC_o):

$$NTC_o = (134 \text{ plants})(24.6 \text{ yr})(\$1135/\text{py}) = \$3.7\text{E}+06$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$6.6E+06	\$8.9E+06	4.3E+06

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 55, Failure of Class IE Safety-Related Switchgear Circuit Breakers to Close on Demand

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: Licensee Event Reports show a low frequency of class IE circuit breakers failing to close during routine inspection/test procedures. If such a failure would occur during an emergency, the diesel generators may not be connected to the safety-related electrical load. Issue resolution involves increasing the frequency of inspection and maintenance procedures on Class IE breakers in the diesel generator supply system.

<u>AFFECTED PLANTS:</u>	BWR: Operating = 24	Planned = 20
	PWR: Operating = 47	Planned = 43

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION =	2.0
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OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	750
Total of Above =	750
Accident-Avoidance =	0.022

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	1.2
SIR Operation/Maintenance =	22
Total of Above =	23
Accident-Avoidance =	0.0018

NRC COSTS:

SIR Development =	0.12
SIR Implementation Support =	0.16
SIR Operation/Maintenance Review =	0.22
Total of Above =	0.50

FAILURE OF CLASS IE SAFETY-RELATED SWITCHGEAR
CIRCUIT BREAKERS TO CLOSE ON DEMAND
ISSUE 55

1.0 SAFETY ISSUE DESCRIPTION

The NRC initiated safety issue 55 in September 1982 following a peer review of the Preliminary Report "Failures of Class IE Safety-Related Switchgear Circuit Breakers to Close on Demand" (Chiramal 1982). This report documented results of an ongoing study by the Office for Analysis and Evaluation of Operations Data (AEOD). A review of Licensee Event Reports (LERs) showed 94 failures of the subject breakers to close on demand during surveillance testing or inspections at 42 nuclear power plants over a 5 1/4-year period. Approximately 25% of the occurrences involved a diesel generator output breaker.

The Class IE (Emergency Power System) switchgear provide power to associated safety-related electrical loads that mitigate consequences of transients and accident conditions and/or bring the plant to a safe shutdown. Failure of the subject breakers to close, in the event that offsite power is lost, prevents connection of the safety-related loads to the diesel generators. The principal causes of these failures are defects in the closing circuit. Breakers may be closed manually by a local ratchet-type spring charging mechanism and release bar. Color lights on the main control panel indicate the position of the breakers. Typically, green and red lights are used to indicate that the main breaker contacts are open or closed, respectively. By observing the breaker position lights, it is assumed that an operator could authorize manual bypass of a failed breaker closing circuit within minutes of detecting its abnormal operation.

AEOD Preliminary Recommendations

The following three major recommendations were made in the AEOD Preliminary Report to mitigate the consequences of this issue:

1. install closing circuit status monitors in the main control room
2. improve maintenance procedures tending to preclude electrical problems in the breaker closing circuit
3. establish regular local surveillance to monitor the readiness status of each spring charging motor

The Office of Nuclear Reactor Regulation (NRR) reviewed the AEOD report (Denton 1982) and concurred with recommendations 2 and 3 but not with recommendation 1. The NRR reasoned that, because most of the failures were discovered by surveillance and testing, the installation of monitoring systems alone would not provide assurance that the breakers would be more reliable during an event that required their operation. Additional monitors would add to the number of indicators on existing control panels and tend to increase the chance of human response errors and inadvertent activation.

This SIR, therefore, assumes implementation only of recommendations 2 and 3.

2.0 SAFETY ISSUE RISK AND DOSE

The analyses of the public risk reduction and occupational dose associated with SIR are presented in this section. Results are summarized in Tables 1 and 2, respectively. Attachments 1 and 2 to the Public Risk Reduction and Occupational Dose Work Sheets, respectively, are provided to detail some of the related calculations.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Failure of Class IE Safety-Related Switchgear Circuit Breakers to Close on Demand (55)

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

All plants are affected

	<u>N</u>	<u>$\bar{T}$</u>
PWRs	90	28.8 yr
BWRs	44	27.4 yr

3. Plants Selected for Analysis:

Representative PWR: Arkansas Number One-Unit 1 (ANO-1)
Representative BWR: Grand Gulf 1

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

4. Parameters Affected by SIR:

CBF1 and CBF2

The Risk equations assessed in the ANO-1 Study (Kolb 1982) are redefined to include CBF1 and CBF2 (see Attachment 1).

5. Base-Case Values for Affected Parameters:

CBF1 = $8.1\text{E-}4$
CBF2 = $8.1\text{E-}4$

These base-case values assume an inspection frequency of 1 month for circuit breaker closing circuits.

6. Affected Accident Sequences and Base-Case Frequencies:

Based on the ANO-1 analysis (see Attachment 1):

T(LOP)LD1 YC	α (PWR-1) = 3.5E-14/py
	δ (PWR-2) = 7.0E-11/py
	β (PWR-4) = 2.5E-12/py
	ϵ (PWR-6) = 2.8E-10/py

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 3.5E-14/py
PWR-2 = 7.0E-11/py
PWR-4 = 2.5E-12/py
PWR-6 = 2.8E-10/py

8. Base-Case, Affected Core-Melt Frequency (F):

ANO-1 = 3.5E-10/py
Grand Gulf 1³ = 2.6E-10/py

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

ANO-1 = 3.9E-4 man-rem/py
Grand Gulf 1 = 9.8E-4 man-rem/py

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

CBF1 = 2.3E-4
CBF2 = 2.3E-4

This corresponds to increasing the inspection frequency to once per week (see Attachment 1).

11. Affected Accident Sequences and Adjusted-Case Frequencies:

T(LOP)LD1 YC	α = 2.8E-15/py
	δ = 5.7E-12/py
	β = 2.0E-13/py
	ϵ = 2.3E-11/py

12. Affected Release Categories and Adjusted Case Frequencies:

PWR-1 = 2.8E-15/py
PWR-2 = 5.7E-12/py
PWR-4 = 2.0E-13/py
PWR-6 = 2.3E-11/py

13. Adjusted-Case, Affected Core-Melt Frequency (F*):

ANO-1 = 2.8E-11/py

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

ANO-1 = 3.1E-15 man-rem/py

³ Results for BWR scaled for PWR according to the method described in Attachment 1.

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

$$(\Delta\bar{F})_{PWR} = 3.2E-10/\text{py}$$

$$(\Delta\bar{F})_{BWR} = 2.4E-10/\text{py}$$

Results for the representative BWR were scaled according to the methodology described in Attachment 1.

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

$$(\Delta W)_{PWR} = 3.6E-14 \text{ man-rem/py}$$

$$(\Delta W)_{BWR} = 9.0E-04 \text{ man-rem/py}$$

17. Total Public Risk Reduction, (ΔW):

Best Estimate	Error Bounds (man-rem)	
<u>(man-rem)</u>	<u>Upper</u>	<u>Lower</u>
2.0	6.6	0

ATTACHMENT 1

Analysis of the public risk reduction associated with resolution of issue 55 requires redefinition of the base case to include failure of Class IE switchgear circuit breakers to close on demand. The affected switchgear circuit breakers are assumed to be those connected with the diesel generators. The analysis is presented below.

FAILURE EVENT FAULT TREE

A fault tree for failure to energize Class IE safety-related electrical loads is shown in Figure 1. This is caused by either diesel generator failure or failure of the circuit breaker in its open position. The circuit breaker is failed open as a result of a failure in the breaker closing circuit and failure of the operator to order that the local manual bypass be activated. The latter is dominated by human error events. Failure to close the circuit breaker normally results from either an incorrect operation response or failure of the operator to respond to breaker position indicator lights. An incorrect operator response may occur if the operator responds to the wrong indicator light or his order to manually operate the breaker is misunderstood by another workman and the wrong breaker is closed. Failure of the operator to respond could be caused by improper indication of the breaker position or the operator failing to respond to correct light indications. The table below lists the probabilities per demand for the basic events of this fault tree.

Basic Event Probabilities (per Demand)

<u>Event</u>	<u>Probability</u>	<u>Information Source</u>
(1) Operator responds to wrong light	$P(1) = 5E-3$	NUREG/CR-1278
(2) Workman throws wrong breaker	$P(2) = 5E-3$	NUREG/CR-1278
(3) Improper light indication	$P(3) = \text{neg}^{(a)}$	WASH-1400
(4) Operator fails to respond to lights	$P(4) = 2.5E-1$	$^{(b)}$ NUREG/CR-1278
<hr/>		
(a) Improper light indication requires two simultaneous demand-type failures of indicator lights, i.e., green-light failure due to run out and redlight indication due to spurious current. Probabilities for light failures are on the order of $1E-6$ and negligible compared to those for human error.		
(b) From data for failure to respond to one of M lights on panel; value used corresponds to $M > 40$.

FIGURE 1. Fault Tree for Failure of Class IE Electrical Loads from Diesel Generator x to be Energized When Offsite Power is Lost

Unavailability of Manual Bypass

Failure of the operator on demand to effect on demand a manual bypass of the failed closing circuit is equivalent to an unavailability given by

$$A_{BF} = P(1) + P(2) + P(3) + P(4) = 2.6E-1$$

Unavailability of Breaker to Close on Demand

The unavailability of the breaker closing circuit can be estimated from the incidents reported in the AEOD preliminary report:

Number of failure to close events	= 94
Number of reactors affected	= 42
Period considered	= 5.25 years

Analysis of these data shows that the number of incidents varied from 0 to 3 per reactor-year and 1 to 8 per 5.25-year period for individual reactors. As these data were derived from required LERS, it is assumed that no such failures of the Class IE circuit breakers were reported from any other operating reactor.

Assuming periodic inspection every W weeks and a 1-day repair time if a breaker closing circuit is found defective,

$$\text{Average down-time } (T) = 1/2W + 1/7 \text{ weeks.}$$

$$\text{Failure frequency } (\lambda) = N_F / (341)(52) \text{ per week}$$

where N_F = number of failures
341 = the sum of reactor-years in reporting period (Andrews et al. 1983)
52 = the number of weeks/year

Unavailability of breaker closing circuit (A_{BCC}) is given by

$$A_{BCC} = \lambda T$$

Unavailability of breaker to close on demand (A_{CBF}) is

$$\begin{aligned} A_{CBF} &= A_{BCC} \times A_{BF} \\ &= 0.26 A_{BCC} \end{aligned}$$

The unavailability of the breakers to close on demand is critical in the case that transfer of safety-related electrical loads to diesel generator power is required. The AEOD report noted that approximately 25% of the incidents reported involved a diesel generator output breaker. The failure frequency calculation therefore assumes $N_F = (94)(0.25) = 24$. Therefore,

$$\lambda = 24 / (341)(52) = 1.4E-3/\text{wk}$$

The following table summarizes A_{BCC} and A_{CBF} for three inspection frequencies:

Breaker Unavailability as a Function of Inspection Frequency

<u>Inspection Frequency (W)</u>	A_{BCC} <u>(ΔT)</u>	A_{CBF} <u>($0.26 A_{BCC}$)</u>
1/month	3.1E-3	8.1E-4
1/2 weeks	1.5E-3	4.0E-4
1/week	8.7E-4	2.3E-4

AFFECTED ACCIDENT SEQUENCE

For ANO-1, only one dominant accident sequence corresponds to loss of emergency power--T(LOP)LDIYC (Kolb 1982). Of its minimal cut sets (dominant), only the following involve diesel-generator-related failures:

$$\begin{aligned} &T(LOP) \cdot LF-AC-DG1 \cdot LF-AC-DG2 \cdot LF-EFS-E11 \cdot [0.36] \\ &T(LOP) \cdot LF-AC-DG1 \cdot LF-AC-DG2 \cdot LF-EFC-D1D2CM \cdot [0.05] \end{aligned}$$

where the numbers in brackets [] represent the probabilities of nonrecovery within the estimated one-hour duration prior to onset of core melting.

The terms related to diesel generator failure (LF-AC-DG1 and LF-AC-DG2) are redefined as follows to include the circuit breaker failure CBF_x (from Figure 1):

$$\begin{aligned} LF-AC-DG1 &= (LF-AC-DG1)_o + CBF1 \\ LF-AC-DG2 &= (LF-AC-DG2)_o + CBF2 \end{aligned}$$

where the terms with the subscripts "o" represent the original terms and the designators "1" and "2" on CBF correspond to diesel generators "1" and "2" respectively.

These term redefinitions result in the generation of two "new" minimal cut sets, representing the affected minimal cut sets for this SIR:

$$\begin{aligned} &T(LOP) \cdot CBF1 \cdot CBF2 \cdot LF-EFS-E11 \cdot [0.36] \\ &T(LOP) \cdot CBF1 \cdot CBF2 \cdot LF-EFC-D1D2CM \cdot [0.05] \end{aligned}$$

Assuming current inspection procedures require only monthly inspections of the circuit breakers, the base-case frequencies of these affected cut sets become $3.0E-10$ /py and $4.8E-11$ /py, respectively. Original values from the ANO-1 study are used for all terms except CBF1 and CBF2 (these are taken as $8.1E-4$ as assessed previously). Thus, the affected sequence's base-case frequency is $3.5E-10$ /py.

SCALING PWR RESULTS FOR BWRS

The reliability studies for ANO-1 and Grand Gulf 1 (Hatch 1981) give total core-melt frequencies (F_o) of $5.0E-5$ /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, Andrews et al. 1983), one obtains total public risks (W_o) of 100 man-rem/py and 250 man-rem/py, respectively, for ANO-1 and Grand Gulf. For the purposes of scaling the base-case, affected core-melt frequency (F) and public risk (W), and the reductions in the core-melt frequency (ΔF) and public risk (ΔW) from ANO-1 to Grand Gulf, the following are assumed:

$$F_{BWR}/F_{PWR} = (F_o)_{BWR}/(F_o)_{PWR}$$

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

$$W_{BWR}/W_{PWR} = (W_o)_{BWR}/(W_o)_{PWR}$$

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

Using the original values of F_o and W_o for ANO-1 and Grand Gulf, the scaling equations become:

$$F_{BWR}/(\Delta F)_{BWR} = 0.74 F_{PWR}/0.74 (\Delta F)_{PWR}$$

$$W_{BWR}/(\Delta W)_{BWR} = 2.5 W_{PWR}/2.5 (\Delta W)_{PWR}$$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:
 Failure of Class IE Safety-Related Switch Gears Circuit Breakers to Close on Demand (55)
2. Affected Plants (N):
 All plant are assumed to be affected
 N = 90 PWRs and 44 BWRs
3. Average Remaining Lives of Affected Plants ($\bar{T}$):

PWRs	$\bar{T} = 28.8 \text{ yr}$
BWRs	$\bar{T} = 27.4 \text{ yr}$
All plants	$T = 28.3 \text{ yr}$
4. Per-Plant Occupational Dose Reduction Due to Accident-Avoidance, (ΔFDR):

PWR = (19,860 man-rem)(3.2E-10/py) = 6.4E-6 man-rem/py
BWR = (19,860 man-rem)(2.4E-10/py) = 4.8E-6 man-rem/py
5. Total Occupational Dose Reduction Due to Accident-Avoidance (ΔU):

<u>Best Estimate</u> <u>(Man-rem)</u>	<u>Error Bounds (man-rem)</u> <u>Upper</u>	<u>Lower</u>
2.2E-2	1.5E-1	0
- 6-8. Steps Related to Occupational Dose Increase for SIR Implementation:
 Because SIR implementation involves the specification and authorization of inspection procedures, no occupational dose will be accumulated. Thus, D = 0.
9. Per-Plant Utility Labor in Radiation Zones for SIR Operation and Maintenance:
 Increased Labor = 79.6 man-hr/py (see Attachment 2).
10. Per-Plant Occupational Dose Increase for SIR Operation and Maintenance (D_o):
 Assuming all work is performed in areas where the dose rate is 2.5 mR/hr,
 $D_o = (0.0025 \text{ R/hr})(79.6 \text{ man-hr/py}) = 0.20 \text{ man-rem/py}$.
11. Total Occupational Dose Increase for SIR Operation and Maintenance ($\bar{NTD}_o$):
 $\bar{NTD}_o = (134)(28.3 \text{ yr})(0.20 \text{ man-rem/py}) = 750 \text{ man-rem}$.

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem)	
	<u>Upper</u>	<u>Lower</u>
750	2.3E+3	250

ATTACHMENT 2

Basic SIR operation and maintenance requires weekly inspection and light maintenance (i.e., component condition check and cleaning, if necessary), versus the assumed current practice of monthly inspections. The most conservative case assumes that the repair time stays the same as in the base case and that preventive maintenance will substantially reduce repair frequency. To account for this it is assumed that the SIR reduces the repair frequency to that of the better plants as indicated by the AEOD data. Of 42 plants represented, 15 plants recorded 1 breaker failure to close on demand during the reporting period. Assuming each failure requires a two-person crew one eighthour shift to repair including preparation time,

$$\begin{aligned}\text{Repair time} &= \frac{(2 \text{ men})(8 \text{ hr})(1 \text{ failure/plant})}{(\text{subsequent to SIR}) \quad (5.25 \text{ yr})(4)^4} \\ &= 0.8 \text{ man-hr/py}\end{aligned}$$

$$\begin{aligned}(\text{prior to SIR}) &= (2 \text{ men})(8 \text{ hr})(1.4\text{E-}3 \text{ failure/plant-wk}^5)(52 \text{ wk/yr}) \\ &= 1.2 \text{ man-hr/py}\end{aligned}$$

Reduction in repair time = 0.4 man-hr/py.

While a 0.4 man-hr/py reduction in repair time will be realized from this SIR, the increase in inspection frequency will increase the time spent for routine inspection. A two-person crew is assumed to require one hour to inspect all diesel generator circuit breakers including preparation time,

$$\begin{aligned}\text{Inspection time} &= (2 \text{ men/plant})(1 \text{ hr/inspection})(1 \text{ insp./wk})(52 \text{ wk/yr}) \\ (\text{subsequent to SIR}) &= 104 \text{ man-hr/py}\end{aligned}$$

$$\begin{aligned}\text{Inspection time} &= (2 \text{ men/plant})(1 \text{ hr/insp.})(1 \text{ insp./month})(12 \text{ months/yr}) \\ (\text{prior to SIR}) &= 24 \text{ man-hr/py}\end{aligned}$$

Increase in inspection time = 80 man-hr/py

Therefore, the overall increase in time spent for SIR operation and maintenance is $[80 - 0.4] = 79.6 \text{ man-hr/py}$.

⁴ Factor of 1/4 applied to account for number of breaker failures in diesel generator system repair time (prior to SIR).

⁵ See Attachment 1.

3.0 SAFETY ISSUE COSTS

The analysis of the industry and NRC costs associated with SIR are presented in this section. Results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Failure of Class IE Safety-Related Security Circuit Breakers to Close on Demand (55)

2. Affected Plants (N):

N = 90 PWRs and 44 BWRs (All plant are assumed to be affected)

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

PWRs $\bar{T}$ = 28.8 yr

BWRs $\bar{T}$ = 27.4 yr

All plants $\bar{T}$ = 28.3 yr

Industry Costs (Steps 4 through 12):

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

PWR = $(\$1.65E+9)(3.2E-10/py)$ = \$0.53/py

BWR = $(\$1.65E+9)(2.4E-10/py)$ = \$0.40/py

5. Total Industry Cost Savings due to Accident Avoidance:

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.8E+3	\$1.2E+4	0

6. Per-Plant Industry Resources for SIR Implementation:

Backfit plants will presumably be required to prepare plans for SIR implementation as well as to amend their operating licenses. The resource breakdown is as follows:

Prepare Plan

Engineering Staff = 4 man/wk/plant

Secretarial Staff = 1 man/wk/plant

Management Staff = 0.5 man-wk/plant

Total = 5.5 man-wk/plant

The cost for a class III license amendment fee is included in the next step.

Forward-fit plants can incorporate the above plans in their initial procedures development. Thus, no additional cost for SIR implementation is foreseen (no license amendment needed either).

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

$$\begin{aligned} \text{Labor} &= (5.5 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$1.25\text{E}+4/\text{plant} \\ \text{License} &= \$4.0\text{E}+3/\text{plant} \\ \text{Amendment} &= \$1.65\text{E}+4/\text{plant} \end{aligned}$$

This applies only to backfit plants.

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

$$\text{NI} = (71)(\$1.65\text{E}+4/\text{plant}) = \$1.17\text{E}+6$$

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

As discussed in Attachment 2, an increase in labor of 79.6 man-hr/py results from weekly instead of monthly inspection/maintenance of Class IE diesel generator circuit breakers. In addition, training and qualification of crews may involve five men (one foreman and two two-man crews) for four hours per year, or a total of 20 man-hr/py. No incremental costs are assumed for tools and equipment or plant down-time. Thus, the total labor hours for SIR operation and maintenance become:

$$\begin{aligned} \text{Inspection time} &= 79.6 \text{ man-hr/py} \\ \text{Training time} &= 20 \text{ man-hr/py} \\ \text{Total} &= 99.6 \text{ man-hr/py, or } 2.5 \text{ man-wk/py} \end{aligned}$$

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

$$\text{I}_o = (2.5 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$5680/\text{py}$$

11. Total Industry Cost for SIR Operation and Maintenance (N $\bar{\text{I}}$ _o):

$$\text{N}\bar{\text{I}}_o = (134)(28.3 \text{ yr})(\$5680/\text{py}) = \$2.15\text{E}+7$$

12. Total Industry Cost (SI):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$2.3E+07	\$3.3E+07	\$1.2E+07

NRC Costs (Steps 13 through 21):

13. NRC Resources for SIR Development:

SIR development by the NRC is assumed to involve the following:

$$\begin{aligned} \text{Continue AEOD assessment of safety issue} &= 0.5 \text{ man-yr} \\ \text{Contract independent study} &= 0.5 \text{ man-yr} \\ \text{Issue I\&E Bulletin} &= 0.2 \text{ man-yr} \\ \text{Total} &= 1.2 \text{ man-yr} \end{aligned}$$

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

$$C_D = (1.2 \text{ man-yr}) (\$1.0\text{E}+5/\text{man-yr}) = \$1.2\text{E}+5$$

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

It is assumed that NRC review and approval of industry plans for compliance with the I&E Bulletin would involve one man-wk/plant. This applies only to backfit plants as this compliance would be part of initial procedures development at forward-fit plants.

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

$$C = (1 \text{ man-wk/plant}) (\$2270/\text{man-wk}) = \$2270/\text{plant}$$

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

$$NC = (\$2270/\text{plant}) (71) = \$1.61\text{E}+5$$

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

NRC labor to check utility compliance with the SIR through inspection/verification is assumed to be one man-hr/py.

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

$$C_o = (1 \text{ man-hr/py}) (1 \text{ man-wk}/40 \text{ man-hr}) (\$2270/\text{man-wk}) = \$56.8/\text{py}$$

20. Total NRC Cost for Review of SIR Operation and Maintenance (NTC_o):

$$NTC_o = (134) (28.3 \text{ yr}) (\$56.8/\text{py}) = \$2.15\text{E}+5$$

21. Total NRC Cost (SN):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$5.0E+5	\$6.4E+5	\$3.5E+5

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 76, Instrumentation and Control Power Interactions

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: DC electric power systems provide control and motive power to several vital and non-vital components, including Instrumentation and Control systems (ICS), emergency diesel generator controls, valves, and control rod drive (CRD) mechanisms. Many of these components are required to operate under abnormal and accident conditions. Experience has indicated that failures in DC power systems have occurred at a significant frequency. Under some conditions, losses of DC power may result in operator "blindness" due to a lack of ICS. The proposed resolution to this safety issue involves implementing technical specification and operating procedure revisions to increase the reliability of DC power systems at all plants and to implement hardware modifications at a subset of plants that exhibit relatively low DC power system reliability.

AFFECTED PLANTS: All plants, including those operating and under construction, are affected.

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION =	7.0E+3
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	1.0E+2

COST RESULTS (\$10⁶):

INDUSTRY COSTS:	
SIR Implementation =	2.1E+1
SIR Operation/Maintenance =	1.0E+1
Total of Above =	3.1E+1
Accident Avoidance =	8.3E+0
NRC COSTS:	
SIR Development =	1.9E-1
SIR Implementation =	2.7E-1
SIR Operation/Maintenance Review =	8.7E-1
Total of Above =	1.3E+0

INSTRUMENTATION AND CONTROL POWER INTERACTIONS

ISSUE 76

1.0 SAFETY ISSUE DESCRIPTION

DC electric power systems provide control and motive power to several vital and non-vital components, including Instrumentation and Control systems (ICS), emergency diesel generator controls, valves, and control rod drive (CRD) mechanisms. Many of these components are required to operate under abnormal and accident conditions. Typical DC electric power system designs include large-capacity batteries that are attached to DC electric buses that supply electric power to the components. Battery chargers are provided to ensure an adequate charge is maintained by the batteries.

Experience has indicated that failures in DC power systems have occurred at a significant frequency, as stated in an earlier draft prioritization analysis for this safety issue (NRC 1984 DRAFT). A number of these failures have had potential safety significance. A list of potentially significant events was provided by NRC (1984), including loss of DC power supplies for one hour, partial and total losses of normal and emergency AC power, loss of control room annunciators, control system malfunctions, reduction or loss of feedwater, and a variety of inadvertent valve actuations. The effects of these failures ranged from minimal effects on plant operation to reactor trips. Fixes that have been implemented to prevent recurrence of these events include modifications to operating procedures, changes to technical specifications, and repair or replacement of failed components.

Losses of DC power systems are typically recoverable in a short period of time. However, the subsequent effects of the DC power system failures may result in transients involving a series of multiple, propagating interactions that may lead to adverse conditions that are not readily reversed or correctable. As an example, under station blackout conditions (i.e., total loss of normal and emergency AC power supplies), the DC power system is intended to provide instrumentation and control power to vital plant instrumentation and valves. This includes providing power to primary coolant level and temperature indicators to allow operators to properly control primary coolant conditions. Under station blackout conditions with a simultaneous loss of DC power systems, operators may be "blinded" to the actual condition of the plant and may perform incorrect operations based on this lack of knowledge. In addition, DC power is needed to properly control AC-independent primary coolant makeup systems, such as turbine-powered high pressure coolant injection (HPCI), reactor core isolation cooling (RCIC) pumps in BWRs and auxiliary feedwater (AFW) pumps in PWRs. Under such conditions (operator blindness and loss of control), feed-and-bleed operation of the primary coolant system is precluded even though the partial or total loss of components may not necessarily rule out this mode of decay heat removal. It can be seen from this example that DC power is needed to perform the necessary operations to both diagnose reactor conditions and may also contribute to a lack of capability to recover from an incident.

One of the objectives of the present analysis is to update the database that was prepared in support of NRC (1984). This will be performed by searching the

Licensee Event Report (LER) database for the period 1984 to 1990. The second objective is to develop estimates of the core-damage frequency and public risk reduction that may be achievable through resolution of this issue. The LER database will be used where appropriate to develop core-melt frequency and public risk estimates. The occupational doses and costs associated with safety issue resolution (SIR) will also be estimated in this analysis.

PROPOSED RESOLUTION

The proposed resolution of this issue involves activities that are believed to be necessary to achieve an increase in the reliability of DC power systems. The overall goal of the SIR is to ensure that the plants can withstand a loss of DC power plus a single failure in any other system (NRC 1984). Initially, each plant is postulated to perform a failure modes and effects analysis (FMEA) of their DC power systems to determine the potential likelihood of DC power system failures at their plants, identify and evaluate the interactions of the DC power systems with other plant systems, and develop the basis for postulating plant-specific fixes to increase DC power system reliability. The plant-specific fixes are envisioned to range from relatively minor modifications to technical specifications or operating procedures to major hardware modifications to the DC power systems.

AFFECTED PLANTS

This safety issue affects all plants, including PWRs and BWRs. Plants operating and under construction are affected.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this issue are estimated in this section and summarized in Tables 1 and 2. Attachment 1 describes the basis for the risk and dose changes that are estimated in the Tables.

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

Instrumentation and Control Power Interactions (76)

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

All PWRs and BWRs are assumed to be affected.

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

3. Plants Selected for Analysis:

Representative PWR: Oconee-3

Representative BWR: Grand Gulf-1

4. Parameters Affected by SIR:

SIR was modeled as affecting DC power system failures as both potential initiating events that could lead to transients as well as contributing events to potential loss of offsite power transients. The approach to modeling the effects of SIR on DC power system failures as contributing events to core-melt sequences was the traditional prioritization analysis approach described by Andrews et al. (1983). The affected parameters in this instance are BATA and BATB in the Grand Gulf PRA. Losses of DC power systems were not addressed explicitly as initiating events in Andrews et al. (1983). Therefore, new event tree models were developed to address loss of DC power initiating events. Attachment 1 provides a more complete description of the approach to investigating DC power failures as both initiating and contributing events.

5. Base-Case Values for Affected Parameters:

The base-case values of BATA and BATB are both 0.001. These values were used in the analysis of DC power failures as contributing events. The base-case value for the frequency of DC power failures as initiating events was estimated to be 0.06/RY based on a review of LER data from 1984 to 1990. See Attachment 1 for additional details.

6. Affected Accident Sequences and Base-Case Frequencies:

	Frequency, RY^{-1}	
Initiating Event	6.0E-7	(Sequences A to F in Attachment 1)
Contributing Event	4.9E-7	
TOTAL	1.1E-6	

7. Affected Release Categories and Base-Case Frequencies:

The affected containment failure modes and release categories are containment failure modes γ (BWR-3) and δ (BWR-4), as described in Attachment 1.

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

$$\bar{F}_{\text{BWR}} = 1.1\text{E-6/RY} \quad \bar{F}_{\text{PWR}} = 1.1\text{E-6/RY}$$

The analysis was performed for Grand Gulf (BWR) and scaled to Oconee (PWR) using scaling relationships given by Andrews et al. (1983).

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

$$W_{\text{BWR}} = 3.1 \text{ person-rem/RY} \quad W_{\text{PWR}} = 2.6 \text{ person-rem/RY}$$

10. Adjusted-Case Values for Affected Parameters:

The adjusted-case values for parameters BATA and BATB were 3.3E-4 for both parameters. The adjusted-case value of the DC power initiating event frequency was 0.02/RY.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

	Frequency, RY^{-1}	
Initiating Event	2.0E-7	(Sequences A to F in Attachment 1)
Contributing Event	1.6E-7	
TOTAL	3.6E-7	

12. Affected Release Categories and Adjusted-Case Frequencies:

The affected containment failure modes and release categories are containment failure modes γ (BWR-3) and δ (BWR-4), as described in Attachment 1.

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

$$\bar{F}^*_{\text{BWR}} = 3.6\text{E-7/RY} \quad \bar{F}^*_{\text{PWR}} = 8.0\text{E-7/RY}$$

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

$$W^*_{\text{BWR}} = 1.0 \text{ person-rem/RY} \quad W^*_{\text{PWR}} = 0.86 \text{ person-rem/RY}$$

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

$$\Delta\bar{F}_{BWR} = 7.4E-7/R\bar{Y}$$

$$\Delta\bar{F}_{PWR} = 1.6E-6/R\bar{Y}$$

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

$$\Delta W_{BWR} = 2.1 \text{ person-rem/R}\bar{Y} \quad \Delta W_{PWR} = 1.7 \text{ person-rem/R}\bar{Y}$$

17. Total Public Risk Reduction, (ΔW)_{Total}:

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
7.0E+3	3.2E+5	0

ATTACHMENT 1

This attachment presents the basis for the estimated changes in core damage frequency and public risk reduction associated with SIR.

The first step in this analysis was to perform a search of Licensee Event Reports (LERs) to develop estimates of the historical frequency of DC power system failures. As indicated in NRC (1984), the frequency of a loss of one division of DC power, based on data collected from 1972 to 1983, was estimated to be about 0.02/RY. However, because the search criteria that resulted in this estimate cannot be verified at this time, these data will be used only for information purposes. A new LER search was performed for the period 1984 to 1990 to form the basis for DC power failure frequencies. This search used the key words, "DC POWER," and, "STATION BATTERY," to find the LERs of interest to this analysis. The resulting LERs were then reviewed in detail to determine which events involved safety-significant failures. This involved removing LERs that occurred during outages, technical specification violations on battery operability that were confined to a single battery or cell within a battery, engineered safeguards system (ESF) actuations that did not result in a reactor trip, and disabling of safety-related components, such as a HPCI pump, that had unaffected backup or redundant systems.

The LER review focused on both sudden complete loss of DC power systems as well potential electric power degradation, such as undervoltage, overvoltage, undercurrent, and overcurrent events. During the period 1984 to 1990, a total of 5 LERs were found that indicated there was DC power degradation. All 5 LERs involved undervoltage to a DC power bus or DC power control system. Of these 5 LERs, three were detected during routine inspection/maintenance during refueling outages and 2 resulted in reactor scrams. The latter two resulted in reductions in MFW flow to steam generators which resulted in low steam generator level and a subsequent reactor trip. All 5 events were detected by undervoltage detection devices on the DC power buses either immediately or during the investigation of the events. All of the devices were equipped with electric power transfer systems which are designed to transfer power to a different battery/bus when an undervoltage condition is detected. Based on this observation, the conservative assumption is that DC power degradation events present similar conditions to a sudden complete loss of DC power and should be included in the database.

There were no observed instances in the reviewed LERs of overvoltage- or overcurrent-induced failures. It is hypothesized that the DC-power driven components are protected by overcurrent and overvoltage protection devices (e.g., fuses and circuit breakers) that prevent the components from being damaged by such conditions. There were a number of fuse failures reported in the LERs which tend to suggest this hypothesis may be true. It appears that the reliability of the protective devices is high enough that there have been no component failures resulting from overcurrent or overvoltage from a DC power system.

The LER search revealed the wide range of components and systems that are reliant on DC power. There were a number of reactor trips that were caused by losses of DC control power to active components that resulted in reactor trips. Examples of affected components include main feedwater (MFW) flow control logic modules, MFW flow control valves, main turbine electro-hydraulic control systems, turbine governor valve control, control rod drive mechanisms, and reactor coolant

pump (RCP) trip breakers. Examples of reactor trips resulting from failure of DC power to all these systems were found. If such a trip were to occur and a plant experienced a simultaneous loss of AC power (station blackout), the plant would be in a difficult recovery situation, as discussed in Section 1.0. Consequently, it was decided to analyze the potential effects of loss of DC power and subsequent reactor trip as a potential initiating event. In addition, the effects of DC power failures will be analyzed separately as a system that is required to recover from a station blackout initiating event. Consequently, both potential safety significant situations in which DC power systems are required to function will be analyzed.

BWR ANALYSIS - DC POWER FAILURE INITIATING EVENTS

An event tree was developed to model a "generic" plant's response to a loss of DC power/reactor trip initiating event. Plant response after the initiating event was based on modifications to the station blackout event trees presented in NUREG-1032 (Baranowsky 1988). The basic accident sequence that was modeled is initiated by loss or degradation of DC control power to a component that results in its abnormal operation. As discussed previously, this abnormal operation could eventually result in a reactor trip. The next events to occur are postulated to be a loss of normal AC power, failure of the emergency AC power system (i.e., emergency diesel generators), and failure to recover AC power in sufficient time to avert core damage. For conservatism, it was assumed that failure to recover electric power in one hour would be sufficient to result in significant difficulties in plant recovery operations. It should be noted that the time in which AC power must be recovered in PWRs ranges from 1 to 2 hours, at worst, if primary coolant system integrity is not maintained; e.g., reactor coolant pump seal failures or stuck-open safety relief valves (Baranowsky 1988). The steam-driven train of the auxiliary feedwater (AFW) system, in conjunction with primary system pressure relief, would be the primary source of decay heat removal. In BWRs, the same recovery time range was presented by Baranowsky (1988) under conditions in which the isolation condensers are unavailable. The same applies to BWRs provided with high-pressure injection/RCIC systems rather than isolation condensers. If primary system integrity is not compromised, the time to recover AC power and prevent core damage may be as long as 4 to 16 hr.

The event tree model developed for this analysis is shown in Figure 1. The event tree also presents sequence outcomes as either "OK," meaning that stable long-term cooling is achieved or, "CD," meaning that core damage is expected. Core melt is assumed to be the likely outcome of all core damage sequences given the uncertainty in the ability to terminate a core melt in progress and the low probability of restoring AC power during this time period (Baranowsky 1988). The criteria for successful termination of an accident sequence are:

- offsite power available and primary coolant integrity maintained
- offsite power available and primary coolant makeup capability available
- emergency AC power available and primary coolant system integrity maintained
- emergency AC power available and primary coolant makeup system available.

OK = Successful termination of accident sequence; success criteria are presented in the text.
 CD = Sequence proceeds to core damage and core melt.

FIGURE 1. Reduced Event Tree Model for DC Power Failure Initiating Event

The next step in the analysis was to quantify the accident sequences in the figure that were designated sequences A to F. Derivation of the failure probabilities is described in the following paragraphs.

Initiating Event: The initiating event frequency was derived from the LER data analysis that was described previously. The results of the LER search are illustrated in Figure 2. The initiating event frequency was assumed to be the frequency of a loss of DC power followed by a reactor trip. This frequency was estimated to be about 0.06/RY over the period 1984 to 1990, assuming there were about 100 RY per calendar year over that period.

Reactor Trip: The probability of unsuccessful reactor trip is independent of DC power system failures. The failure probability of $7.7\text{E-}7$ was taken from NUREG/CR-2800 (Andrews, et al., 1983) for Grand Gulf.

AC Power: The probability of loss of normal AC power was estimated to be 0.118 (Baranowsky 1988). This event was assumed to be independent of loss of DC power systems. This value was also used in NRC (1984).

Emergency AC Power: The reliability of the emergency AC power systems was taken to be 0.994 (Baranowsky 1988). This value represents the reliability of a 1 out of 2 diesel generator system with a diesel generator reliability of 0.975.

Recovery of AC Power Within 1 Hour: The probability of recovery of AC power within 1 hour was estimated to be 0.55 (non-recovery = 0.45) based on information given by Baranowsky (1988) and NRC (1984).

Primary Coolant System Integrity: Primary system integrity is dependent upon success or failure of the prior events in the event tree. If AC power is available, it was assumed that RCP seal cooling is available and an RCP seal LOCA is not likely to occur. However, the subsequent transient is likely to result in an increase in primary coolant system pressure and temperature. The potential exists for a LOCA caused by a stuck-open safety relief valve with a probability of 0.025 (Baranowsky 1988). If AC power is not available, it becomes more likely that an RCP seal LOCA will occur. However, it was indicated by Baranowsky (1988) that the time to recover AC power to prevent an RCP seal LOCA is 4 to 16 hr whereas the AC power recovery time to prevent core damage from a stuck-open relief valve is 1 to 2 hr. Based on this observation, plants are more vulnerable to stuck-open relief valves than RCP seal LOCAs under these conditions. Therefore, a primary coolant system failure probability of 0.025 which is representative of a stuck-open relief valve is a conservative estimate of the failure of primary system integrity for all conditions in the event tree.

Primary Coolant System Makeup Capability: This event is dependent upon the success or failure of AC and DC power systems as well as operator actions. For sequences in which AC power is initially available (either normal or emergency power), it was assumed that operators are provided with accurate plant condition information and I&C systems. In these cases, the failure probability was assumed to be analogous to failure of decay heat removal systems (i.e., high pressure and low pressure injection systems or residual heat removal systems) in normally-operating BWRs. A representative failure probability of $1\text{E-}4$ for these systems was used based on information in Andrews, et al., (1983). For sequences in which AC and DC power are not available initially but AC power is recovered within 1 hr, it was assumed that there is a higher probability that operators take an

ANNUAL FREQUENCY			
Year	Loss of 1		Total
	Division	Reactor	
	DC Power	Trip	Records
1990	0.02	0.03	0.41
1989	0.02	0.07	0.57
1988	0.01	0.07	0.58
1987	0.02	0.07	0.57
1986	0.03	0.05	0.57
1985	0.02	0.06	0.43
1984	0.03	0.07	0.51
AVERAGE	0.021	0.060	0.520

FIGURE 2. Results of LER Search for DC Power System Failures

incorrect action because they are "blinded" for some period of time. The failure probability used in this analysis was 0.005, which is the failure probability for BWRs with HPCI/RCIC systems that was given by Baranowsky (1988). A third potential situation exists in which no AC and DC power is available and AC power is not restored within 1 hr. In this case, there is essentially no I&C power and operators are likely to be blinded to actual plant conditions. However, successful recovery from this sequence is dependent upon primary coolant system integrity. If the primary coolant system is intact, operators will have a substantial period of time to effect recovery actions, which may include manual valve actuations or other actions that may lead to recovery of I&C power. Therefore, a failure probability of 0.01 was used for this sequence. If, however, primary coolant system integrity is lost, e.g. through a stuck-open safety relief valve, recovery time to prevent core damage is significantly reduced. A relatively high failure probability of 0.5 was used in this analysis to represent the potential high probability of failure due to confusion and high stress that will result from operator blindness.

Based on the failure probabilities discussed above, the frequencies of the core-melt accident sequences designated A to F in Figure 1 are:

<u>Sequence</u>	<u>Frequency, RY^{-1}</u>
A	1.3E-7
B	1.8E-8
C	2.9E-9
D	1.9E-7
E	2.4E-7
F	1.6E-8
TOTAL	6.0E-7

The total represents the base-case affected core-melt frequency for loss of DC power as an initiating event.

The affected public risks associated with this issue were calculated by multiplying the affected core-melt frequency given above by the containment failure probabilities and dose consequences associated with similar accident sequences from the Grand Gulf PRA. It was determined that the plant conditions that would result from the accident sequences described in this analysis would be similar to those expected after sequence T₁QUV in the Grand Gulf PRA. This sequence involves a loss of offsite power, failure of the power conversion system, failure of HPCS/RCIC, and failure of low-pressure emergency core cooling systems. This accident sequence is approximately the same as the loss of offsite power, loss of emergency power, failure of primary coolant integrity, and failure of primary coolant makeup systems that are described in Figure 1. The calculations leading to the base-case affected public risks, which involve the affected core-melt frequency associated with this issue and the containment failure probabilities, release categories, and dose consequences for sequence T₁QUV are:

<u>Affected Core melt Frequency, RY⁻¹</u>	<u>Containment Failure Mode, (prob.)</u>	<u>Release Category</u>	<u>Public Dose, person-rem</u>	<u>Public Risk person-rem/RY</u>
6.0E-7	γ (0.5)	BWR-3	5.1E+6	1.53
	δ (0.5)	BWR-4	6.1E+5	<u>0.18</u>
			TOTAL:	<u>1.71</u>

The adjusted-case core-melt frequency and public risk estimates were developed by revising the DC power failure initiating event frequency to account for the increased reliability that could result from implementation of SIR. SIR was assumed to increase the reliability of DC power systems to the high end of battery failure rate distributions given in the Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR - Gertman, et al., 1988). The error factors given in NUCLARR for seven battery failure rate data points ranged from a low of 2 to a high of about 8. The average of these seven data points was 4.76. For conservatism, an increase in battery reliability of 3.0 was assumed in the calculations. This increase in battery reliability was assumed to be applicable to the entire DC power system, given that the proposed resolution involves a program to enhance the reliability of the entire DC power system and is not limited to batteries. Therefore, the adjusted-case DC power system initiating event frequency is 0.06/RY divided by 3 = 0.02/RY. The adjusted-case initiating event frequency was substituted into the core-melt frequency calculations for sequences A to F, resulting the following accident sequence frequencies:

<u>Sequence</u>	<u>Frequency, RY⁻¹</u>
A*	4.3E-8
B*	6.0E-9
C*	9.7E-10
D*	6.3E-8
E*	8.0E-8
F*	<u>5.3E-9</u>
TOTAL:	<u>2.0E-7</u>

The total represents the adjusted-case affected core-melt frequency for loss of DC power as an initiating event.

The adjusted-case public risk calculations were performed similarly to the base-case public risk calculations and are shown below.

<u>Affected Core melt Frequency, RY⁻¹</u>	<u>Containment Failure Mode, (prob.)</u>	<u>Release Category</u>	<u>Public Dose, person-rem</u>	<u>Public Risk person-rem/RY</u>
2.0E-7	γ (0.5)	BWR-3	5.1E+6	0.51
	δ (0.5)	BWR-4	6.1E+5	<u>0.06</u>
			TOTAL:	<u>0.57</u>

BWR ANALYSIS - DC POWER FAILURES AS CONTRIBUTING EVENTS

The other situation in which DC power is vital to reactor safety is in responding to a loss of offsite power. These sequences are addressed in the Grand Gulf PRA (NUREG/CR-2800) so no additional event trees were necessary to model DC

power as a contributing event. The effects of DC power failures as contributing events were added to their effects as initiating events to determine the total affected core-damage frequency and public risks associated with this issue.

DC power system failures are represented by events "BATA" and "BATB," (see Andrews, et al., 1983). The base-case failure probabilities for both these events are 0.001. As was done for the initiating event analysis described previously, the effects of SIR were assumed to increase the reliability of the batteries and the entire DC power system by a factor of 3. Therefore, the adjusted-case values of parameters BATA and BATB were estimated to be 0.00033.

The Grand Gulf PRA computer code was used to determine the base-case affected core-damage frequency and public risks for this issue. The base-case and adjusted-case core-melt frequency from the computer codes were estimated to be $4.9\text{E-}7/\text{RY}$ and $1.6\text{E-}7/\text{RY}$, respectively. The base-case and adjusted-case public risk estimates were 1.4 person-rem/Ry and 0.46 person-rem/Ry, respectively.

BWR TOTAL CORE-MELT FREQUENCY AND PUBLIC RISK ESTIMATES

The total core-melt frequency and public risk estimates for BWRs are the sum of the respective values presented for initiating and contributing events. These calculations are shown below.

<u>DC Power Sequence</u>	<u>Core-Melt Frequency, RY^{-1}</u>		<u>Risk, person-rem/Ry</u>	
	<u>Base-Case</u>	<u>Adjusted-Case</u>	<u>Base-Case</u>	<u>Adjusted-Case</u>
Initiating Event	6.0E-7	2.0E-7	1.7	0.57
Contributing Event	4.9E-7	1.6E-7	1.4	0.46
TOTAL:	<u>1.1E-6</u>	<u>3.6E-7</u>	<u>3.1</u>	<u>1.03</u>

PWR ANALYSIS

Batteries and DC power system failures were not listed as events that occurred in any of the dominant accident sequences at Oconee-3 (Andrews, et al., 1983). This eliminates the possibility of analyzing separately the effects of SIR on PWRs in a similar manner as was done for BWRs. Therefore, the scaling equations given by Andrews, et al., (1983) were used to translate the core-melt frequencies and public risk estimates that were calculated for Grand Gulf to a representative approximation for Oconee. The equations are:

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

$$(W)_{\text{BWR}} = 1.20 (W)_{\text{PWR}}$$

Substituting the quantities derived previously for $(\bar{F})_{\text{BWR}}$ and $(W)_{\text{BWR}}$ yields the following results for the generic PWR:

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

$$(W)_{PWR} = (W)_{BWR}/1.2$$

These equations were applied to the total core-melt frequency and public risk estimates that were presented previously for BWRs (i.e., the sum of the initiating event and contributing event effects shown on p. 14). Using these equations, the following core-melt frequency and public risk estimates for Oconee were calculated.

<u>Core-Melt Frequency, RY⁻¹</u>		<u>Risk, person-rem/RY</u>	
<u>Base-Case</u>	<u>Adjusted-Case</u>	<u>Base-Case</u>	<u>Adjusted-Case</u>
2.4E-6	8.0E-7	2.6	0.86

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Instrumentation and Control Power Interactions (76)

2. Affected Plants (N):

	<u>N</u>
PWRs	90
BWRs	44
All plants	134

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

	<u>$\bar{T}$</u>
PWRs	28.8 yr
BWRs	27.4 yr
All plants	28.3 yr

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

$$\Delta(\bar{F}D_R)_{PWR} = (19,860 \text{ person-rem})(1.6E-6/R\bar{Y}) = 0.032 \text{ person-rem/R}\bar{Y}$$

$$\Delta(\bar{F}D_R)_{BWR} = (19,860 \text{ person-rem})(7.4E-7/R\bar{Y}) = 0.015 \text{ person-rem/R}\bar{Y}$$

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

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
100	900	0

6-12. Steps leading to calculation of total utility labor in radiation zones and occupational dose increases due to SIR implementation, operation, and maintenance.

Incremental labor in radiation zones and occupational dose increases due to SIR implementation, operation, and maintenance are estimated to be 0. This estimate was based on information on potential battery enhancements developed in support of NUREG-1109 (Rubin 1988).

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis. Attachment 2 provides the basis for these cost estimates.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Instrumentation and Control Power Interactions (76)

2. Affected Plants (N):

	<u>N</u>
PWRs	90
BWRs	44
All Plants	134

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

	<u>$\bar{T}$</u>
PWRs	28.8 yr
BWRs	27.4 yr
All plants	28.3 yr

Industry Costs (Steps 4 through 12):

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

$$\Delta(\bar{F}A)_{PWR} = (\$1.65E+9)(1.6E-6/R\bar{Y}) = \$ 2.6E+3/R\bar{Y}$$

$$\Delta(\bar{F}A)_{BWR} = (\$1.65E+9)(7.4E-7/R\bar{Y}) = \$ 1.2E+3/R\bar{Y}$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$ 8.3E+6	\$ 7.5E+7	0

6. Per-Plant Industry Resources for SIR Implementation:

Per-plant industry resources for SIR implementation were estimated in Attachment 2 for plants requiring revised technical specifications and operating procedures at about 9 man-months/plant plus additional resources for preparing technical specification and operating procedure revisions. Industry labor at plants requiring hardware modifications were estimated at about 2,060 man-hr/plant plus engineering/QA and subcontractor markup.

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

Per-plant industry costs for SIR implementation were estimated as follows:

Technical specification/procedure revisions = \$ 99,000/plant
Hardware modifications = \$ 275,000/plant

The former applies to all 134 plants and the latter applies to 27 plants with relatively low DC power system reliability.

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

NI = (134 plants)(\$99,000/plant) + (27 plants)(\$ 275,000/plant)
= \$ 2.07E+7

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

Industry labor for SIR operation/maintenance was estimated to be approximately 48 man-hr/Ry at the plants requiring hardware modifications to increase DC power reliability. No additional resources are foreseen for plants requiring only technical specification revisions.

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

I_o = [(48 man-hr/Ry)/(40 man-hr/man-wk)] (\$ 2,270/man-wk)
= \$ 2,724/Ry

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

NTI_o = (\$2,724/Ry)(134 plants)(28.3Ry) = \$ 1.03E+7

12. Total Industry Cost (S_i):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$ 3.1E+7	\$ 4.2E+7	\$ 1.9E+7

NRC Cost (Steps 13 through 21)

13-20. Steps Related to NRC Costs for SIR Development, Support of SIR Implementation, and Review of SIR Operation and Maintenance

13. NRC Resources for SIR Development:

NRC development costs are estimated to include a 1.0 man-yr contractor study, 0.1 man-yr for an NRC program monitor, and 0.75 man-yr for value-impact and backfit analyses.

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

$$C_D = (1.85 \text{ man-yr})(\$ 100,000/\text{man-yr}) = \$ 185,000$$

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

NRC resources for support of SIR implementation are estimated to include approximately 0.5 man-wk/plant for reviewing proposed technical specification/operating procedure revisions and 2 man-wk/plant for those plants requiring hardware modifications as well as technical specification revisions.

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

$$\begin{aligned} C (\text{tech. spec. revisions}) &= (0.5 \text{ man-wk/plant})(\$ 2,270/\text{man-wk}) \\ &= \$ 1,135/\text{plant} \\ C (\text{hardware modifications}) &= (2.0 \text{ man-wk/plant})(\$ 2,270/\text{man-wk}) \\ &= \$ 4,540/\text{plant} \end{aligned}$$

The former applies to all 134 plants and the latter applies to 27 plants with relatively low DC power system reliability.

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

$$\begin{aligned} NC &= (\$ 1,135/\text{plant})(134 \text{ plants}) + (\$ 4540/\text{plant})(27 \text{ plants}) \\ &= \$ 273,000 \end{aligned}$$

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

Additional labor requirements to support SIR operation and maintenance were estimated in Attachment 2 to be about 0.5 man-wk/Ry. This estimate is assumed to be applicable only to the 27 plants that were judged to have relatively low DC power system reliability.

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

$$C_o = (0.5 \text{ man-wk/Ry})(\$ 2,270/\text{man-wk}) = \$ 1,135/\text{Ry}$$

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

$$\overline{N}C_o = (\$ 1,135/\text{Ry})(27 \text{ plants})(28.3 \text{ Ry}) = \$ 867,000$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.3E+6	\$1.8E+6	\$8.4E+5

ATTACHMENT 2 (To Table 3)

The proposed resolution is intended to result in an increase in the reliability of DC power systems. The overall goal of the SIR is to ensure that the plants can withstand a loss of DC power plus a single failure in any other system (NRC 1984). Because of significant differences in the designs among plants in the designs of their DC power systems, the proposed resolution is assumed to involve plant-specific resolutions to be developed by each licensee.

LICENSEE COSTS

As a first step, each plant is postulated to be required to perform a failure modes and effects analysis (FMEA) of their DC power systems. The FMEA will be used to determine the potential likelihood of DC power system failures at each plant, identify and evaluate the interactions of the DC power systems with other plant systems, and develop the basis for postulating plant-specific fixes to increase DC power system reliability.

The plant-specific fixes are envisioned to range from relatively minor modifications to technical specifications and operating procedures to major hardware modifications to the DC power systems. Plants that appear to have especially unreliable DC power systems, either from a historical or analytical standpoint, are postulated to implement hardware modifications to achieve the increased reliability. Possible hardware modifications to DC power systems that may be capable of increasing reliability are described by Clark, et al., (1984). For this analysis, the costs for only battery enhancements are included in the hardware modification costs. The cost estimates for these enhancements are shown below (Clark, et al., 1984):

• Labor hours (1380 hrs at 0.67 productivity)	=	2059 man-hr	
• Labor cost	=	\$ 56,905	
• Materials	=	118,548	
	SUBTOTAL:	\$ 175,500	(Rounded)
• Engineering/quality assurance (@ 25%)	=	43,900	
	SUBTOTAL:	219,400	
• Markup (@ 25%)	=	55,000	
	TOTAL:	275,000	(Rounded)

This cost estimate is assumed to apply to only those plants with relatively unreliable DC power systems. A total of 27 plants were placed in this category, based on information in NUREG-1109 (Rubin 1988).

The costs for plants that will increase DC power system reliability through revised technical specifications and operating procedures were taken primarily from Sciacca (1989). Technical specification revisions may be implemented to increase DC power inspection/test frequencies. Current technical specifications at a typical plant require surveillance testing of battery electrolyte levels, electrolyte specific gravity, minimum voltage for each cell and total battery voltage, and charger current on a 92-day surveillance interval. It is postulated that all plants will receive some improved reliability by reducing the surveillance interval to 30 days for these parameters. The costs for such a

change to technical specifications are estimated to be \$18,000/plant (Sciacca 1989).

Revisions to plant operating procedures are also anticipated to be involved in the proposed resolution. This will include the costs to rewrite procedures as well as provide operator training, including simulator training, in the new procedures. According to Sciacca (1989), the costs to rewrite procedures are approximately \$3,900/plant. Training costs were estimated at about \$27,240/plant, assuming approximately 3 man-months of licensee resources are required to develop and implement the revised training programs.

Annual costs necessary for operation/maintenance of the proposed SIR are assumed to include approximately 48 man-hr/Ry. This estimate includes periodic retraining as well as additional time required to perform more frequent surveillance tests on the batteries. The estimated annual cost is therefore \$2,724/Ry.

A summary of the industry costs for implementation, operation, and maintenance of the proposed SIR is:

A. Implementation Costs

All Plants

A-1 FMEA (0.5 man-yr/plant @ \$100,000/man-yr)	=	\$	50,000
A-2 Revise Tech Specs	=		18,000
A-3 Rewrite Operating Procedures	=		3,900
A-4 Training (@ 3 man-months/plant)	=		27,240
			<hr/>
	SUBTOTAL	=	\$ 99,000
			<hr/>
	Number of affected plants	=	134
			<hr/>
	TOTAL	=	\$ 13,300,000

Plants Requiring Hardware Modifications

A-5 Hardware modifications	=	\$	275,000
			<hr/>
	Number of affected plants	=	27
			<hr/>
	TOTAL	=	\$ 7,400,000

TOTAL IMPLEMENTATION COSTS = \$ 20,700,000

B. Operation/Maintenance Costs

All Plants	=	\$	2,724/Ry
			<hr/>
	Number of affected plants	=	134
			<hr/>
	Average remaining lifetime	=	28.3 yr

TOTAL OPERATION/MAINTENANCE COSTS = \$ 10,300,000

NRC COSTS

Estimated NRC resource requirements and costs to develop the SIR, support its implementation, and for review of operation/maintenance are summarized below.

C. SIR Development

C-1	Review and update existing data, establish reliability goal, determine SIR feasibility, and develop technical findings document (@ 1 man-yr of contractor effort)		\$	100,000
C-2	NRC technical oversight (@ 10%)			10,000
C-3	Prepare value/impact and backfit analysis			<u>75,000</u>
TOTAL NRC DEVELOPMENT COSTS		=	\$	<u>185,000</u>

D. Support SIR Implementation

D-1	Review FMEA and tech. spec. revisions (0.5 man-wks/plant)		\$	1,135
	Number of affected plants	=		<u>134</u>
	SUBTOTAL	=	\$	150,000
D-2	Review hardware modifications (2 man-wk/plant)		\$	4,540
	Number of affected plants	=		<u>27</u>
	SUBTOTAL	=	\$	123,000
TOTAL NRC COSTS FOR IMPLEMENTATION SUPPORT		=	\$	<u>273,000</u>

E. Support for Operation/Maintenance

E-1	0.5 man-wk/Ry for plants requiring hardware modifications	=	\$	1,135
	Number of affected plants	=		27
	Average remaining lifetime	=		<u>28.3 yr</u>
TOTAL OPERATION/MAINTENANCE SUPPORT COSTS		=	\$	<u>867,000</u>

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 78, Monitoring of Fatigue Transient Limits for Reactor Coolant System

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: Standard Technical Specifications for all new nuclear power plants require utilities to keep account of transient occurrences to assure that transient limits based on design assumptions are not exceeded. However, a number of older plants for which detailed fatigue analyses were performed on ASME Class 1 components do not have technical specification requirements to monitor actual transient occurrences. The fatigue life of the reactor coolant system may be reduced by the number of actual transient occurrences. The proposed resolution of this issue is to require the affected plants to implement technical specifications to monitor plant transients and to verify that the design life of all Class 1 components has not been exceeded.

AFFECTED PLANTS: This issue affects a total of 46 operating plants.

<u>RISK/DOSE RESULTS (man-rem):</u>	<u>Current Operating Lifetime</u>	<u>Extended Plant Lifetime</u>
PUBLIC RISK REDUCTION =	1.3E+3	2.8E+3
OCCUPATIONAL DOSES:		
SIR Implementation =	0	0
SIR Operation/Maintenance =	0	0
Total of Above =	0	0
Accident Avoidance =	1.1E+1	2.4E+1

COST RESULTS (\$10⁶):

INDUSTRY COSTS:		
SIR Implementation =	4.6E+0	4.6E+0
SIR Operation/Maintenance =	9.1E-1	1.4E+0
Total of Above =	5.5E+0	6.0E+0
Accident Avoidance =	9.3E-1	2.0E+0
NRC COSTS:		
SIR Development =	9.1E-2	9.1E-2
SIR Implementation =	6.3E-1	6.3E-1
SIR Operation/Maintenance Review	9.1E-1	1.4E+0
Total of Above =	1.6E+0	2.1E+0

MONITORING OF FATIGUE TRANSIENT LIMITS ON REACTOR COOLANT SYSTEM

ISSUE 78

1.0 SAFETY ISSUE DESCRIPTION

Fatigue is a material degradation process that involves repeated applications of a stress. Although the stress applied to the material may be too low to result in appreciable plastic deformation, the repeated application of the load may result in a structural failure. Transients which result in a reactor shutdown will cause the temperatures of the reactor vessel and reactor coolant system (RCS) components to decrease. Such cooling produces stresses in the RCS materials which, in and of themselves, are too small to produce appreciable damage to the materials. However, repeated shutdowns and RCS cooldowns produces some degree of fatigue degradation of the RCS materials.

A particularly severe example of a transient involves inadvertent actuation of safety injection (SI). Actuation of SI results in injection of cold water into the RCS. Injection of cold water into the RCS may result in the RCS temperature being reduced at a rate much faster than the normal rate of 100°F per hour. This and other transients which may cool the RCS at a rate faster than the normal cooldown rate may reduce the fatigue life of the RCS.

Standard Technical Specifications for all new nuclear power plants require utilities to keep account of the number of transient occurrences to assure that transient limits based on design assumptions are not exceeded. However, a number of older plants for which detailed fatigue analyses were performed on ASME Class 1 (reactor coolant pressure boundary) components do not have technical specification requirements to monitor actual transient occurrences.

Two sets of cost and risk information are developed for this analysis. The first assumes plants will operate for their presently-licensed operating lifetimes. The second assumes extended plant operating lifetimes. An additional 20 yrs. of plant operating life is included in the extended plant life case. Of this 20 yrs., a 75% availability factors was applied, effectively adding 15 yrs of power operation to the current plants.

PROPOSED RESOLUTION

The proposed resolution of this issue is to require the affected plants to implement technical specifications to monitor plant transients and to verify that the design life of all Class 1 components has not been exceeded. Plants that have experienced cooldown events that exceeded design limits may find it necessary to perform fatigue analyses to determine the number of remaining cooldowns before fatigue limits are exceeded. The requirement will be transmitted from the NRC to the affected plants in the form of a letter requesting submittal of a proposed technical specification for monitoring of design basis fatigue transient limits of the RCS and retention of records of transients or operational cycles for the duration of the Unit Operating License (Koo 1983).

AFFECTED PLANTS

This safety issue affects a total of 46 operating plants (Koo 1983). In NUREG/CR-2800 (Andrews, et al., 1983), it was shown that there were 47 operating PWRs and 24 operating BWRs (total of 71 operating plants) at the time of its publication. Maintaining the ratios of operating PWRs and BWRs to total plants, it is estimated that 30 operating PWRs and 16 operating BWRs will be affected by this issue.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this issue are estimated in this section and summarized in Tables 1 and 2. Attachment 1 describes the basis for the risk and dose changes that are estimated in the Tables.

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

Monitoring of Fatigue Transient Limits in Reactor Coolant System (78)

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

A total of 30 operating PWRs and 16 operating BWRs are estimated to be affected. Two cases are evaluated regarding the average remaining lives of the affected plants. The first assumes the currently licensed average remaining lifetime (27.7 and 25.2 yrs for PWRs and BWRs, respectively). The second case assumes plant life extension occurs. An additional 20 yrs of plant operations were assumed. Over this additional 20 yrs, plant availability was assumed to be 75%, so the plant life extension case adds an additional 15 operating yrs to the currently licensed operating lifetime (42.7 and 40.2 yrs. for PWRs and BWRs, respectively).

3. Plants Selected for Analysis:

Representative PWR: Oconee-3
Representative BWR: Grand Gulf-1

4. Parameters Affected by SIR:

<u>Oconee</u>	<u>Grand Gulf</u>
S_1 : Large LOCA	S : LOCA
S_2 : Medium LOCA	
S_3 : Small LOCA	

5. Base-Case Values for Affected Parameters:

Current Plant Lifetime Case

<u>Oconee</u>	<u>Grand Gulf</u>
$S_1 = 0.000102/\text{RY}$	$S : \text{LOCA} = 0.00143/\text{RY}$
$S_2 = 0.00041/\text{RY}$	
$S_3 = 0.00133/\text{RY}$	

Extended Plant Lifetime Case

<u>Oconee</u>	<u>Grand Gulf</u>
$S_1 = 0.000103/\text{RY}$	$S : \text{LOCA} = 0.00144/\text{RY}$
$S_2 = 0.000412/\text{RY}$	
$S_3 = 0.00134/\text{RY}$	

See Attachment 1 for details of the derivation of these values.

6. Affected Accident Sequences and Base-Case Frequencies:

The core-melt frequency calculations were performed using the OCONEE and GRAND GULF computer codes. The results for the current plant operating lifetime cases are as follows. The results for the extended plant lifetime case are not presented here for brevity.

<u>Oconee</u>			<u>Grand Gulf</u>		
S ₃ H	γ (PWR-3)	= 5.12E-6/RY	SI	α (BWR-1)	= 4.70E-8/RY
	β (PWR-5)	= 7.47E-8/RY		δ (BWR-2)	= 4.70E-6/RY
	ϵ (PWR-7)	= 5.12E-6/RY			
S ₁ D	α (PWR-1)	= 6.83E-8/RY			
	γ (PWR-3)	= 1.37E-8/RY			
	β (PWR-5)	= 4.99E-8/RY			
	ϵ (PWR-7)	= 5.47E-6/RY			
S ₃ FH	γ (PWR-2)	= 2.15E-6/RY			
	β (PWR-4)	= 3.14E-8/RY			
	ϵ (PWR-6)	= 2.15E-6/RY			
S ₂ FH	α (PWR-1)	= 1.33E-8/RY			
	β (PWR-4)	= 9.73E-9/RY			
	ϵ (PWR-6)	= 1.07E-6/RY			
S ₂ D	α (PWR-1)	= 2.05E-8/RY			
	γ (PWR-3)	= 4.10E-7/RY			
	β (PWR-5)	= 1.50E-8/RY			
	ϵ (PWR-7)	= 1.64E-6/RY			
S ₃ D	γ (PWR-3)	= 7.16E-7/RY			
	β (PWR-5)	= 1.05E-8/RY			
	ϵ (PWR-7)	= 7.16E-7/RY			

Note: These sequence frequencies do not contain the contributions from the non-dominant minimal cut sets.

7. Affected Release Categories and Base-Case Frequencies:

<u>Oconee</u>		<u>Grand Gulf</u>	
PWR-1:	1.12E-7/RY	BWR-1:	5.22E-8/RY
PWR-2:	2.50E-6/RY	BWR-2:	4.77E-6/RY
PWR-3:	8.00E-6/RY		
PWR-4:	5.20E-8/RY		
PWR-5:	1.56E-7/RY		
PWR-6:	4.16E-6/RY		
PWR-7:	1.38E-5/RY		

Note: These frequencies do contain the contributions from the non-dominant minimal cut sets. Only the current plant lifetime case is presented.

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

Current Plant Lifetime Case

$$\bar{F}_{PWR} = 2.88E-5/RY \quad \bar{F}_{BWR} = 4.82E-6/RY$$

Extended Plant Lifetime Case

$$\bar{F}_{PWR} = 2.90E-5/RY \quad \bar{F}_{BWR} = 4.855E-6/RY$$

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

Current Plant Lifetime Case

$$W_{PWR} = 56.73 \text{ person-rem/RY} \quad W_{BWR} = 34.14 \text{ person-rem/RY}$$

Extended Plant Lifetime Case

$$W_{PWR} = 57.17 \text{ person-rem/RY} \quad W_{BWR} = 34.38 \text{ person-rem/RY}$$

10. Adjusted-Case Values for Affected Parameters:

These values apply to both the current and extended plant lifetime cases.

Oconee

$$S_1 = 1E-4/RY$$

$$S_2 = 4E-4/RY$$

$$S_3 = 0.0013/RY$$

Grand Gulf

$$S : \text{LOCA} = 0.0014/RY$$

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Oconee:

$$\begin{aligned} S_3H \quad \gamma \text{ (PWR-3)} &= 5.00E-6/RY \\ \beta \text{ (PWR-5)} &= 7.30E-8/RY \\ \epsilon \text{ (PWR-7)} &= 5.00E-6/RY \end{aligned}$$

$$\begin{aligned} S_1D \quad \alpha \text{ (PWR-1)} &= 6.70E-8/RY \\ \gamma \text{ (PWR-3)} &= 1.34E-8/RY \\ \beta \text{ (PWR-5)} &= 4.89E-8/RY \\ \epsilon \text{ (PWR-7)} &= 5.36E-6/RY \end{aligned}$$

$$\begin{aligned} S_3FH \quad \gamma \text{ (PWR-2)} &= 2.10E-6/RY \\ \beta \text{ (PWR-4)} &= 3.07E-8/RY \\ \epsilon \text{ (PWR-6)} &= 2.10E-6/RY \end{aligned}$$

$$\begin{aligned} S_2FH \quad \alpha \text{ (PWR-1)} &= 1.30E-8/RY \\ \beta \text{ (PWR-4)} &= 9.49E-9/RY \\ \epsilon \text{ (PWR-6)} &= 1.04E-6/RY \end{aligned}$$

$$\begin{aligned} S_2D \quad \alpha \text{ (PWR-1)} &= 2.00E-8/RY \\ \gamma \text{ (PWR-3)} &= 4.00E-7/RY \\ \beta \text{ (PWR-5)} &= 1.46E-8/RY \\ \epsilon \text{ (PWR-7)} &= 1.60E-6/RY \end{aligned}$$

Grand Gulf:

$$\begin{aligned} SI \quad \alpha \text{ (BWR-1)} &= 4.60E-8/RY \\ \delta \text{ (BWR-2)} &= 4.60E-6/RY \end{aligned}$$

$$\begin{array}{lcl}
 S_3D & \gamma \text{ (PWR-3)} & = 7.00E-7/RY \\
 & \beta \text{ (PWR-5)} & = 1.02E-8/RY \\
 & \epsilon \text{ (PWR-7)} & = 7.00E-7/RY
 \end{array}$$

Note: These sequence frequencies do not contain the contributions from the non-dominant minimal cut sets.

12. Affected Release Categories and Adjusted-Case Frequencies:

Oconee	Grand Gulf
PWR-1: 1.10E-7/RY	BWR-1: 5.11E-8/RY
PWR-2: 2.44E-6/RY	BWR-2: 4.67E-6/RY
PWR-3: 7.82E-6/RY	
PWR-4: 5.08E-8/RY	
PWR-5: 1.53E-7/RY	
PWR-6: 4.06E-6/RY	
PWR-7: 1.35E-5/RY	

Note: These frequencies do contain the contributions from the non-dominant minimal cut sets.

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

$$\bar{F}^*_{PWR} = 2.81E-5/RY \quad \bar{F}^*_{BWR} = 4.72E-6/RY$$

These values apply to both the current and extended plant lifetime cases.

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

$$W^*_{PWR} = 55.48 \text{ person-rem/RY} \quad W^*_{BWR} = 33.42 \text{ person-rem/RY}$$

These values apply to both the current and extended plant lifetime cases.

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

Current Plant Lifetime Case

$$\Delta\bar{F}_{PWR} = 6.34E-7/RY \quad \Delta\bar{F}_{BWR} = 1.01E-7/RY$$

Extended Plant Lifetime Case

$$\Delta\bar{F}_{PWR} = 8.58E-7/RY \quad \Delta\bar{F}_{BWR} = 1.35E-7/RY$$

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

Current Plant Lifetime Case

$$\Delta W_{PWR} = 1.26 \text{ person-rem/RY} \quad \Delta W_{BWR} = 0.72 \text{ person-rem/RY}$$

Extended Plant Lifetime Case

$$\Delta W_{PWR} = 1.70 \text{ person-rem/RY} \quad \Delta W_{BWR} = 0.96 \text{ person-rem/RY}$$

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

	Best Estimate (person-rem)	Error Bounds (person-rem)	
		<u>Upper</u>	<u>Lower</u>
Current Plant Lifetimes	1.3E+3	1.8E+6	0
Extended Plant Lifetimes	2.4E+3	2.9E+6	0

ATTACHMENT 1

As discussed previously, fatigue may reduce the life of RCS components to below the lifetime predicted during plant design. This results in a reduction in the reliability of RCS components or, alternatively, increases the failure rate of these components. The failure rates of RCS components are manifested in the frequencies of loss-of-coolant accidents (LOCAs). The effects on plant risk of the fatigue degradation of RCS components would be to increase the likelihood (or frequency) of a LOCA.

The affected parameters in the Grand Gulf and Oconee PRAs (see Andrews, et al., 1983) were determined to be the LOCA initiating event frequencies. The original values of these parameters are shown below:

<u>Oconee</u>	<u>Grand Gulf</u>
S_1 : Large LOCA = $1E-4/RY$	S : LOCA = $0.0014/RY$
S_2 : Medium LOCA = $4E-4/RY$	
S_3 : Small LOCA = $0.0013/RY$	

The values for these parameters were developed assuming that the fatigue life for RCS components were as-designed. They did not take into account potential overcooling events or transients that may result in RCS cooldown rates in excess of limits which may reduce the design life of RCS components. Therefore, these values represent the adjusted-case values of affected parameters as they were derived assuming the design limits would not be exceeded and the number of allowable thermal cycles is the as-designed number. Therefore, these values represent plant conditions representative of the improvement they would achieve through resolution of this issue.

The base-case values of the affected parameters were developed in consideration of the actual fatigue degradation that may be experienced by RCS components. This degradation has been shown to reduce the lifetime of RCS components which will in turn increase the frequency of RCS piping failures. This increase in frequency is assumed to be inversely proportional to the reduction in the number of allowable cooldown cycles with respect to the as-designed number of cycles that a plant may experience.

A fatigue analysis was performed in support of the restart of the Rancho Seco nuclear power plant (see NRC 1987). Rancho Seco was said to have experienced several transients during which the normal cooldown rate was exceeded. The fatigue analysis examined the limiting fatigue usage factors in the reactor coolant piping, pressurizer, control rod drive mechanisms, and the reactor coolant pump casings. It was concluded that the allowable number of cooldowns should be reduced from 240 to 235. This represents a reduction of about 2% of the number of allowable cooldowns. The inverse ratio, which represents the reduction in the frequency of RCS pipe breaks, is a factor 1.02. This factor was multiplied by the adjusted-case values given above to estimate the base-case values of the affected parameters. These values are shown below:

<u>Oconee</u>	<u>Grand Gulf</u>
S_1 : Large LOCA = $0.000102/RY$	S : LOCA = $0.00143/RY$
S_2 : Medium LOCA = $0.00041/RY$	
S_3 : Small LOCA = $0.00133/RY$	

These values are applicable to the current plant lifetime case. Excess significant figures are included in these values for calculation purposes. The calculations were performed using the OCONEE and GRAND GULF computer codes.

For the extended plant lifetime case, the estimated number of cooldowns was extended to the mid-point of the remaining lifetimes, which was estimated to be an additional 15 operating years. This results in an additional 2 cooldowns, which is equivalent to about a 3% reduction in the allowable number of cooldowns. The corresponding reduction in RCS pipe break frequency is therefore a factor of about 1.03. The base-case values of the affected parameters for the extended plant lifetime case are therefore:

<u>Oconee</u>	<u>Grand Gulf</u>
S ₁ : Large LOCA = 0.000103/RY	S : LOCA = 0.00144/RY
S ₂ : Medium LOCA = 0.000412/RY	
S ₃ : Small LOCA = 0.00134/RY	

The failure probability of the reactor vessel as a result of thermal cycle fatigue was not included in base-case or adjusted-case values. This is because it is believed that the frequency of reactor vessel failure is so small relative to the frequencies of LOCA sequences that it would be insignificant. This conclusion is supported by data in the Reactor Safety Study (NRC 1975) and by the fact that it was not examined in the Rancho Seco restart SER (NRC 1987).

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Monitoring of Fatigue Transient Limits for Reactor Coolant System (78)

2. Affected Plants (N):

	<u>N</u>
PWRs	30
BWRs	16
All plants	46

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

	<u>$\bar{T}$ (yrs),</u>	
	<u>Operating Life Case</u>	
	<u>Current</u>	<u>Extended</u>
PWRs	27.7	42.7
BWRs	25.2	40.2

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

Current Plant Lifetime

$$\Delta(\bar{F}D_R)_{PWR} = (6.3E-7/RY(19860 \text{ person-rem})) = 1.25E-2 \text{ person-rem/RY}$$

$$\Delta(\bar{F}D_R)_{BWR} = (1.0E-7/RY(19860 \text{ person-rem})) = 2.01E-3 \text{ person-rem/RY}$$

Extended Plant Lifetime

$$\Delta(\bar{F}D_R)_{PWR} = (8.6E-7/RY(19860 \text{ person-rem})) = 1.70E-2 \text{ person-rem/RY}$$

$$\Delta(\bar{F}D_R)_{BWR} = (1.3E-7/RY(19860 \text{ person-rem})) = 2.68E-3 \text{ person-rem/RY}$$

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

	<u>Best Estimate</u>	<u>Error Bounds (person-rem)</u>	
	<u>(person-rem)</u>	<u>Upper</u>	<u>Lower</u>
Current Lifetime	1.2E+1	3.1E+3	0
Extended Lifetime	2.4E+1	4.8E+3	0

6-12 Steps Leading to Calculation of Total Occupational Doses for SIR Implementation and Operation/Maintenance:

No work in radiation zones is required for SIR implementation or operation/maintenance. SIR involves primarily monitoring and reporting requirements and perhaps evaluations of thermal cycle fatigue limits for the RCS. Therefore, radiation doses due to SIR implementation and operation/maintenance activities are zero.

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Monitoring of Fatigue Transient Limits in Reactor Coolant System (78)

2. Affected Plants (N):

	<u>N</u>
PWRs	30
BWRs	16
All Plants	46

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

	<u>$\bar{T}$ (yrs),</u>	
	<u>Operating Life Case</u>	
	<u>Current</u>	<u>Extended</u>
PWRs	27.7	42.7
BWRs	25.2	40.2

Industry Costs (Steps 4 through 12):

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

$$\Delta(\bar{F}A)_{PWR} = (\$1.65E+9/\text{event})(8.6E-7/\text{RY}) = \$1.4E+3/\text{RY}$$

$$\Delta(\bar{F}A)_{BWR} = (\$1.65E+9/\text{event})(1.3E-7/\text{RY}) = \$2.2E+2/\text{RY}$$

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

	<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
Current Lifetime	\$9.3E+5	\$2.6E+8	0
Extended Lifetime	\$2.0E+6	\$4.0E+8	0

6. Per-Plant Industry Resources for SIR Implementation:

SIR implementation includes preparation of an initial report and development of a proposed technical specification in response to the letter issued by the NRC. Koo (1983) estimates that approximately 20 man-hrs per plant are required to perform these activities. In addition, fatigue analyses may be necessary for plants that have experienced cooldowns in excess of design limits. This will be needed to determine the actual number of thermal cycles remaining before fatigue limits of Class

1 components are exceeded. The estimated resource requirements for this analysis are 2 man-yr/plant. It is estimated that half of the affected plants have experienced cooldowns in excess of limits and will perform the fatigue analysis.

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

Prepare response: (20 man-hr/plant)(\$56.75/man-hr) = \$1135/plant
 Fatigue analysis: (2 man-yr/plant)(\$100,000/man-yr) = \$200,000/plant

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

The following is applicable to both current and extended plant lifetime cases.

$$\begin{aligned} NI &= (\$1135/\text{plant})(46 \text{ plants}) + (0.5)(46 \text{ plants})(\$200,000/\text{plant}) \\ &= \$4.6\text{E}+6 \end{aligned}$$

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

SIR operation/maintenance is anticipated to involve weekly monitoring of fatigue transient limits. Koo (1987) estimated that approximately 15 minutes are required for this inspection and recording of information. The total annual resource requirements are therefore 13 man-hrs/Ry.

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

$$I_o = (13 \text{ man-hr/Ry})(\$56.75/\text{man-hr}) = \$738/\text{Ry}$$

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

Current Plant Lifetime

$$\begin{aligned} NTI_o &= (\$738/\text{Ry})\{(30 \text{ PWRs})(27.7\text{Ry}) + (16 \text{ BWRs})(25.2\text{Ry})\} \\ &= \$9.1\text{E}+5 \end{aligned}$$

Extended Plant Lifetime

$$\begin{aligned} NTI_o &= (\$738/\text{Ry})\{(30 \text{ PWRs})(42.7\text{Ry}) + (16 \text{ BWRs})(40.2\text{Ry})\} \\ &= \$1.4\text{E}+6 \end{aligned}$$

12. Total Industry Cost (S_t):

	<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
Current Lifetime	\$5.5E+6	\$7.9E+6	\$3.2E+6
Extended Lifetime	\$6.0E+6	\$8.4E+6	\$3.6E+6

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC development of SIR includes development of a model technical specification and preparation and approval of a letter to licensees. According to Sciacca (1989), approximately 3 man-wks of NRC labor are required for these activities, including technical staff as well as legal and management reviews. In addition, development costs include NRC contractor effort to review the responses received from the licensees and prepare a Technical Evaluation Report. Koo (1983) estimated the costs for the contractor review to be about \$84,000.

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

$$\begin{aligned} C_D &= (3 \text{ man-wk})(\$2270/\text{man-wk}) + \$84,000 \\ &= \$90,810 \end{aligned}$$

This value is applicable to both current and extended plant lifetime cases.

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

SIR implementation support includes resources necessary to 1) review and approve proposed license amendments and technical specifications and 2) prepare a safety evaluation. Sciacca (1989) estimates the resource requirements for these activities, including technical staff as well as legal and management reviews, at about 2.5 man-wks and 3.5 man-wks, respectively. The total labor requirement for SIR implementation is therefore 6 man-wks/plant.

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

$$C = (6 \text{ man-wk})(\$2270/\text{man-wk}) = \$13,620/\text{plant}$$

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

$$NC = (\$13,620/\text{plant})(46 \text{ plants}) = \$6.3E+5$$

This value is applicable to both current and extended plant lifetime cases.

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

NRC labor requirements for review of SIR operation and maintenance are estimated to be approximately the same as the industry operation and maintenance labor. This was estimated in Step 9 to be about 15 minutes/week or 13 man-hr/Ry.

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

$$C_o = (13 \text{ man-hr/Ry})(\$56.75/\text{man-hr}) = \$738/\text{Ry}$$

20. Total NRC Cost for Review of SIR Operation and/Maintenance ($\overline{N\overline{TC}}_o$):

Current Plant Lifetime Case

$$\begin{aligned}\overline{N\overline{TC}}_o &= (\$738/\text{RY})\{(30 \text{ PWRs})(27.7\text{RY}) + (16 \text{ BWRs})(25.2\text{RY})\} \\ &= \$9.1\text{E}+5\end{aligned}$$

Extended Plant Lifetime Case

$$\begin{aligned}\overline{N\overline{TC}}_o &= (\$738/\text{RY})\{(30 \text{ PWRs})(42.7\text{RY}) + (16 \text{ BWRs})(40.2\text{RY})\} \\ &= \$1.4\text{E}+6\end{aligned}$$

21. Total NRC Cost (S_N):

	<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
Current Lifetime	\$1.6E+6	\$2.2E+6	\$1.1E+6
Extended Lifetime	\$2.1E+6	\$2.9E+6	\$1.4E+6

REFERENCES (For Issue 78)

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. Nuclear Regulatory Commission (NRC). 1987. *Safety Evaluation Report Related to the Restart of Rancho Seco Generating Station Unit 1 Following the Event of December 26, 1985*. NUREG-1286. Washington D.C. WASH-1400, NUREG-75/014. Washington D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 81, Effect of Locked Doors and Barriers on Plant and Personnel Safety

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: This issue is related to the possibility that, in the event of an accident, locked doors and mechanical barriers may impede an operator's actions which could lead to recovery. The doors may become obstacles during abnormal or accident situations when operators may need prompt and unlimited access to safety equipment to assure proper plant operation. The proposed resolution involves an evaluation by each plant to identify specific vulnerabilities and appropriate procedural and hardware changes that might be required to prevent or minimize these vulnerabilities.

AFFECTED PLANTS: This issue affects all currently operating and planned PWRs and BWRs.

RESULTS: Five sensitivity cases with varying values for the probability of operator failure due to locked doors were evaluated. The results are:

CASE NUMBER	1	2	3	4	5
LOCKED DOOR PROBABILITY	0.0001	0.0005	0.001	0.01	0.1
RISK/DOSE RESULTS (man-rem):					
PUBLIC RISK REDUCTION	5.2E+2	2.6E+3	5.2E+3	5.2E+4	5.2E+5
OCCUPATIONAL DOSES:					
SIR Implementation	0	0	0	0	0
SIR Operation/Maintenance	0	0	0	0	0
Total of Above	0	0	0	0	0
Accident Avoidance	3.5E+0	1.7E+1	3.4E+1	3.4E+2	3.4E+3
COST RESULTS (\$10E+6)					
INDUSTRY COSTS:					
SIR Implementation	2.7E+1	2.7E+1	2.7E+1	2.7E+1	2.7E+1
SIR Operation/Maintenance	1.7E+2	1.7E+2	1.7E+2	1.7E+2	1.7E+2
Total of Above =	2.0E+2	2.0E+2	2.0E+2	2.0E+2	2.0E+2
Accident Avoidance =	2.9E-1	1.4E+0	2.8E+0	2.8E+1	2.9E+2
NRC COSTS:					
SIR Development =	0	0	0	0	0
SIR Implementation =	2.7E+1	2.7E+1	2.7E+1	2.7E+1	2.7E+1
SIR Operation/Maintenance	0	0	0	0	0
Total of Above =	2.7E+1	2.7E+1	2.7E+1	2.7E+1	2.7E+1

EFFECT OF LOCKED DOORS AND BARRIERS ON PLANT AND PERSONNEL SAFETY

ISSUE 81

1.0 SAFETY ISSUE DESCRIPTION

This issue is related to the possibility that, in the event of an accident, locked doors and mechanical barriers may impede an operator's actions which lead could to recovery. These doors and barriers are needed for fire protection, radiation protection, flood protection, and administrative controls. However, they may become obstacles during abnormal or accident situations when operators may need prompt and unlimited access to safety equipment to assure proper plant operation. This issue was previously prioritized with respect to failure of locked doors with mechanical locks. Additional background information on the previous analysis may be found in NUREG-0933 (NRC 1983) and is not included here. It was decided to reevaluate this issue in light of an event that occurred at Surry in which a pipe break event sprayed water onto a card reader, resulting in failure of the card reader system, and subsequent locking of all card-controlled doors.

PROPOSED RESOLUTION

The proposed resolution to this issue consists of a one-time evaluation of each plants's locked doors and barriers that might be needed in abnormal and accident conditions. The purpose of this evaluation would be to identify specific vulnerabilities, such as the Surry event, and to identify appropriate procedural and hardware changes that could be implemented to prevent or minimize these vulnerabilities. The evaluation is intended to ensure that operator access is unimpeded during emergency, abnormal, or accident conditions and that prompt operator action is possible under these conditions.

SIR is postulated to be resolved through issuance of mechanical keys to operations personnel as a backup in the event that a locked door is encountered during an accident. SIR will also involve cross-training of security and operations personnel to ensure that the security force is aware of potential operator actions in abnormal and accident conditions and to instruct operators in applicable security provisions regarding locked doors and barriers. In addition, the utilities are also anticipated to initiate a program to ensure that the impacts of safeguards provisions on plant safety are appropriately identified and minimized.

AFFECTED PLANTS

This safety issue affects all plants, including PWRs and BWRs. Plants operating and under construction are affected.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this issue are estimated in this section and summarized in Tables 1 and 2. Attachment 1 describes the basis for the risk and dose changes that are estimated in the Tables.

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

Effect of Locked Doors and Barriers on Plant and Personnel Safety, 81

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

All PWRs and BWRs are assumed to be affected.

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

3. Plants Selected for Analysis:

Representative PWR: Oconee-3

Representative BWR: Grand Gulf-1

4. Parameters Affected by SIR:

The parameters affected by SIR include the operator error events in the Oconee and Grand Gulf PRAs described by Andrews, et al., (1983). These parameters are presented in Attachment 1.

5. Base-Case Values for Affected Parameters:

A parametric study was performed to model the effects of SIR on plant risk. Locked door failures were not addressed in the Oconee and Grand Gulf PRAs. The base-case represents the case in which the issue has not been resolved. Therefore, as discussed in Attachment 1, the locked door probabilities were added to the original values of the affected parameters to obtain the base-case values. These parameters and their values were derived in Attachment 1 for five parametric cases in which the locked door probabilities were 0.0001, 0.0005, 0.001, 0.01, and 0.1. The smaller values contribute small fractions to the operator error probabilities given in the Grand Gulf and Oconee PRAs and the larger values become dominant contributors. The values given above are a representative range of possible locked door failure probabilities.

6. Affected Accident Sequences and Base-Case Frequencies:

The OCONEE and GRAND GULF PRA computer codes were used to perform the calculations in support of this analysis. The resulting base-case and adjusted core melt frequencies are presented in Attachment 1. A review of the Oconee and Grand Gulf PRAs in Andrews, et al., (1983) indicates that

nearly every accident sequence is affected by resolution of this issue; i.e., a dominant cut set containing one or more of the parameters listed in Attachment 1 appears in nearly all of the Grand Gulf and Oconee accident sequences. Because the calculations were performed using the OCONEE and GRAND GULF computer codes, the accident sequences and base-case frequencies will not be shown here.

7. Affected Release Categories and Base-Case Frequencies:

All of the release categories are affected by this issue.

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

Five separate cases for each plant type were evaluated in this analysis. The base-case core melt frequencies calculated by the computer codes are as follows:

Case	Locked Door Probability	Base Case Core Melt Freq. RY^{-1}	
		PWR	BWR
1	0.0001	1.411E-05	2.482E-05
2	0.0005	1.425E-05	2.509E-05
3	0.001	1.442E-05	2.543E-05
4	0.01	1.754E-05	3.149E-05
5	0.1	4.968E-05	9.216E-05

NOTE: Excess significant figures are included for calculation purposes.

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

The base-case public risk values calculated by the computer codes are as follows:

Case	Locked Door Probability	Base Case Public Risk, person-rem/R $\bar{Y}$	
		PWR	BWR
1	0.0001	3.767E+01	1.722E+02
2	0.0005	3.804E+01	1.731E+02
3	0.001	3.850E+01	1.743E+02
4	0.01	4.682E+01	1.951E+02
5	0.1	1.326E+02	4.033E+02

NOTE: Excess significant figures are included for calculation purposes.

10. Adjusted-Case Values for Affected Parameters:

For this issue analysis, the adjusted-case values for the affected parameters are equal to the original values given in the Oconee and Grand Gulf PRAs. These values are provided in Attachment 1.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

As discussed in Step 6, nearly all of the accident sequences in the Oconee and Grand Gulf PRAs are affected by this issue. Because the calculations were performed using the OCONEE and GRAND GULF computer codes, the accident sequences and adjusted-case frequencies will not be shown here.

12. Affected Release Categories and Adjusted-Case Frequencies:

All of the release categories were affected by this issue.

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

The adjusted-case affected core melt frequency is equal to the affected portion of the original Oconee and Grand Gulf core-melt frequencies. These are, as determined by the computer codes:

$$\bar{F}^*_{PWR} = 1.408E-05/RY \qquad \bar{F}^*_{BWR} = 2.475E-05/RY$$

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

$$W^*_{PWR} = 3.758E+01 \text{ person-rem/RY} \qquad W^*_{BWR} = 1.719E+02 \text{ person-rem/RY}$$

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

The reduction in core-melt frequency for the five cases for each plant type are shown below.

Case	Locked Door Probability	Change in Core Melt Freq. (RY ⁻¹)	
		PWR	BWR
1	0.0001	3.500E-08	7.000E-08
2	0.0005	1.730E-07	3.400E-07
3	0.001	3.456E-07	6.770E-07
4	0.01	3.462E-06	6.744E-06
5	0.1	3.560E-05	6.741E-05

NOTE: Excess significant figures are included for calculation purposes.

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

Case	Locked Door Probability	Change in Risk, person-rem/RY	
		PWR	BWR
1	0.0001	9.200E-02	2.300E-01
2	0.0005	4.608E-01	1.158E+00
3	0.001	9.216E-01	2.315E+00
4	0.01	9.241E+00	2.314E+01
5	0.1	9.503E+01	2.314E+02

NOTE: Excess significant figures are included for calculation purposes.

17. Total Public Risk Reduction, (ΔW)_{Total}:

Case	Locked Door Probability	Best Estimate (person-rem)	Error Bounds (person-rem)	
			Upper	Lower
1	0.0001	5.2E+2	9.2E+6	0
2	0.0005	2.6E+3	9.2E+6	0
3	0.001	5.2E+3	9.3E+6	0
4	0.01	5.2E+4	1.1E+7	0
5	0.1	5.2E+5	2.5E+7	0

ATTACHMENT 1

This issue is related to the possibility that, in the event of an accident, locked doors and mechanical barriers may impede an operator actions which could to recovery. Therefore, this issue affects operator error-related parameters in the Grand Gulf and Oconee PRAs (see Andrews, et al., 1983).

The effects of locked doors on operator actions were not addressed in the PRAs described by Andrews, et al., (1983). The probabilities of these failures are also not well known. As a result, a parametric sensitivity analysis approach was taken to quantify the effects of this issue on plant risks. Plant risk reduction estimates were developed for five cases which represented the range of possible failure probabilities resulting from locked doors.

First-order approximations of the probability that a locked door will impede an operator was estimated to be about 0.0005 or less. This includes consideration that: (1) a card reader fails; (2) doors are failed in a locked position; (3) no mechanical key override exists; and (4) operators fail to break down the door within the required time. Obviously, operator actions and the associated probabilities vary widely. The parametric studies were conducted to examine this possible wide range in the probabilities of locked doors. Five separate parametric studies were performed for both plant types for a total of ten cases. The locked door probabilities used in these cases were 0.0001, 0.0005, 0.001, 0.01, and 0.1. This represents a range whose contribution to operator error probabilities is from small fractions to as much as a dominant contributor.

The Oconee and Grand Gulf PRAs in Andrews, et al., (1983) were examined to identify the affected operator errors and their associated probabilities. These parameters and their values are shown below. The parameters described above represents the adjusted-case because these particular types of failures were not addressed in the Grand Gulf and Oconee PRAs. Therefore, they represent the case in which the issue has been resolved and these types of failures have been addressed at the plants.

The list shown below does not include the events that have operator error contributions to basic plant failures, such as test/maintenance faults, calibration errors, and valve misalignments. These errors are not affected by this issue. They are included, indirectly, in the probabilities that operators can restore components that are failed as a result of, for example, improper valve alignment. They are included because locked doors may impede an operator's ability to restore these components.

The base-case was represented by the range of locked door probabilities discussed above. The base-case values for these parameters were determined by including the locked door probabilities as a contributory failure. Most of the events are remote manual actuation of specific safety equipment or systems. In these cases (HPMAN, HHMAN, and OP), it was assumed that, should remote manual actuation fail, an operator would be sent to a location outside the control room to implement local manual actuation of the component, in spite of the fact that this may not be possible for equipment located inside containment. The other two parameters, RECOVERY and RECOVERY1 were simply modified to account for the locked door probabilities. In both situations, the total probability becomes the unmodified probabilities given in the PRAs plus the probability of failure due

to locked doors. The base-case and adjusted-case values of the affected parameters are given below.

Parameter	Event Description	Value
HHMAN HPMAN HPMAN1	OCONEE Operator fails to start high-head auxiliary service water system (remote manual)	0.1
	Operator fails to start high-pressure injection system (remote manual)	0.015
	Operator fails to start high-pressure injection system during ATWS; high stress (remote manual)	0.1
OP	GRAND GULF Failure of operator to manually initiate the Automatic Depressurization System	0.0015
RECOVERY	Failure to restore test/maintenance faults within 28 hr	0.23
RECOVERY1	Failure to restore test/maintenance faults within 30 hr	0.21

Parameter	Adjusted Case Values	Parametric Base-Case Values				
		1	2	3	4	5
OCONEE						
HHMAN	0.1	0.1001	0.1005	0.101	0.11	0.2
HPMAN	0.015	0.0151	0.0155	0.016	0.025	0.115
HPMAN1	0.1	0.1001	0.1005	0.101	0.11	0.2
GRAND GULF						
OP	0.0015	0.0016	0.002	0.0025	0.0115	0.1015
RECOVERY	0.23	0.2301	0.2305	0.231	0.24	0.33
RECOVERY1	0.21	0.2101	0.2105	0.211	0.22	0.31

The Grand Gulf and Oconee PRA computer codes were used to calculate the core-melt frequency and public risk reduction for each of these cases. The results are as follows:

Case	Plant Type	Affected Core-Melt Frequency, per RY			Affected Per-Plant Risk, Person-rem/RY		
		F*	F	ΔF	W*	W	ΔW
1	PWR	1.411E-05	1.408E-05	3.500E-08	3.767E+01	3.758E+01	9.200E-02
	BWR	2.482E-05	2.475E-05	7.000E-08	1.722E+02	1.719E+02	2.300E-01
2	PWR	1.425E-05	1.408E-05	1.730E-07	3.804E+01	3.758E+01	4.608E-01
	BWR	2.509E-05	2.475E-05	3.400E-07	1.731E+02	1.719E+02	1.158E+00
3	PWR	1.442E-05	1.408E-05	3.456E-07	3.850E+01	3.758E+01	9.216E-01
	BWR	2.543E-05	2.475E-05	6.770E-07	1.743E+02	1.719E+02	2.315E+00
4	PWR	1.754E-05	1.408E-05	3.462E-06	4.682E+01	3.758E+01	9.241E+00
	BWR	3.149E-05	2.475E-05	6.744E-06	1.951E+02	1.719E+02	2.314E+01
5	PWR	4.968E-05	1.408E-05	3.560E-05	1.326E+02	3.758E+01	9.503E+01
	BWR	9.216E-05	2.475E-05	6.741E-05	4.033E+02	1.719E+02	2.314E+02

NOTE: Excess significant figures are shown for calculation purposes. Abbreviations are as follows:

F = Base-case affected core-melt frequency;
 F* = Adjusted-case affected core melt frequency
 ΔF = Reduction in core-melt frequency associated with issue resolution
 W = Base-case affected public risk
 W* = Adjusted-case affected public risk
 ΔW = Reduction in public risk associated with issue resolution

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Effect of Locked Doors and Barriers on Plant and Personnel Safety, 81

2. Affected Plants (N):

	<u>N</u>
PWRs	90
BWRs	44
All plants	134

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

	<u>$\bar{T}$</u>
PWRs	28.8 yr
BWRs	27.4 yr
All plants	28.3 yr

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

Example:

Case 1 - PWR (19860 person-rem)(3.5E-8/Ry) = 6.95E-4 person-rem/Ry

The remaining per-plant occupational dose savings are as follows:

<u>Case</u>	<u>$\Delta\bar{F}D_R$ (person-rem/Ry)</u>	
	<u>PWR</u>	<u>BWR</u>
1	6.95E-4	1.39E-3
2	3.44E-3	6.75E-3
3	6.87E-3	1.35E-2
4	6.88E-2	1.34E-1
5	7.07E-1	1.34E+0

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

Case	Locked Door Probability	Best Estimate (person-rem)	Error Bounds (person-rem)	
			Upper	Lower
1	0.0001	3.5E+0	7.9E+3	0
2	0.0005	1.7E+1	8.0E+3	0
3	0.001	3.4E+1	8.1E+3	0
4	0.01	3.4E+2	9.9E+3	0
5	0.1	3.4E+3	2.9E+4	0

6-11 Steps Related to Per-Plant Utility Labor in Radiation Zones for SIR Operation and Maintenance:

SIR is conservatively assumed to involve no work in radiation zones. A substantial amount of the effort necessary to comply with the proposed SIR involves reviews of plant design drawings, equipment specifications, assessment of operator actions in abnormal and accident conditions, and identification of possible barriers resulting from locked doors. It is anticipated that plants may be able reduce these potential vulnerabilities by issuing mechanical keys that are capable of opening doors in addition to the cards that will be used if card-reader systems are functioning properly. As a result, occupational doses due to SIR implementation, operation, and maintenance are estimated to be 0.

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
0	0	0

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Effect of Locked Doors and Barriers on Plant and Personnel Safety, 81

2. Affected Plants (N):

	<u>N</u>
PWRs	90
BWRs	44
All Plants	134

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

	<u>$\bar{T}$</u>
PWRs	28.8 yr
BWRs	27.4 yr
All plants	28.3 yr

Industry Costs (Steps 4 through 12):

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

Example:

$$\text{Case 1 - PWR } (\$1.65\text{E}+9)(3.5\text{E}-8/\text{RY}) = \$ 5.78\text{E}+1/\text{RY}$$

The remaining per-plant cost savings are as follows:

<u>Case</u>	<u>$\Delta\bar{F}A$ (\$/RY)</u>	
	<u>PWR</u>	<u>BWR</u>
1	5.78E+1	1.16E+2
2	2.85E+2	5.61E+2
3	5.70E+2	1.18E+3
4	5.71E+3	1.11E+4
5	5.87E+4	1.11E+5

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

Case	Locked Door Probability	Best Estimate (person-rem)	Error Bounds (person-rem)	
			Upper	Lower
1	0.0001	2.9E+5	6.6E+8	0
2	0.0005	1.4E+6	6.6E+8	0
3	0.001	2.8E+6	6.7E+8	0
4	0.01	2.8E+7	8.3E+8	0
5	0.1	2.9E+8	2.4E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

SIR implementation resources and costs were taken directly from NUREG-0933 (NRC 1983). Industry implementation resources were estimated at \$200,000/plant for a one time evaluation of existing plant locked doors and barriers.

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

The per-plant industry cost for SIR implementation is \$200,000/plant.

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

$$NI = (134 \text{ plants})(\$200,000/\text{plant}) = \$ 2.7E+7$$

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

Per-plant operation and maintenance costs are estimated directly in Step 10.

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

The plant evaluations to be performed are anticipated to identify potential vulnerabilities that need to be corrected. This cost category is to implement changes to hardware, procedures, or administrative controls to reduce these vulnerabilities. The adverse safety findings are postulated to be resolved through issuance and maintenance of mechanical door keys to control room operators. NRC (1983) assumed that a total of 25 keys will be needed for the rest of the reactor lifetimes. The costs for these 25 keys were estimated to be about \$21,875/RY.

SIR will also involve cross-training of security and operations personnel to ensure that the security force is aware of potential operator actions in abnormal and accident conditions and to instruct operators in applicable security provisions regarding locked doors and barriers. This

training was estimated to require 1 day/yr per plant for a total of 50 operators and security staff. The total annual costs were estimated to be \$13,972/RY.

The final aspect of SIR involves a program to ensure that safeguards impacts on plant safety and, in particular, operator actions under accident conditions, are appropriately identified. The program is also anticipated to include development of provisions, procedures, etc. to ensure that these effects on plant safety are minimized. The costs for this program were estimated to be \$ 10,000/RY (NRC 1983).

The total costs for SIR operations and maintenance are therefore \$45,847 per RY.

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

$$\overline{NTI}_0 = (\$45,847/\text{RY})(134 \text{ plants})(28.3 \text{ RY/plant}) = \$ 1.7\text{E}+8$$

12. Total Industry Cost (S_i):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$ 2.0E+8	\$ 2.9E+8	\$ 1.1E+8

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

The analyses in NUREG-0933 (NRC 1983) did not specifically address NRC development costs. Therefore, these costs are assumed to be zero for this analysis. Development costs are included in NRC costs for support of SIR implementation.

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

$$C_D = 0$$

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

NRC costs for SIR implementation support are estimated directly in Step 16.

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

NRC costs for support of SIR implementation were estimated in NRC (1983) to be about \$200,000 per plant for NRC reviews of plant modifications.

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

$$NC = (\$200,000/\text{plant})(134 \text{ plants}) = \$ 2.7\text{E}+7$$

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

The analyses in NRC (1983) included no costs for review of SIR operation and maintenance. Therefore, these costs are assumed to be zero.

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

$$C_o = 0$$

20. Total NRC Cost for Review of SIR Operation and/Maintenance ($\overline{N\overline{C}}_o$):

$$\overline{N\overline{C}}_o = 0$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$ 2.7E+7	\$ 4.0E+7	\$ 1.3E+7

REFERENCES (For Issue 144)

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. Nuclear Regulatory Commission (NRC). 1983. *A Prioritization of Generic Safety Issues*. NUREG-0933. Washington D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 89, Stiff Pipe Clamps

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: This issue addresses potential design deficiencies related to pipe clamps in Class 1 piping systems of nuclear power plants. Use of specialized "stiff" pipe clamps without proper evaluation may result in unsafe plant operation. These stiff pipe clamps could potentially induce loadings on piping systems which may result in significant localized stresses that should be considered in the piping design. Resolution of this issue involves piping and pipe-to-clamp interface evaluations that account for the localized loadings induced by stiff pipe clamps. At plants in which these re-analyses indicate the pipe clamp loadings result in piping system loads in excess of specified criteria, installation of new pipe clamps will be required.

AFFECTED PLANTS: This issue affects a total of 17 BWRs and 27 PWRs (includes plants that are under construction and operating plants).

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 2.5E+3

OCCUPATIONAL DOSES:

SIR Implementation = 5.3E+4

SIR Operation/Maintenance = 2.8E+2

Total of Above = 5.3E+4

Accident Avoidance = 2.1E+1

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation = 92

SIR Operation/Maintenance = 0.63

Total of Above = 92

Accident Avoidance = 1.7

NRC COSTS:

SIR Development = 0.55

SIR Implementation = 0.53

SIR Operation/Maintenance Review = 0.32

Total of Above = 1.4

STIFF PIPE CLAMPS

ISSUE 89

1.0 SAFETY ISSUE DESCRIPTION

This issue deals with the possibility that some pipe clamps that have been installed in nuclear power plant piping systems may not have been properly evaluated and may result in unsafe operation of the plant. The issue arose as a result of allegations of improper evaluations of pipe clamps. The NRC issued IE Information Notice 83-30 (NRC 1983) as a result of these allegations.

The NRC staff performed an evaluation of the allegations concerning Class 1 piping (Vollmer 1983). The NRC staff determined that certain loadings induced by specialized "stiff" pipe clamps can result in significant localized stresses in the piping that have not been previously considered in piping designs. Vollmer (1983) found that piping designers often assumed that the clamp effects on the piping systems were negligible and did not warrant explicit consideration. This assumption was stated to be acceptable for most clamp applications; however, for some applications, certain piping system conditions coupled with specific pipe clamp design requirements could result in interaction effects that should be examined (Vollmer 1983).

Stiff pipe clamps have been in use for only the last several years. Their origin derives from requirements for piping systems to withstand dynamic loads, including safety-relief valve (SRV) discharges to the suppression pool and loss-of-coolant accident (LOCA) induced loads, combined with seismic loadings. These requirements are found in 10 CFR 50, Appendix A, General Design Criterion 2. These large loadings, which may be 100,000 lbs. or more, resulted in the need to design pipe supports to accommodate these large loadings. Pipe clamps must also be designed to withstand loads equivalent to those of the pipe supports. It was determined that simply increasing the thickness of existing pipe clamps would not be sufficient to achieve the stiffness values required by piping support design specifications. Therefore, some manufacturers required large preloading of the clamp to achieve the desired stiffness. Preloading the clamp imposes a constant compressive load on the piping and is necessary to prevent the clamp from lifting off the pipe under dynamic loading conditions. This preloading may result in degraded pressure-retaining capabilities of the piping system under dynamic loading conditions. Since clamp-induced stresses on piping systems had not been included in piping system analyses, it is not known if these stresses are significant.

In addition to the large preloading of the clamp, four additional new design features have evolved, including:

- use of high-strength or non-ASME approved materials
- local surface contact on the pipe
- uncommonly-thick and/or wide design
- clamp application on piping components other than straight pipe (e.g., on pipe elbows).

Vollmer (1983) stated that these new design features represented a significant change from conventional pipe clamps and thus should be supported by detailed analyses, particularly of the pipe to clamp interaction and interface design requirements.

These relatively new design features may result in localized piping stresses that are higher than those induced by conventional pipe clamps. Piping designers in the past neglected these localized loadings because of the low stresses induced by conventional clamps. It may not be appropriate to neglect the clamp-induced stresses for the "stiff" pipe clamps for the reasons discussed above. If this issue results in an overestimate of the pressure-retaining capabilities of piping systems, the probabilities of pipe breaks that result in or after LOCAs or reactor transients may be higher than previously estimated. This could potentially result in increases in public risks.

PROPOSED RESOLUTION

The resolution of this issue involves qualifying the clamp-induced local pipe stresses caused by stiff pipe clamps. For plants that are provided with stiff pipe clamps, analyses of the clamp-induced local stresses under dynamic conditions will be required. Such analyses were typically not performed explicitly in the past, but rather were assumed to be negligible as is appropriate for conventional pipe clamps. The safety issue resolution will require an explicit evaluation of the localized stresses for those plants provided with stiff pipe clamps. The performance criteria for these analyses are the existing ASME Code requirements and the Section 3.9.3 of the Standard Review Plan (NRC 1981). In the future, a request will be submitted to ASME to recommend that Section III of the Code be revised to include:

- procedures for the categorization of pipe stresses due to clamp-induced loads, and
- procedures for evaluating those clamp applications where the ASME code stress indices and flexibility factors are not applicable (Vollmer 1983).

The results of these analyses could result in confirmation that the existing piping systems which are provided with stiff pipe clamps meet the appropriate requirements and are acceptable or could indicate that the clamp-induced stresses result in a condition in which the piping system may not be capable of withstanding the prescribed loads. With regard to the latter, significant hardware fixes (i.e., replace the stiff pipe clamps with different designs) may be required to resolve the issue.

To make future pipe-to-clamp analyses less burdensome, Vollmer (1983) recommended that the NRC staff initiate a technical assistance program to experimentally and analytically evaluate the effects between pipes and pipe clamps. This program should confirm the appropriateness of the various analytical methods used to calculate clamp-induced stresses by comparing with actual test results and provide a simplified method to facilitate evaluations of clamp-induced stresses.

AFFECTED PLANTS

The resolution of this issue is applicable to plants presumed to have installed stiff pipe clamps on their piping systems. Based on Vollmer (1983), LaSalle 1 and 2, Quad Cities 1 and 2, Dresden 2 and 3, and all plants whose operation began in 1983 or later are assumed to have installed stiff pipe clamps. Based on the most recent PNL updates, a total of 27 operating PWRs and 17 operating BWRs are therefore presumed to be affected. Their respective average remaining lifetimes are 33.4 and 28.9 yr.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this issue are estimated in this section and summarized in Tables 1 and 2. Attachment 1 describes the basis for the risk and dose changes that are estimated in the Tables.

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

Stiff Pipe Clamps, 89

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

	<u>N</u>	<u>$\bar{T}$ (yr)</u>
PWR	27	33.4
BWR	17	28.9

3. Plants Selected for Analysis:

Generic PWR and BWR plants are used in this analysis.

4. Parameters Affected by SIR:

See Attachment 1.

5. Base-Case Values for Affected Parameters:

The change in core-melt frequencies and public risks associated with SIR are calculated directly. See Attachment 1 for a detailed explanation of the approach used to calculate the changes in these parameters.

6. Affected Accident Sequences and Base-Case Frequencies:

The change in core-melt frequencies and public risks associated with SIR are calculated directly (See Attachment 1).

7. Affected Release Categories and Base-Case Frequencies:

The change in core-melt frequencies and public risks associated with SIR are calculated directly (See Attachment 1).

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

	<u>Type 1</u>	<u>Type 2</u>	<u>Total</u>
PWR	6.1E-8/RY	5.2E-7/RY	5.8E-7/RY
BWR	7.9E-7/RY	2.6E-7/RY	1.05E-6/RY

Since SIR presumably eliminates the vulnerability to stiff pipe clamp related failures, the reductions in core melt frequency estimated in Attachment 1 may be taken as the base-case values (See Attachment 1).

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

	<u>Base-case Risk, person-rem/RY</u>		
	<u>Type 1</u>	<u>Type 2</u>	<u>Total</u>
PWR	0.30	0.57	0.87
BWR	2.1	1.4	3.5

Since SIR presumably eliminates the vulnerability to stiff pipe clamp related failures, the reductions in public risk estimated in Attachment 1 may be taken as the base-case values (See Attachment 1).

10. Adjusted-Case Values for Affected Parameters:

The change in core-melt frequencies and public risks associated with SIR are calculated directly (See Attachment 1).

11. Affected Accident Sequences and Adjusted-Case Frequencies:

The change in core-melt frequencies and public risks associated with SIR are calculated directly (See Attachment 1).

12. Affected Release Categories and Adjusted-Case Frequencies:

The change in core-melt frequencies and public risks associated with SIR are calculated directly (See Attachment 1).

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

Since SIR presumably eliminates the failure potential, the adjusted-case values may be taken as zero.

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

Since SIR presumably eliminates the failure potential, the adjusted-case values may be taken as zero.

15. Reduction in Core-Melt Frequency (ΔF):

	<u>Type 1</u>	<u>Type 2</u>	<u>Total</u>
PWR	6.1E-8/RY	5.2E-7/RY	5.8E-7/RY
BWR	7.9E-7/RY	2.6E-7/RY	1.05E-6/RY

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

	<u>Risk Reduction, person-rem/RY</u>		
	<u>Type 1</u>	<u>Type 2</u>	<u>Total</u>
PWR	0.30	0.57	0.87
BWR	2.1	1.4	3.5

17. Total Public Risk Reduction, (ΔW)_{Total}:

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
2.5E+3	7.5E+4	0

ATTACHMENT 1 (To Table 1)

The overall effects of SIR on plant safety will be to reduce the risks associated with failures of nuclear power plant piping systems. These will include risks associated with pipe breaks which can be divided into the following two types:

1. seismic-induced pipe breaks resulting in LOCAs and/or reactor transients
2. pipe breaks in Class 1 piping systems resulting from dynamic loads following LOCAs and reactor transients.

The analysis for Type 1 risk will focus on seismic-induced failures of piping systems. For the second type, risks from pipe breaks in Class 1 piping systems are assumed to arise from the dynamic loads induced following LOCAs and transients. Estimation of this risk will focus on all accident sequences which require functioning of the Class 1 piping systems, essentially all accident sequences. The stress induced by stiff pipe clamps creates a failure potential among all Class 1 piping that is clamped in this manner. An estimate of the failure potential as a percentage of core-melt frequency is needed to quantify the Type 2 risk.

Type 1

The principal source of quantitative risk information for this analysis was a study which was performed to identify risk-sensitive components in nuclear power plants during and after a seismic event (Azarm, Boccio, and Farahzad 1983). This study expanded risk-sensitivity analyses by accounting for seismicity and component fragility data that were taken from existing nuclear power plant probabilistic risk assessments (PRAs) and seismic hazard analyses. The existing quantitative seismic risk data were varied and the effects on plant core melt frequencies and public risks were then calculated. In addition, the effects of equipment upgrades were evaluated, including the effects of upgrading various piping systems (e.g., buried pipes from the Condensate Storage Tanks, Main Steam Lines inside containment, low pressure coolant injection (LPCI) piping). This was accomplished by increasing/decreasing the median peak ground acceleration (defined as the level of peak ground acceleration at which a component has a 50% probability of failure) by a factor of five (5). The effects of these increases/decreases in component failure probabilities were then calculated in terms of the reductions/increases in core melt frequency and public risk.

For this analysis, it is assumed that SIR results in a factor of 5 increase in the median peak ground acceleration for piping systems equivalent to that used by Azarm, Boccio, and Farahzad (1983). The effects on PWR and BWR core-melt frequencies that result from these decreases in piping system failure probabilities are summarized in the table below.

The base-case affected core-melt frequencies for PWRs and BWRs, also taken from Azarm, Boccio, and Farahzad (1983) are $5.20\text{E-}6/\text{RY}$ and $9.1\text{E-}5/\text{RY}$, respectively. Other plant components were examined by Azarm, Boccio, and Farahzad (1983), including building structures and various pumps, valves, and electrical distribution components but these are not the subject of the present study and are neglected from further consideration.

Component	% Reduction in Core-Melt Frequency, RY ⁻¹
PWR	
Buried pipes from condensate storage tank	3.8
Pipes (all sizes, including buried pipes)	4.2
	TOTAL 8.0
BWR	
Main Steam Line break inside containment	1.7
LPCI piping	1.3
Core spray piping	2.1
Feedwater line break	0.1
Recirculation line break	0.1
High pressure Coolant Injection piping	0.7
	TOTAL 6.0

Assuming that the increases in piping system reliability and reductions in core melt frequency shown above are additive, improvements to all of these piping systems could potentially result in seismic-induced core melt frequency reductions equivalent to the totals shown above. This is conservative in that this assumption is only technically valid if two or more of the listed piping system failures do not appear in the same cut sets. Given that this is not a true statement because the same cut sets involve more than one of these piping system failures, the total core melt frequency reduction is less than the totals shown above and therefore the results in this analysis are conservative.

The change in piping system reliability that results from SIR is less than the factor of 5 that was used by Azarm, Boccio, and Farahzad (1983). This is because the pipe clamps are only one component of the piping system and there are other components whose failures contribute to the piping system failure probability (e.g., welds, elbows, branch connections, snubbers, etc.). Therefore, a factor was developed to model the portion of the pipe system reliability improvement that is due to improvements to the pipe clamps. This factor was assumed to be represented by fractional difference between the upper and median bending moment capacity of a reference pipe segment as described by Kennedy, et al., (1982). The moment capacities, which accounted for strain hardening effects and low probabilities of large pre-existing flaws, represent to some degree the stiffness (i.e., resistance to bending) that is provided by the stiff pipe clamps. The derived upper bound and median moment capacities given by Kennedy, et al., (1982) for a reference pipe segment were 1.65E+6 in-lbs and 1.41E+6 in-lbs, respectively. The factor that represents the ratio between pipe clamp failure probabilities before and after SIR is therefore $1 - (1.41E+6 / 1.65E+6)$ or 0.145.

The change in core-melt frequency (ΔF) that results from SIR is the product of the base-case core melt frequency, the change in core-melt frequency that results from the factor of 5 increase in median peak ground acceleration for piping system components, and the factor that represents the ratio between pipe

clamp failure probabilities before and after SIR. These calculations are shown below:

$$\Delta F_{PWR}: (5.20E-6/RY)(0.08)(0.145) = 6.1E-8/RY$$

$$\Delta F_{BWR}: (9.10E-5/RY)(0.06)(0.145) = 7.9E-7/RY$$

The public risk reduction associated with SIR is the product of the change in core-melt frequency and public dose consequences. Because the core damage sequences encompass all release categories, a weighted average consequence value was developed to approximate the consequences of the core melt accidents. Azarm, Boccio, and Farahzad (1983) presented the relative contributions of the seismic-induced core melt frequencies to the PWR and BWR release categories from the *Reactor Safety Study* (NRC 1975). The weighted average consequence value was calculated by multiplying the relative contributions to each release category by the dose consequences for each release category that are presented in Appendix D of Andrews et al. (1983) and then summing over all the release categories. The resulting weighted consequence values were 4.9E+6 person-rem for PWRs and 2.7E+6 person-rem for BWRs. The per-plant risk reduction estimates (ΔW) were then calculated as follows:

$$\begin{aligned}\Delta W_{PWR} &= (6.1E-8/RY)(4.9E+6 \text{ person-rem}) = 3.0E-1 \text{ person-rem/RY} \\ \Delta W_{BWR} &= (7.9E-7/RY)(2.7E+6 \text{ person-rem}) = 2.1E+0 \text{ person-rem/RY}\end{aligned}$$

Type 2

Andrews, et al., (1983) served as the principal source of quantitative risk information for the analysis of Type 2 risk, specifically the Grand Gulf and Oconee accident sequences and minimal cut sets of Appendices A and B. First, the cut set elements in Tables A.4 and B.4 were reviewed to identify those containing a "hardware" failure in an ECCS (typically associated with a valve or pump). For each element so identified, the largest hardware failure probability was identified, and its percentage contribution to the element's total failure probability was calculated.

For example, cut set element LA2 [from Table B.4 in Andrews, et al., (1983)] refers to a failure in the low pressure coolant injection system at Grand Gulf. Its failure probability is the sum of the failure probabilities from a check valve, a normally-open motor-operated valve, a normally-open manual valve, and a pump. The check valve has a hardware failure probability of 0.0001 while the pump has a hardware failure probability of .001, ten times larger. This latter value of 0.001 was identified as the largest hardware failure probability for element LA2, and its contribution to the element's total failure probability (0.014) was estimated as $0.001/0.014 = 0.071$, or 7.1%.

These percentage contributions were calculated for all the identified elements at Grand Gulf and Oconee. These were averaged to yield values of 6.1% for Grand Gulf and 6.0% for Oconee. These values were assumed to represent the failure contribution to the core-melt frequency resulting from pipe breaks in ECCSs arising from dynamic loads induced following LOCAs and transients, essentially a surrogate measure for use in estimating the risk contribution from Type 2. These values (6.1% for a BWR and 6.0% for a PWR) appear reasonable, if somewhat conservative, in light of the typical contribution to the core-melt

frequency of <10% from non-dominant accident sequences (e.g., at Oconee, non-dominant sequences contribute 7.5%).

As for the Type-1 risk analysis, the factor of 0.145 seems appropriate to represent the portion of the pipe system reliability improvement that is due to improvements to the pipe clamps. Based on the non-seismic total core-melt frequencies reported by Azarm, Boccio, and Farahzad (1983) for the *Reactor Safety Study* plants (6.0E-5/RY for the PWR and 2.9E-5/RY for the BWR), the reductions in core-melt frequency resulting from SIR are estimated as follows:

$$\begin{aligned}\Delta F_{PWR} &= (6.0E-5/RY)(0.060)(0.145) = 5.2E-7/RY \\ \Delta F_{BWR} &= (2.9E-5/RY)(0.061)(0.145) = 2.6E-7/RY\end{aligned}$$

Azarm, Boccio, and Farahzad (1983) reported the release category frequencies for the Reactor Safety Study plants (non-seismic). These can be combined with the dose consequence factors reported in Appendix D by Andrews, et al., (1983) to calculate weighted consequence values for the PWR and BWR, 1.1E+6 and 5.2E+6 person-rem, respectively. Combining these with the reductions in core-melt frequency yields the following estimates for the per-plant public risk reduction due to SIR:

$$\begin{aligned}\Delta W_{PWR} &= (5.2E-7/RY)(1.1E+6 \text{ person-rem}) = 5.7E-1 \text{ person-rem/RY} \\ \Delta W_{BWR} &= (2.6E-7/RY)(5.2E+6 \text{ person-rem}) = 1.4E+0 \text{ person-rem/RY}\end{aligned}$$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Stiff pipe Clamps (89)

2. Affected Plants (N):

	<u>N</u>
PWRs	27
BWRs	17

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

	<u>$\bar{T}$</u>
PWRs	33.4
BWRs	28.9

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

$$\Delta(\bar{F}D_R)_{PWR} = (19860 \text{ person-rem})(5.8E-7/RY) = 1.2E-2 \text{ person-rem/RY}$$

$$\Delta(\bar{F}D_R)_{BWR} = (19860 \text{ person-rem})(1.05E-6/RY) = 2.1E-2 \text{ person-rem/RY}$$

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

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
2.1E+1	1.3E+2	0

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

SIR involves labor in radiation zones to install and test pipe clamps at plants that determine that their stiff pipe clamps are not in compliance with NRC requirements. It was assumed that all 44 affected plants would require these hardware modifications. Utility labor hours for completing these modifications were estimated from data provided by Sciacca (1989). Data for installation of pipe hangers was used to represent installation of new pipe clamps. Installation labor hours were given as a function of the type of pipe hanger and the number of linear feet of pipe affected. For this analysis, it was assumed that stainless steel pipe hangers would be used on 1000 linear feet of 8-in. diameter Safety Class 1 pipe. Unit labor requirements for this installation activity were given as 4.6 man-hr/linear foot for a total labor requirement of 4,600 man-hr/plant. Labor productivity and learning curve effects factors were applied to the estimated labor requirements to account for work in radiation zones, in congested areas, manageability, and access/handling difficulties (see Table 3, Step 6), as described by Sciacca (1989). The value of the adjustment factor was calculated to be 10.08. Therefore, the total labor requirements in radiation zones for SIR implementation were estimated to be 4.6E+4 man-hr/plant.

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

Most of the pipe clamp installation activities will be performed within containment during reactor shutdowns. The radiation dose rate was assumed to be 25 mrem/hr (Andrews, et al., 1983).

$$D = (4.6E+4 \text{ man-hr})(0.025 \text{ rem/hr}) = 1.2E+3 \text{ person-rem/plant}$$

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

$$ND = (44 \text{ plants})(1.2E+3 \text{ person-rem/plant}) = 5.3E+4 \text{ person-rem}$$

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

Operation and maintenance labor for SIR is estimated to be needed for periodic inspection and testing of the new pipe clamps. This is assumed to be necessary because they are not conventional pipe clamps and the NRC may desire to monitor their performance on a routine basis. Labor requirements in radiation zones for this inspection activity are assumed to amount to 8 man-hr/RY. No labor productivity and learning curve effects adjustments were made for this relatively uncomplicated activity.

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

$$D_o = (8 \text{ man-hr/RY})(0.025 \text{ rem/hr}) = 0.2 \text{ rem/RY}$$

11. Total Occupational Dose Increase for SIR Operation/Maintenance, (NTD_o):

$$NTD_o = (0.2 \text{ rem/RY})[(27 \text{ PWRs})(33.4 \text{ RY/PWR}) + (17 \text{ BWRs})(28.9 \text{ RY/BWR})] = 280 \text{ person-rem}$$

12. Total Occupational Dose Increase (G):

<u>Best Estimate (man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
5.3E+4	1.6E+5	1.8E+4

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Stiff Pipe Clamps (89)

2. Affected Plants (N):

	<u>N</u>
PWRs	27
BWRs	17

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

	<u>$\bar{T}$</u>
PWRs	33.4
BWRs	28.9

Industry Costs (Steps 4 through 12):

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

$$\Delta(\bar{F}A)_{PWR} = (\$1.65E+9)(5.8E-7/R\bar{Y}) = \$9.6E+2/R\bar{Y}$$

$$\Delta(\bar{F}A)_{BWR} = (\$1.65E+9)(1.05E-6/R\bar{Y}) = \$1.7E+3/R\bar{Y}$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.7E+6	\$1.0E+7	0

6. Per-Plant Industry Resources for SIR Implementation:

SIR implementation involves two major steps. The first step is to perform the piping analyses to verify the adequacy of the existing stiff pipe clamp designs and, if applicable, to qualify the stiff pipe clamps. The second major step is applicable to plants that determine that their existing stiff pipe clamps are not in compliance with NRC requirements.

All 44 affected plants are assumed to be required to replace the existing pipe clamps in order to achieve compliance with the requirements.

Industry resource requirements for Step 1 are estimated to include 4 man-wks of utility labor to review as-built piping system designs, design specifications, and design analyses. This will be accomplished in order to identify areas in which stiff pipe clamps have been installed. In

addition, it is estimated that approximately 6 man-months of utility labor are required to perform the piping design analyses necessary to qualify the stiff pipe clamps. This includes computer resources.

Industry resource requirements for Step 2, the hardware modifications, were taken from Sciacca (1989). As discussed in Table 2, the hardware modifications are assumed to be represented by replacement of pipe hangers on 1000 linear feet of 8-in. diameter safety class 1 piping. The estimated unit costs for the pipe clamps, represented by pipe hangers, are given by Sciacca (1989) at \$21/linear foot for a total hardware cost of about \$21,000. Unit installation labor requirements are given at 4.6 man-hr/linear foot for a total installation labor requirement of about 4600 man-hr.

A labor productivity adjustment factor (Sciacca 1989) was applied to this labor estimate to account for work in radiation zones, in congested areas, difficulties in access and handling, and learning curve effects. The formula that was used to calculate this factor is:

$$C_L' = C_L [0.625 + 0.375 * (1 + F_L)(1 + F_{LC})]$$

where: C_L' = Adjusted installation labor estimate
 C_L = Installation labor from Sciacca (1989)
 F_L = Sum of labor productivity factors (see Abstract 2.1.7 of Sciacca 1989)
 F_{LC} = Learning curve factor (see Abstract 6.5 of Sciacca 1989)

The sum of the labor productivity factors (F_L) was calculated to be 7.0, based on the work being performed within containment and at dose rates of 25 mrem/hr. The learning curve factor (F_{LC}) was assigned a value of 2.6, based on the SIR being a first-time activity of this type. The total labor adjustment factor was estimated to be 10.08. Total SIR implementation labor requirements were therefore estimated to be 4.6E+4 man-hr/plant.

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

A total of 28 man-wks of utility labor were estimated for performance of the pipe-pipe clamp qualification analyses. At \$2270/man-wk, the per-plant costs for this activity were estimated to be \$63,560/plant. For hardware modifications, the per-plant costs were estimated at \$21,000/plant for hardware plus installation labor costs. Installation labor costs were estimated based on \$44/man-hr burdened labor rates (Andrews, et al., 1983). The total installation labor costs are therefore \$2.0E+6/plant. Total costs for this activity are estimated at \$2.1E+6/plant.

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

$$NI = (\$2.1E+6/\text{plant})(44 \text{ plants}) = \$9.2E+7$$

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

As discussed in Table 2, a total of 8 man-hr/Ry are estimated for inspection of the replacement pipe clamps at those plants requiring hardware modifications.

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

$$\begin{aligned} I_o &= [(8 \text{ man-hr/Ry}) / (40 \text{ man-hr/man-wk})] (\$2270/\text{man-wk}) \\ &= \$454/\text{Ry} \end{aligned}$$

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

$$\begin{aligned} NTI_o &= (\$454/\text{Ry}) [(27 \text{ PWRs})(33.4 \text{ Ry/PWR}) \\ &\quad + (17 \text{ BWRs})(28.9 \text{ Ry/BWR})] = \$6.3\text{E}+5 \end{aligned}$$

12. Total Industry Cost (S_I):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$9.3E+7	\$1.4E+8	\$4.7E+7

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

As discussed in Section 1.0, NRC implementation of the SIR includes development of proposed procedures for the categorization of pipe stresses due to clamp-induced loads, and procedures for evaluating those clamp applications where the ASME code stress indices and flexibility factors are not applicable. Development of these procedures would be a complicated problem and are estimated to require approximately 2 man-yr of labor to develop, review, and approve.

SIR implementation also includes establishment of a technical assistance program to verify analytical techniques and results using experimental data. The experimental program would involve design and construction of test equipment, preparation of test procedures, quality assurance activities, and analysis of test results. The design and construction of test equipment is estimated to cost approximately \$250,000. Analytical and quality assurance activities are estimated to require approximately 1 man-yr of labor, and would take advantage of the analytical work performed in support of the development of pipe stress evaluation techniques in support of the ASME code.

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

$$\begin{aligned} C_D &= (2 \text{ man-yr})(\$100,000/\text{man-yr}) + \$250,000 \\ &\quad + (1 \text{ man-yr})(\$100,000/\text{man-yr}) \\ &= \$550,000 \end{aligned}$$

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

NRC labor to support SIR implementation include development and issuance of a Generic Letter to the potentially affected plants and review and approval of the pipe stress analyses and pipe clamp-induced stresses. NRC labor requirements for these activities were taken from Sciacca (1989). Labor requirements to prepare and issue the Generic Letter were given by Sciacca (1989) at approximately 6 man-wks, including technical staff labor to prepare the Generic Letter and labor for management a legal reviews and concurrences. This effort is spread over all 44 potentially affected plants for a per-plant labor requirement of about 0.14 man-wk/plant. Review of utility submittals was estimated at 5 man-wks/plant (represented by review of a "complicated" Technical Specification change). The total NRC labor for support of SIR implementation is therefore about 5.1 man-wk/plant.

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

$$C = (5.1 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$1.2\text{E}+4/\text{plant}$$

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

$$NC = (\$1.2\text{E}+4/\text{plant})(44 \text{ plants}) = \$5.3\text{E}+5$$

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

The labor estimate for these activities, which are anticipated to be relatively minor, are assumed to be a nominal 0.5 man-day/Ry.

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

$$\begin{aligned} C_o &= [(4 \text{ man-hr/Ry})/(40 \text{ man-hr/man-wk})] (\$2270/\text{man-wk}) \\ &= \$227/\text{Ry} \end{aligned}$$

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

$$\begin{aligned} N\bar{T}C_o &= (\$227/\text{Ry})[(27 \text{ PWRs})(33.4 \text{ Ry/PWR}) \\ &\quad + (17 \text{ BWRs})(28.9 \text{ Ry/BWR})] = \$3.2\text{E}+5 \end{aligned}$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.4E+6	\$1.7E+6	\$1.1E+6

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 95, Loss of Effective Volume For Containment Recirculation Spray

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: The principal concern of this issue is the potential for water entrapment in the refueling canal for certain operating PWRs. The concern arises because if the refueling canal drains in a PWR dry containment are closed during plant operation and the plant undergoes a LOCA, that fraction of the containment spray which falls into the refueling canal would be prevented from returning to the containment emergency sump. Eventually the entire volume of the refueling canal could be filled with water which would then not be available for the recirculation mode of containment and reactor cooling. The proposed resolution involves changes in procedures to require operation with the drain valve in the open position to ensure that adequate water is available for the recirculation mode.

AFFECTED PLANTS: Operating PWRs = 26 (only those PWRs which were not reviewed under SRP 6.2.2)

Planned PWRs = 0

RISK/DOSE RESULTS (person-rem):

PUBLIC RISK REDUCTION = 2.2E+05

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	1.8E+03

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	0.41
SIR Operation/Maintenance =	0
Total of Above =	0.41
Accident Avoidance =	140

NRC COSTS:

SIR Development =	0.12
SIR Implementation =	0.03
SIR Operation/Maintenance Review =	0
Total of Above =	0.15

LOSS OF EFFECTIVE VOLUME FOR
CONTAINMENT RECIRCULATION SPRAY

ISSUE 95

1.0 SAFETY ISSUE DESCRIPTION

The principal concern of this issue is the potential for water entrapment in the refueling canal of certain operating PWRs. The concern arises because if the refueling canal drains in a PWR dry containment are closed during plant operation and the plant undergoes a LOCA, that fraction of the containment spray which falls into the refueling canal would be prevented from returning to the containment emergency sump. Eventually the entire volume of the refueling canal could be filled with water which would then not be available for the recirculation mode of containment and reactor cooling.

Standard Review Plan 6.2.2, "Containment Heat Removal Systems," includes the consideration of loss of effective recirculation volume as a specific review area. However, 26 PWRs have been identified which were licensed prior to the issuance of this Standard Review Plan (i.e., prior to 1975) and may not have been adequately reviewed for unacceptable water entrapment. If the volume of water held up in the refuelling canal is greater than the margin of excess water that is allotted in the design of the containment sump and recirculation systems, containment cooling and/or reactor cooling could be lost for some LOCA events. Although it is likely that the potential for entrapment of containment recirculation was considered during the OL reviews for these plants, the extent of the review in this area has not been documented. It is for this reason that this issue is being prioritized prior to initiating any extensive reexamination of those PWRs that were not reviewed under SRP 6.2.2.

PROPOSED RESOLUTION

For purposes of this analysis, the proposed Safety Issue Resolution (SIR) is assumed to consist of the following action(s):

1. Review containment sump and recirculation system (including refueling canal drain valve position while the plant is at power). Evaluate the recirculation system to determine if water entrapment could occur such that recirculation heat removal may be inadequate.
2. If necessary, change procedures such that the refuelling canal drain valve is in the open position while the reactor is operating to assure that the most water which can be available is available for recirculation.

AFFECTED PLANTS

The resolution of this safety issue is assumed to affect only those PWRs not reviewed under SRP 6.2.2 (i.e., plants that were licensed before 1975).

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with the safety issue resolution are estimated in this section.

PUBLIC RISK REDUCTION

It is assumed that resolution of this issue will serve to identify a need for procedural changes at approximately one-half of those PWRs identified as affected by this issue. The remaining one-half are assumed to already operate with the refuelling canal drain valve in the open position. A reduction in public risk at those plants which are currently operating with the drain valve in the closed position will result from an increase in the reliability of the containment spray recirculation and emergency coolant recirculation systems. The results of the public risk reduction estimates are summarized in Table 1.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Loss of Effective Volume for Containment Recirculation Spray, 95

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

The issue is assumed to affect 26 operating PWRs which were licensed prior to 1975, and hence not reviewed under Standard Review Plan (SRP) 6.2.2, "Containment Heat Removal Systems." However, only one-half of these are expected to be vulnerable to water entrapment, and hence only 13 plants are expected to experience a reduction in public risk. The average remaining life of the plants is 24 years.

3. Plants Selected for Analysis:

Oconee-3 is used as a representative PWR. The Oconee risk equations are extracted from its RSSMAP study, NUREG/CR-1659/2 (Kolb 1981).

4. Parameters Affected by SIR:

F - Containment Spray Recirculation System failure
H - Emergency Coolant Recirculation System failure

5. Base-Case Values for Affected Parameters:

$P(F) = 0.4$ given event H (dependent failure)
 $P(H) = 7.7E-02/\text{demand}$

6. Affected Accident Sequences and Base-Case Frequencies:

T2MQH	- τ	(PWR-3)	5.5E-05/py
T2MQH	- β	(PWR-5)	8.0E-07/py
T2MQH	- ϵ	(PWR-7)	5.5E-05/py
S3H	- τ	(PWR-3)	5.0E-05/py
S3H	- β	(PWR-5)	7.3E-07/py
S3H	- ϵ	(PWR-7)	5.0E-05/py
T2MQFH	- τ	(PWR-2)	2.5E-05/py
T2MQFH	- β	(PWR-4)	3.7E-07/py
T2MQFH	- ϵ	(PWR-6)	2.5E-05/py
S3FH	- τ	(PWR-2)	2.1E-05/py
S3FH	- β	(PWR-4)	3.1E-07/py
S3FH	- ϵ	(PWR-6)	2.1E-05/py
S2FH	- α	(PWR-1)	1.3E-07/py
S2FH	- β	(PWR-4)	9.5E-08/py
S2FH	- ϵ	(PWR-6)	1.0E-05/py
TOTAL:			<u>3.1E-04/py</u>

7. Affected Release Categories and Base-Case Frequencies:

PWR-1	1.30E-07/py
PWR-2	4.60E-05/py
PWR-3	1.05E-04/py
PWR-4	7.75E-07/py
PWR-5	1.53E-06/py
PWR-6	5.60E-05/py
PWR-7	<u>1.05E-04/py</u>
TOTAL:	<u>3.10E-04/py</u>

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

$$\bar{F} = 3.1E-04/py$$

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

$$W = 801 \text{ person-rem/py}$$

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

$$P(F) = 0.4, \text{ given event H}$$

$$P(H) = 7.7E-03/\text{demand}$$

11. Affected Accident Sequences and Adjusted-Case Frequencies:

T2MQH	- τ	5.5E-06/py
T2MQH	- β	8.0E-08/py
T2MQH	- ϵ	5.5E-06/py
S3H	- τ	5.5E-06/py
S3H	- β	7.3E-08/py
S3H	- ϵ	5.0E-06/py
T2MQFH	- τ	2.5E-06/py
T2MQFH	- β	3.7E-08/py
T2MQFH	- ϵ	2.5E-06/[y
S3FH	- τ	2.1E-06/py
S3FH	- β	3.1E-08/py
S3FH	- ϵ	2.1E-06/py
S2FH	- α	1.3E-08/py
S2FH	- β	9.5E-09/py
S2FH	- ϵ	1.0E-06/py
TOTAL:		3.1E-05/py

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1	1.30E-08/py
PWR-2	4.60E-06/py
PWR-3	1.05E-05/py
PWR-4	7.75E-08/py
PWR-5	1.53E-07/py
PWR-6	5.60E-06/py
PWR-7	<u>1.05E-05/py</u>
TOTAL:	3.10E-05/py

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

$$F = 3.1E-05/\text{py}$$

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

$$W^* = 80.1 \text{ person-rem/py}$$

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

$$\Delta\bar{F} = 2.8E-04$$

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

$$\Delta W = 721 \text{ person-rem/py}$$

17. Total Public Risk Reduction, (ΔW)_{Total}:

<u>Best Estimate</u> <u>person-rem</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
2.2E+05	6.6E+05	0

ATTACHMENT 1 (to Table 1)

This prioritization analysis of Generic Issue 95 uses the Oconee-3 Probabilistic Risk Assessment (Kolb 1981) as a basis for evaluating the expected reduction in public risk due to issue resolution. The principal concern of this issue is the potential for water entrapment in the refueling canal for certain operating PWRs. If the refueling canal drains in a PWR dry containment are closed during plant operation and the plant undergoes a LOCA, that fraction of the containment spray which falls into the refueling canal would be prevented from returning to the containment emergency sump. Eventually the entire volume of the refueling canal could be filled with water which would then not be available for the recirculation mode of containment and reactor cooling. This could result in inadequate containment pressure control and/or inadequate long-term decay heat removal.

The principal system failures of concern in the recirculation mode are the containment spray recirculation and emergency coolant recirculation systems. These system failures are represented in the Oconee-3 PRA by events "F" and "H" respectively.

Following a postulated LOCA and after the injection phase of containment spray is complete, it is likely that containment spray flow could be discontinued while maintaining containment pressure reduction with the containment fan cooler units, and returning all of the recirculated water to the core. Since the containment fan coolers may be sufficient to maintain containment pressure following spray injection, the dominant safety concern is the potential loss of emergency coolant recirculation, event H. In this analysis, it is assumed that if there is not enough water for effective emergency coolant recirculation, then core-melt will ensue. It has been estimated that containment spray recirculation will fail, given a failure of the emergency coolant recirculation system, with a conditional probability of approximately 0.4. This conditional probability of containment spray recirculation failure is derived from the original Oconee-3 PRA accident sequence quantifications.

In order to estimate the safety significance of this issue for the limiting case (i.e., the volume of containment spray water entrapped in the refueling canal is of sufficient quantity to degrade spray and/or emergency coolant recirculation), it is necessary to estimate the probability of containment spray recirculation and emergency coolant recirculation system failure, given significant water entrapment. This is needed to predict the "base-case" risk of a vulnerable plant which is modeled by adjusting the original affected parameter values, events F and H, to reflect a greater likelihood of recirculation failure. This simulates the oversight of the potential for water entrapment in the plant's original safety analyses. The frequencies used for events F and H in the original Oconee-3 PRA do not include consideration of the potential for water entrapment and thus are used here to represent the system failure frequencies for a plant not vulnerable to water entrapment. These original PRA values will serve as the "adjusted-case" (after resolution) model of a plant which is currently vulnerable to water entrapment.

The overall reduction in public risk (core-melt frequency) due to issue resolution is estimated by subtracting the "adjusted-case," affected public risk (core-melt frequency) from the "base-case," affected public risk (core-melt frequency). This risk reduction is due to the expected improvement in the

reliability of the emergency coolant recirculation (decay heat removal) and containment spray recirculation systems.

There are five dominant accident sequences identified in the PRA that involve a loss of emergency coolant recirculation. These five sequences contribute to approximately 38% of the total core damage frequency at Oconee-3. The dominant accident sequence is characterized by the following scenario: a loss of the power conversion system (T_2M) occurs as an initiating event which is followed by failure of one of the pressurizer safety/relief valves to reclose (Q). This is followed by failure of the emergency coolant recirculation system (H). This sequence is a transient induced LOCA (T_2MQ) in which the emergency core cooling system fails in the recirculation phase (H). The sequence assumes that the LOCA systems perform successfully during the injection phase (i.e., water source is the borated water storage tank) but the emergency coolant system fails during the recirculation phase (i.e., water source is the containment sump). The frequency of the accident sequence, as given in the PRA (with no consideration of water entrapment vulnerability) is estimated as $1.1E-05/py$.

The five dominant accident sequences affected by this issue, along with their base-case frequencies (before issue resolution) and estimated adjusted-case frequencies (after resolution, with no water entrapment vulnerability) are shown below:

<u>Sequence</u>	<u>Base-Case,* Affected Sequence Frequency (/py)</u>	<u>Adjusted-Case,* Affected Sequence Frequency (/py)</u>
T_2MQH	$1.5E-05$	$1.1E-05$
S_3H	$1.4E-05$	$1.0E-05$
T_2MQFH	$6.9E-06$	$5.0E-06$
S_3FH	$5.8E-06$	$4.2E-06$
S_2FH	$1.4E-06$	$1.0E-06$
	<u>$4.3E-05$</u>	<u>$3.1E-05$</u>

* Base-Case frequencies are based on the limiting case system configuration (i.e., water entrapment significantly degrades recirculation operations). Adjusted-Case frequencies reflect the removal of the water entrapment vulnerability which improves plant safety to the level where it had previously been assumed to be.

Therefore, the base-case frequency of core damage due to water entrapment (in the limiting case) is estimated to be $3.1E-04/py$. The base-case accident frequencies shown above reflect the use of an order of magnitude increase in the probability of emergency coolant recirculation system failure. This assumption is based on the engineering judgment that if water entrapment can occur, an order of magnitude reduction in the emergency coolant recirculation system availability will result. This adjustment results in a "base-case" (limiting case) value for event H of $7.7E-02/demand$ (c.f. original value of $7.7E-03/demand$). It should be recalled that the limiting case is assumed to represent only one-half of the 26 affected plants. The remaining one-half of the plants are assumed to not to have the water entrapment vulnerability (due to hardware or procedural improvements

from the limiting case). These assumptions are believed to result in an overall conservative estimate of the issue's safety significance. However, the results of these "limiting case" assumptions indicate that a reexamination of those reactors not reviewed under SRP 6.2.2 may be justifiable.

OCCUPATIONAL DOSE

Since this safety issue resolution (SIR) involves procedural changes for the most part, it is assumed that the additional personnel exposure in radiation zones due to the implementation and operation/maintenance would be negligible. The results of the occupational dose estimates are summarized in Table 2. Table 2 includes the expected occupational dose reduction due to accident avoidance.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Loss of Effective Volume for Containment Recirculation Spray, 95

2. Affected Plants (N):

The issue is assumed to affect 26 operating PWRs which were licensed prior to 1975. However, only one-half of these plants are expected to, after review, become vulnerable to water entrapment. Therefore, only 13 plants are expected to experience a reduction in public risk.

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

The average remaining life of the plants is 24 years.

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

$$\Delta(\bar{F}D_R) = (19,860 \text{ person-rem})(2.8E-4/\text{py}) = 5.6 \text{ person-rem/py}$$

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

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Upper Bound</u> <u>(person-rem)</u>	<u>Lower Bound</u> <u>(person-rem)</u>
1,750	5,250	0

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

The occupational dose increase for SIR implementation is zero because the implementation will consist of modifications to plant procedures.

9-11. Steps Related to Occupational Dose Increase for SIR Operation and Maintenance:

The occupational dose increase for SIR operation and maintenance is anticipated to be zero because plant operation with the refueling canal drain valve in the open position has no affect on occupational exposure.

12. Total Occupational Dose Increase (G):

Best Estimate
(person-rem)

negligible

3.0 SAFETY ISSUE COSTS

The industry and NRC costs associated with resolution of this safety issue 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:

Loss of Effective Volume for Containment Recirculation Spray, 95

2. Affected Plants (N):

The issue is assumed to affect 26 PWRs which were licensed prior to 1975, and hence not reviewed under Standard Review Plan (SRP) 6.2.2, "Containment Heat Removal Systems." However, only one-half of these plants are expected to be vulnerable to water entrapment, and hence only 13 plants are expected to experience a reduction in public risk.

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

The average remaining life of the plants is 24 years.

Industry Costs (Steps 4 through 12):

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

$$(\bar{F}A) = (2.8E-04/\text{py})(\$1.65E+09/\text{event}) = \$4.6E+05/\text{py}$$

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

<u>Best Estimate</u>	<u>Uppper Estimate</u>	<u>Lower Estimate</u>
\$ 1.4E+08	\$4.2E+08	0

6. Per-Plant Industry Resources for SIR Implementation:

2 man-weeks/plant to evaluate present containment recirculation system and to review normal valve position and current operating procedures.

2 man-weeks/plant to respond to NRC request for system analysis

3 man-weeks/plant to revise Technical Specifications

3 man-weeks/plant to make Administrative changes/modifications.

10 man-weeks/plant, for those plants needing changes

4 man-weeks/plant, for those plants not needing changes

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

13 plants: (10 man-weeks/plant)(\$2270/man-week) = \$22,700/plant

13 plants: (4 man-weeks/plant)(\$2270/man-week) = \$9,080/plant

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

$$\begin{aligned}(\$22,700/\text{plant})(13 \text{ plants}) &= \$295,100 \\(\$9,080/\text{plant})(13 \text{ plants}) &= \$118,040 \\ \text{TOTAL:} &= \$413,140\end{aligned}$$

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

No additional labor expected for operation and maintenance.

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

No additional costs.

11. Total Industry Cost for SIR Operation and Maintenance:

No additional costs.

12. Total Industry Cost (S_I):

<u>Best Estimate</u>	<u>Upper Estimate</u>	<u>Lower Estimate</u>
\$4.1E+05	\$6.2E+05	\$21.E+05

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development

Resolution of this issue involves a review by the responsible technical branches to determine which of the 26 affected plants has a vulnerability to water entrapment. This analysis assumes that a review of two man-weeks per plant will be required to verify the licensee's analysis of the matter of water entrapment.

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

$$C_D = (2 \text{ man-weeks/plant})(\$2270/\text{man-week})(26 \text{ plants}) = \$1.2E+05$$

15. Per-Plant NRC Labor Support of SIR Implementation

Resolution of this issue assumes procedural changes at one-half of the potentially affected plants. Proposals and reviews of such changes are estimated to require 1 man-week/plant.

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

$$C = (1 \text{ man-week/plant})(\$2270/\text{man-week}) = \$2270/\text{plant}$$

17. Total NRC Cost for Support of SIR Implementation:

$$NC = (\$2270/\text{plant})(13 \text{ plants}) = 3.0E+04$$

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

No reviews of operation and maintenance activities are anticipated beyond those that are presently performed.

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.5E+05	\$2.1E+05	\$8.8E+04

REFERENCES (for Issue 95)

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.

Houston, R. W. 1984. "Task Interface Agreement (TIA) #83-144: Loss of Effective Volume for Containment Recirculation Spray for H. B. Robinson, Unit 2 (TAC #53223)." Memorandum to G. Lainas, U.S. Nuclear Regulatory Commission, Washington, D.C.

Kolb, G. J. et al. 1981. *Reactor Safety Study Methodology Applications Program: Oconee #3 PWR Power Plant*. NUREG/CR-1659/2, Sandia National Laboratories, Albuquerque, New Mexico.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 100, Once-Through Steam Generator Level

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: A concern has arisen regarding potential operation of Once-Through Steam Generators (OTSGs) with the water level above the aspirator ports. The concern is that covering the aspirator ports will prevent the preheating of relatively cold feedwater which may increase thermal stresses in the downcomer region of the of the OTSG vessel. In addition, backflow of the feedwater through the aspirator ports onto the OTSG tubes could produce increased thermal stresses on the tubes. The proposed resolution is to establish technical specification requirements on the maximum operating levels of OTSGs to prevent covering of the aspirator ports.

AFFECTED PLANTS: Only Babcock and Wilcox (B&W) PWR plants are provided with OTSGs. A total of 10 plants are affected.

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION =	3.7E+1
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	2.7E-1

COST RESULTS (\$10⁶):

INDUSTRY COSTS:	
SIR Implementation =	4.1
SIR Operation/Maintenance =	2.8
Total of Above =	6.9
Accident Avoidance =	0.02
NRC COSTS:	
SIR Development =	0.32
SIR Implementation =	0.09
SIR Operation/Maintenance Review =	0.32
Total of Above =	0.73

ONCE-THROUGH STEAM GENERATOR LEVEL

ISSUE 100

1.0 SAFETY ISSUE DESCRIPTION

Once-Through Steam Generators (OTSGs) are a unique feature of reactors designed by Babcock and Wilcox (B&W). A schematic diagram of a typical OTSG is provided in Figure 1. The method of injecting feedwater is unique to the OTSG design. Feedwater is injected from a header located at approximately mid-elevation of the OTSG. A total of 32 feedwater nozzles extend radially through the shell about the circumference of the vessel into an annular downcomer region. As the feedwater is sprayed into the downcomer, the condensing action of the relatively cold (455°F) feedwater draws quality steam from the tube bundle of the boiler through the aspirator ports in the inner shroud. This bleed steam heats the feedwater rapidly to saturation temperature (about 535°F), thus preventing thermal shocking of the shell.

In 1984, operators of the Crystal River-3 plant submitted a Technical Specification change request to raise the OTSG operating level limit. At that time, a question arose regarding operation of OTSGs with the water levels above the aspirator ports. The concern is that covering the aspirator ports would prevent the feedwater from being preheated with aspirating steam which would produce significant temperature gradients in the downcomer inner shroud, with increased thermal stresses. In addition, backflow of the relatively cold feedwater through the aspirator ports onto the OTSG tubes could produce increased thermal stresses on the tubes. If the tube or shell wall stresses exceed design limits, failures may occur. Failure of the downcomer inner shroud by cracking could also introduce feedwater directly onto the tube bundle resulting in similar concerns as water backflowing through the aspirator ports.

PROPOSED RESOLUTION

The proposed resolution of this safety issue is to require B&W plants to implement technical specifications which limit the maximum operating levels of OTSGs to a level below the aspirator ports. OTSG operating levels must be established based on plant-specific parameters, including required superheat and steam quality, which will determine the maximum operating limit. A detailed "generic" analysis will also be needed to provide some degree of confidence that operation of OTSGs at levels near or above the aspirator ports does not introduce a significant safety problem at B&W plants.

Figure 1. Typical Once-Through Steam Generator

AFFECTED PLANTS

This safety issue affects only plants that are provided with OTSGs, i.e., B&W-designed PWRs. A list of the affected plants is provided below:

- Three Mile Island 1 and 2⁶
- Oconee 1, 2, and 3
- Crystal River 3
- Davis-Besse
- Arkansas Nuclear One 1 (ANO-1)
- Rancho Seco
- Bellefonte 1 and 2⁷

⁶ This plant is indefinitely shut-down and is not included in the analysis.

⁷ Construction of these two plants has been indefinitely delayed. They are, however, included in the analysis.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this issue are estimated in this section and summarized in Tables 1 and 2. Attachment 1 describes the basis for the risk and dose changes that are estimated in the Tables.

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

Once-Through Steam Generator Level (100)

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

This issue affects only plants provided with OTSGs, i.e., all B&W-designed PWRs.

	$\frac{N}{10}$	$\frac{\bar{T} \text{ (yr)}}{28.2}$
PWRs		

The number and average remaining lifetimes of affected plants were calculated from Appendix C of NUREG/CR-2800 (Andrews, et al., 1983).

3. Plants Selected for Analysis:

Representative PWR: Oconee-3. Oconee-3 is a B&W plant.

4. Parameters Affected by SIR:

A new accident sequence was developed to model the effects of the proposed safety issue resolution. The parameters affected by this issue are defined in Attachment 1.

5. Base-Case Values for Affected Parameters:

See Attachment 1 for definitions and quantification of the affected parameters and their base-case values.

6. Affected Accident Sequences and Base-Case Frequencies:

The base-case value of the affected accident sequence was estimated to be $7.27\text{E-}8/\text{RY}$.

7. Affected Release Categories and Base-Case Frequencies:

$$7.27\text{E-}8/\text{RY} \left\{ \begin{array}{l} \gamma \text{ (PWR-3)} = 3.6\text{E-}8 / \text{RY} \\ \beta \text{ (PWR-5)} = 5.3\text{E-}10/\text{RY} \\ \epsilon \text{ (PER-7)} = 3.6\text{E-}8 / \text{RY} \end{array} \right.$$

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

$$\bar{F} = 7.27\text{E-}8/\text{RY}$$

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

$$W = 0.20 \text{ person-rem/RY}$$

10. Adjusted-Case Values for Affected Parameters:

The parameter affected by the proposed resolution deals with the probability of failure to reduce feedwater flow given the occurrence of a feedwater overfeed incident. See Attachment 1 for details.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

The adjusted-case value of the affected accident sequence was estimated to be $2.43\text{E-}8/\text{RY}$.

12. Affected Release Categories and Adjusted-Case Frequencies:

$$2.43\text{E-}8/\text{RY} \begin{cases} \gamma \text{ (PWR-3)} & = 1.2\text{E-}8 / \text{RY} \\ \beta \text{ (PWR-5)} & = 1.8\text{E-}10/\text{RY} \\ \epsilon \text{ (PER-7)} & = 1.2\text{E-}8 / \text{RY} \end{cases}$$

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

$$\bar{F}^* = 2.43\text{E-}8/\text{RY}$$

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

$$W^* = 0.066 \text{ person-rem/RY}$$

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

$$\Delta\bar{F} = 4.8\text{E-}8/\text{RY}$$

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

$$\Delta W = 0.13 \text{ person-rem/RY}$$

17. Total Public Risk Reduction, (ΔW)_{Total}:

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
37	1700	0

ATTACHMENT 1

This attachment describes the basis for the core-melt frequency and public risk reduction estimates presented in Table 1. Basically, the proposed safety issue resolution (SIR) would preclude plant operations when OTSG water levels exceeded a pre-set maximum level. The effects on plant risk of exceeding this level could be the potential structural failure of the steam generator shell and/or downcomer shroud which, if it were to occur, could result in a total loss of heat removal capability through the steam generator. A second potential effect could involve rupture of steam generator tubes and the subsequent loss-of-coolant accident conditions associated with a small rupture in the primary coolant system.

The safety significance of this issue is related to the potential effects of forcing main feedwater through the aspirator ports rather than into the downcomer region where it is normally heated to saturation temperature by bleed steam drawn through the aspirator ports. The first effect of allowing the steam generator liquid level to exceed the level of the aspirator ports may be to halt the flow of bleed steam into the downcomer region. If the flow of bleed steam were to be halted, the rapid feedwater heating in the downcomer region would be degraded and thermal shocking of the shell may occur.

The second effect of allowing the aspirator ports to become covered is the possibility of forcing main feedwater back through the aspirator ports and onto the steam generator tubes. This could potentially result in thermal shocking of the tubes and subsequent tube rupture events. However, the effects on steam generator tube integrity from this particular event are within the steam generator design bases. A partial list of these design bases is summarized below:

- fifteen thousand cycles of adding 40°F feedwater at 875 gpm when at hot standby conditions (normal condition)
- five hundred cycles of adding 40°F feedwater at 875 gpm during loading conditions (normal condition)
- five hundred cycles of adding 100°F feedwater at 875 gpm during loading conditions (normal condition)
- seven cycles of adding 40°F feedwater at 1750 gpm during a steam line break (faulted condition)
- two hundred and eighty cycles of adding 40°F feedwater at 1750 gpm with the flow initiated 30 seconds after a loss of main feedwater (faulted condition).

Given that main feedwater is normally at about 450°F during loading conditions, the thermal effects of adding 40°F emergency feedwater would be substantially greater than those of spilling main feedwater through the aspirator ports and onto the tubes. In addition, the aspirator ports are located near the middle of the tube sheet which is a less-stressed position than the location of the emergency feedwater nozzle where the 40°F feedwater would be introduced. This further indicates that the steam generator tubes are likely to withstand the

introduction of 450°F main feedwater through the aspirator ports and onto the tubes.

Structural failure of the OTSG shell or downcomer would result in a loss of SG cooling for the primary coolant system. The plant would respond by tripping the reactor. Operators would then be forced to take action to isolate the affected steam generator by closing the Main-Steam Isolation Valves (MSIVs) and tripping the main feedwater (MFW) pump. Manual action would be necessary to accomplish these functions because water level in the steam generator would decrease as a result of the failure, which would send a low steam generator level signal to MFW pumps. The MFW pumps would increase their flow as a result of the low level signal, necessitating a manual trip of the MFW pumps. Similarly, Emergency Feedwater (EFW) pumps would have to be manually tripped. As a result, the reactor coolant temperature and pressure will increase. The High Pressure Injection (HPI) system would then be needed to maintain primary coolant inventory and core heat removal functions. With heat removal through the intact steam generator, cooling can be maintained for an extended period. Operators could use HPI in conjunction with manual operation of pressurizer spray or pilot-operated relief valves (PORVs) to reduce primary coolant pressure and cool the primary system to the entry conditions for decay heat removal (DHR) systems. When these conditions are met, the plant could proceed to cold shutdown.

Steam generator tube rupture (SGTR) incidents are somewhat specialized cases of small-break loss of coolant accidents (LOCAs). The main difference between small LOCAs and SGTR events is that the energy released in a SGTR would not be relieved to the reactor building so HPI actuation on high containment pressure would not occur. In addition, the energy release rate in the SGTR event is smaller than in a small LOCA.

The SGTR will cause a slow decrease in primary coolant pressure until the low-pressure setpoint for the reactor protection system is reached. With feedwater available, the primary system will continue to depressurize until the setpoint for HPI actuation. If MFW is not available and EFW does not operate as intended, the energy released through the SGTR will not be sufficient to prevent the primary system from repressurizing. In this case, HPI must be manually initiated to prevent core uncover.

Since leakage from the SGTR would be into the secondary system, the leakage would not flow to the containment sump. Therefore, high-pressure recirculation would not be possible. Operators would need to take action to replenish the source of HPI water, which is typically the borated water storage tank (BWST), or establish another mode of cooling. The most desirable mode of cooling would be the DHR systems. Operators could lower steam pressure by controlling the turbine bypass valves or opening the valves off the main steam line. Alternatively, operators could restart the reactor coolant pumps and reduce pressure by manual operation of the pressurizer spray. PORVs could also be cycled to reduce primary system pressure. Core cooling could also be maintained by replenishing the BWST and continuing HPI cooling or recirculating coolant that has blown down through the PORV into the containment sump. The latter would require isolation of the leakage to the secondary system by closing the Main Steam Relief Valves (MSRVs).

Based on the preceding discussion, the effects of operating a OTSG at high water levels may affect the reliability of the steam generator in terms of

responding to a transient. The transient of concern to this issue involves an increase in feedwater flow to one or more loops. If the increased feedwater flow continues, e.g., if the steam generator high-level detection system is failed, feedwater control valves are failed, or the main feedwater pump fails to trip, the steam generator water level may exceed the aspirator port level. The resultant backflow of feedwater through the aspirator ports may then result in the SGTR or steam generator shell failure accidents described above. The base-case and adjusted-case frequencies of these accidents are calculated in the following paragraphs.

Transient initiating event frequencies were developed in detail in NUREG/CR-3862 (Mackowiak, Gentillon, and Smith 1985). This document develops frequencies for a number of initiating event, including one that closely approximates the initiating events of interest to this issue. This would be PWR Category 19, "Increase in feedwater flow." The average frequency of this transient category was determined to be 0.13 transients/RY for the B&W plants of interest to this analysis (see Section 1.0).

The next event to occur involves failure to reduce main feedwater flow. The probability of this event was modeled in NUREG/CR-4386 (Bickford and Tabatabai 1985) as the simultaneous occurrence of undetected failure of the steam generator high-level trip AND operator failure to terminate the overfeed event. The probabilities of these events were estimated at 0.047/demand and 0.7/demand, respectively (Note: a range of values from 0.1 to 0.7 was given for operator failure to terminate the event. The highest value was used in the analysis for conservatism). The combined probabilities of these events were calculated to be 0.033/demand.

Assuming the overfeed event is not terminated and the steam generator level exceeds the aspirator port level, the next events to occur would be either SGTR or steam generator shell failure. The conditional probability of SGTR was taken from NUREG-0844 (NRC 1988). This study gave a conditional probability of 0.027 that the differential pressure conditions across the steam generator tubes that are produced in a main steam line break (MSLB) accident will result in rupture of one steam generator tube. This probability is considered to be a conservative approximation of the SGTR probability for this analysis in that the thermal stress conditions that would be produced by introducing 450°F main feedwater onto the steam generator tubes would be substantially less severe than the differential pressure conditions produced in a MSLB accident.

The conditional SGTR probability discussed above will also be used to approximate the probability of steam generator shell failure because no information was available to estimate this probability. However, this value is considered to be conservative because structural failures of steam generators were not significant contributors to failures of the emergency feedwater system (events CONST1 and CONST2) in the Oconee PRA summarized by Andrews, et al., (1983). Since failures other than steam generators dominated the probabilities given for CONST1 and CONST2 and since the failure probabilities given for these events were both less than 0.001, it was concluded that the probability of steam generator structural failures is significantly less than 0.001. Therefore, a probability of 0.027 given the occurrence of a loss of the main feedwater preheat function that is provided by the aspirator ports in OTSGs is considered conservative.

The final event in this accident sequence involves failure to respond to the SGTR or steam generator structural failure events and bring the plant to a stable shutdown condition. The probability for this event was derived from information in NSAC-60 (Sugnet, et al., 1984) for responding to a SGTR event. Since the initial actions taken to respond to a SGTR and to steam generator structural failure events would be to isolate the affected steam generator, it is assumed that the failure probabilities for responding to both these events are approximately equal. The probability used in this analysis was calculated by dividing the core melt frequency from SGTR sequences ($2.7\text{E-}6/\text{yr}$) by the SGTR initiating event frequency ($8.6\text{E-}3/\text{yr}$). The resulting failure probability is $3.14\text{E-}4/\text{demand}$.

The base-case frequency for this new accident sequence is the product of the initiating event frequency ($0.13/\text{RY}$), the conditional failure probability to reduce feedwater flow ($0.033/\text{demand}$), the sum of the conditional probabilities of SGTR and steam generator structural failures ($0.054/\text{demand}$), and the conditional probability of failure to bring the plant to a stable shutdown condition ($3.14\text{E-}4/\text{demand}$). The resulting base-case accident sequence frequency is estimated to be $7.27\text{E-}8/\text{RY}$.

The containment failure modes and release categories for this accident sequence were selected based on inspection of Andrews, et al., (1984). It was observed that the containment failure modes and release categories for sequences initiated by T_2 and T_3 transients for all but one accident sequence involved release categories PWR-3, PWR-5, and PWR-7. One sequence ($T_2\text{MQFH}$) involved release categories PWR-2, PWR-4, and PWR-6. For conservatism, the release category series that results in the highest consequences was used in this analysis. This was determined by summing the products of the containment failure probabilities and the corresponding dose consequences for both release category series and then selecting the series that resulted in the highest weighted consequence value. This was determined to be release categories PWR-3, PWR-5, and PWR-7. The containment failure probabilities and corresponding dose consequences were combined with the base-case accident sequence frequency to calculate the base-case public risk, as follows:

BASE-CASE PUBLIC RISK					
Accident Sequence Frequency, (/RY)	Containment Failure Mode	Containment Failure Probability	Release Category	Consequence, person-rem	Public Risk person-rem/Ry
$7.27\text{E-}8$	γ	0.5	PWR-3	$5.4\text{E}+6$	0.20
	β	0.0073	PWR-5	$1.0\text{E}+6$	0.00053
	ϵ	0.5	PWR-7	$2.3\text{E}+3$	0.000084
TOTAL					0.20

The proposed safety issue resolution is intended to prevent plants with OTSGs from operating at liquid levels above the aspirator ports. Therefore, the adjusted-case accident sequence frequency will be calculated as if the proposed resolution will increase the reliability of the feedwater flow detection and

feedwater trip systems, i.e., decrease the failure probabilities of these systems. A reduction factor of 0.33 will be used to approximate the effects of issue resolution on the base-case frequency for this event. This reduction was based on information in NUREG/CR-4386 (Bickford and Tabatabai 1985) in which a factor of 0.33 reduction was estimated for an enhanced testing and inspection program for steam generator level instrumentation and feedwater controls to prevent feedwater from flowing into the main steam lines. The resulting conditional probability of failure to reduce main feedwater flow given a feedwater overfeed event is 0.011/demand. The adjusted-case accident sequence frequency is therefore 2.43E-8/RY. The adjusted-case public risk is estimated as follows:

ADJUSTED-CASE PUBLIC RISK					
Accident Sequence Frequency, (/RY)	Containment Failure Mode	Containment Failure Probability	Release Category	Consequence, person-rem	Public Risk person-rem/RY
2.43E-8	γ	0.5	PWR-3	5.4E+6	0.066
	β	0.0073	PWR-5	1.0E+6	0.00018
	ϵ	0.5	PWR-7	2.3E+3	0.000028
TOTAL:					0.066

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Once-Through Steam Generator Level (100)

2. Affected Plants (N):

PWRs	$\frac{N}{10}$
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3. Average Remaining Lives of Affected Plants ($\bar{T}$):

PWRs	$\frac{\bar{T}}{28.2 \text{ yr}}$
------	-----------------------------------

The value of $\bar{T}$ was calculated from the average remaining lifetime data given in Appendix C of NUREG/CR-2800 for the B&W plants listed in Section 1.0.

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

$$\Delta(\bar{F}D_R) = (19,860 \text{ person-rem})(4.8E-8/\text{RY}) = 9.5E-4 \text{ person-rem/RY}$$

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

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
0.27	2.4	0

6-12. Steps leading to calculation of total utility labor in radiation zones and occupational dose increases due to SIR implementation, operation, and maintenance

Incremental labor in radiation zones and occupational dose increases due to SIR implementation, operation, and maintenance are estimated to be 0. There are no significant hardware fixes and no plant modifications in radiation zones associated with SIR.

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Once-Through Steam Generator Level (100)

2. Affected Plants (N):

PWRs	$\frac{N}{10}$
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3. Average Remaining Lives of Affected Plants ($\bar{T}$):

PWRs	$\frac{\bar{T}}{28.2 \text{ yr}}$
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The value of $\bar{T}$ was calculated from the average remaining lifetime data given in Appendix D of NUREG/CR-2800 for the B&W plants listed in Section 1.0.

Industry Costs (Steps 4 through 12):

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

$$\Delta(\bar{F}A) = (\$1.65E+9)(4.8E-8/R\bar{Y}) = \$7.9E+1/R\bar{Y}$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$2.2E+4	\$2.0E+5	0

6. Per-Plant Industry Resources for SIR Implementation:

SIR includes preparation of safety analyses to support a revision to plant-specific Technical Specifications requiring a maximum operating water level in OTSGs. The safety analyses necessary to support such a Technical Specification revision are anticipated to be substantial because of the complexity of the problem, the number of different scientific and engineering disciplines involved, including thermal and thermal-hydraulics analysis, structural analysis, and mechanical/ materials engineering, in addition to substantial computer costs. Industry labor requirements to perform site-specific safety analyses were estimated to be in the range of 2 to 2.5 man-yrs. A 25% adder will be included for engineering/quality assurance activities. Resources will also be needed to prepare the revised Technical Specification and license amendment package.

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

Per-plant industry costs for SIR implementation are:

• Perform safety analysis: (@ 2.5 man-yr/plant)(\$100,000/man-yr)		\$ 250,000
• Computer costs		50,000
	SUBTOTAL	<u>300,000</u>
• Engineering/QA (@ 25%)		75,000
	SUBTOTAL	\$ 375,000
• Prepare Technical Specification, license amendment package (Sciaccia 1989).		<u>\$ 35,000</u>
	TOTAL	\$ 410,000

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

$$NI = (\$410,000/\text{plant})(10 \text{ plants}) = \$4.1\text{E}+6$$

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

Implementation of SIR will result in an increase in the number of transients requiring reactor shutdown at B&W plants. This increase will be due to the requirement to shutdown the plant when steam generator water levels pass the maximum level allowed in the revised Technical Specification. The costs for incremental labor requirements associated with increased inspections are also included.

The costs associated with additional reactor shutdowns were calculated based on the increase in the frequency of feedwater overfeed transients. The transient frequency for PWR Category 19 was assumed to increase from the median value of 0.44/yr to the 95% upper bound value of 0.49/yr represented by the Chi-squared distribution (see NUREG/CR-3862 for descriptions of the transient categories). This represents approximately a 10% increase in these transients. The frequency of these transients for the B&W plants listed in Section 1.0 would then be increased from 0.13/yr (see Attachment 1) to about 0.15/yr. The average plant down-times for these transients were given in NUREG/CR-3862 at 17 hr. The estimated annual number of hours of replacement power is:

$$[(0.15/\text{RY} - 0.13/\text{RY}) * 17 \text{ hr}] = 0.34 \text{ hr}/\text{RY}$$

The annual per-plant costs for increased inspection and testing labor was estimated in NUREG/CR-4386 at approximately \$5500/RV for a similar safety issue resolution.

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

$$\begin{aligned} I_o &= [(0.34 \text{ hr}/\text{RY})/(24 \text{ hr}/\text{d})] * \$300,000/\text{d} + \$5500/\text{RY} \\ &= \$9750/\text{RY} \end{aligned}$$

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

$$\overline{NTI}_o = (\$9.75\text{E}+3/\text{RY})(10 \text{ plants})(28.2 \text{ yr}) = \$2.8\text{E}+6$$

12. Total Industry Cost (S_i):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$6.9E+6	\$9.4E+6	\$4.4E+6

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC resources for SIR development are assumed to be required to perform a "generic" safety analysis, similar to the plant-specific safety analyses that would be performed by the licensees. This analysis is presumed to be necessary because the OTSG conditions of interest to this issue have not been explicitly evaluated by the NRC. Therefore, approximately 3 man-yr of NRC resources, including contractor and NRC staff labor, are estimated to be required. Additional resources are also required to develop model Technical Specifications for the licensees to use in performing their safety analyses.

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

$$C_D = (3 \text{ man-yr})(\$100,000/\text{man-yr}) + \$21,000^8 = \$3.2\text{E}+5$$

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

NRC resources to support SIR development were assumed to be needed to review the plant-specific Technical Specification revisions and safety analyses. The labor requirements for these activities were estimated at 4 man-wk/plant.

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

$$C = (4 \text{ man-wk/plant})(\$2,270/\text{man-wk}) = \$9,080/\text{plant}$$

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

$$NC = (10 \text{ plants})(\$9,080/\text{plant}) = \$9.1\text{E}+4$$

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

NRC resources are assumed to be needed for an annual review of the OTSG operations to determine the effectiveness of the revised Technical

⁸ Source: Sciacca (1989).

Specifications. The resource requirements for this review would be minimal and are estimated to be about 0.5 man-wk/Ry.

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

$$C_o = (0.5 \text{ man-wk/Ry})(\$2,270/\text{man-wk}) = \$1,135/\text{Ry}$$

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

$$\bar{N}TC_o = (\$1,135/\text{Ry})(10 \text{ plants})(28.2 \text{ yr}) = \$3.2E+5$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$7.3E+5	\$9.6E+5	\$5.0E+5

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 106, Piping and Use of Highly Combustible Gases In Vital Areas

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: The occurrence of a release of process hydrogen and the consequences of subsequent explosion with respect to the integrity of safety related equipment is of concern. The proposed corrective action is the installation of excess flow valves in the main branches of the process hydrogen system.

AFFECTED PLANTS: PWR: Operating = 47 Planned = 0
 BWR: Operating = 1 Planned = 0

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 1.4E-01

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	2.6E-03

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	0.25
SIR Operation/Maintenance =	0.68
Total of Above =	0.93
Accident Avoidance =	2.2E-04

NRC COSTS:

SIR Development =	9.1E-03
SIR Implementation Support =	6.5E-02
SIR Operation/Maintenance Review =	0.60
Total of Above =	0.67

PIPING AND USE OF HIGHLY COMBUSTIBLE
GASES IN VITAL AREAS

ISSUE 106

1.0 SAFETY ISSUE DESCRIPTION

Combustible gases are used during normal operation of nuclear power plants. In PWRs, hydrogen is used during plant operation to scavenge oxygen from the primary system as a means of corrosion control. The hydrogen is piped to the volume control tank, located in the auxiliary building, which is a safety-related structure housing components of various safety systems. Leaks in this piping could result in combustible or explosive conditions in areas housing safety-related equipment. The potential disabling of safety related equipment due to combustible gas explosion/fire in an area housing such equipment is of concern. The AEOD has also raised a concern that some plants hydrogen detectors may be relied on too heavily to detect and alarm if leakage were to occur. This concern is based on the fact that these detectors are generally not qualified or supplied by an emergency power source. Therefore, they may not be functional during or following an event that could cause a hydrogen leak or cause a loss of ventilating systems that may have been diluting the gas from an existing but undetected leak.

PROPOSED SAFETY ISSUE RESOLUTION

Planned plants are required to satisfy Standard Review Plan (SRP) 9.5-1 which deals in part with hydrogen lines in safety related areas. Operating plants are not required to comply with SRP 9.5-1.

Installation of excess flow valves is the simplest retrofit procedure which would bring all operating plants using hydrogen into compliance with the requirements of SRP 9.5-1 with respect to hydrogen. It is proposed that excess flow valves be installed in the main branches of the hydrogen system as close to the source as possible. A PWR will therefore need to install two valves on the main hydrogen supply branches to the chemical volume control system and the waste gas system. A BWR is perceived to require one valve on the main supply branch to the coolant chemistry system.

AFFECTED PLANTS

Safety Issue Resolution (SIR) would affect operating plants using hydrogen and not already in compliance with SRP 9.5-1 with respect to hydrogen piping. Specifically this will include all operating PWRs and a single BWR, Dresden 2, which is installing a liquid hydrogen system as part of its water chemistry system.

2.0 SAFETY ISSUE RISK AND DOSE

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

TABLE 1 Public Risk Reduction Work Sheet

1. Title And Identification Number of Safety Issue:

Piping and Use of Highly Combustible Gases In Vital Areas (106)

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

	<u>N</u>	<u>$\bar{T}$ (yr)</u>
PWR	47	27.7
BWR	1	23 (Dresden 2)

3. Plants Selected for Analysis:

Oconee - representative PWR
Grand Gulf - representative BWR

4. Parameters Affected by SIR:

PWR: D, E
BWR: SCVA, SCVB

Note: These parameters are affected assuming that both residual heat removal (RHR) heat exchangers are disabled due to an explosion.

5. Base-case Values for Affected Parameters:

PWR: D = 0.02300045 E = 0.02300045
BWR: SCVA = 0.03200045 SCVB = 0.03200045

See Attachment 1 for details.

6. Affected Accident Sequences and Base-Case Frequencies:

Sequence Base-Case Frequency (/py)

PWR:

S1D	(PWR-1)	4.7E-08
	(PWR-3)	9.4E-07
	(PWR-5)	3.4E-08
	(PWR-7)	3.8E-06

S2D	(PWR-1)	9.8E-10
	(PWR-3)	2.0E-08
	(PWR-5)	7.1E-10
	(PWR-7)	7.8E-08

BWR:

T1QW (BWR-2) 5.1E-08

T23QW (BWR-2) 3.7E-07

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 = 5.3E-08/py

PWR-3 = 1.0E-06/py

PWR-5 = 3.7E-08/py

PWR-7 = 4.1E-06/py

BWR-2 = 4.2E-07/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.

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

$\bar{F}_{PWR} = 5.190391E-06/py$

$\bar{F}_{BWR} = 4.2294826E-07/py$

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

$W_{PWR} = 5.775201 \text{ man-rem/py}$

$W_{BWR} = 3.002932 \text{ man-rem/py}$

10. Adjusted-Case Values for Affected Parameters:

<u>Parameter</u>	<u>Adjusted Value</u>
PWR: D	0.023
E	0.023
BWR: SCVA	0.032
SCVB	0.032

11. Affected Accident Sequences and Adjusted-Case Frequency (/py):

<u>PWR</u>		
	(PWR-1)	4.7E-08
S1D	(PWR-3)	9.4E-07
	(PWR-5)	3.4E-08
	(PWR-7)	3.8E-06
	(PWR-1)	9.8E-10
S2D	(PWR-3)	2.0E-08
	(PWR-5)	7.1E-10
	(PWR-7)	7.8E-08

BWR

T1QW	(BWR-2)	5.1E-08
T23QW	(BWR-2)	3.7E-07

Note 1: 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.

Note 2: Differences between the Base and Adjusted case are not shown by two significant digits.

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 5.3E-08/py	PWR-3 = 1.0E-06/py
PWR-5 = 3.7E-08/py	PWR-7 = 4.1E-06/py

BWR-2 = 4.2E-07/py

Note 1: 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.

Note 2: Differences between the Base and Adjusted case are not shown by two significant digits.

13. Adjusted-Case, Affected Core-Melt Frequency (F*):

$F^*_{PWR} = 5.190290E-06/py$	$F^*_{BWR} = 4.2294232E-07/py$
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14. Adjusted-Case, Affected Public Risk (W*):

$W^*_{PWR} = 5.775088 \text{ man-rem/py}$	$W^*_{BWR} = 3.002890 \text{ man-rem/py}$
---	---

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

$\Delta\bar{F}_{PWR} = 1.0E-10/py$	$\Delta\bar{F}_{BWR} = 5.9E-12/py$
------------------------------------	------------------------------------

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

$\Delta W_{PWR} = 1.1E-04 \text{ man-rem/py}$	$\Delta W_{BWR} = 4.2E-05 \text{ man-rem/py}$
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17. Total Public Risk Reduction, ΔW (Total):

<u>Best Estimate</u> <u>(Man-rem)</u>	<u>Error Bounds (man-rem)</u> <u>Upper</u>	<u>Lower</u>
1.4E-01	2.3E+05	0

ATTACHMENT 1

Hydrogen is used in the normal operation of nuclear power plants to control the amount of dissolved oxygen in the coolant water. The hydrogen system is not explicitly accounted for in the existing probabilistic risk assessments. To incorporate the effect(s) of a hydrogen line break into the risk analysis, the pipe break frequency and consequences were examined.

Pipe break frequency per plant year was determined using an hourly frequency for lines <3", $1\text{E-}09/\text{hr-section}$. The approximate number of hydrogen pipe sections in the chemical volume control system and waste gas systems is 25. The annual pipe break frequency is therefore $(1\text{E-}09 \text{ breaks/hr-section})(25 \text{ sections}) = 2.5\text{E-}08 \text{ breaks/hr}$. Given that a pipe break occurs, assume that the normal heating, ventilation and air conditioning (HVAC) systems of the Auxiliary building fails to prevent accumulation of an explosive level of hydrogen (4 percent to 74 percent) with a frequency of 0.25. Further assume that, if such an accumulation occurs, a detonation occurs in all cases. Thus, the probability of a hydrogen explosion is given by $(2.5\text{E-}08)(0.25)(1) = 6.25\text{E-}09/\text{hr}$. For purposes of this evaluation, the frequency of explosion was used to calculate the unavailability of the residual heat removal (RHR) heat exchangers.

Damage to safety related equipment though conceivable is difficult to assess because detailed information is needed but the level of detail is beyond the scope of this issue analysis. The detailed information that is needed relates to plant layout, ventilation capacities and flow patterns including location of quiescent flow zones, location of safety equipment relative to the explosion location, and blast mitigation by venting (e.g., HVAC ducts, branching corridors), as well as dissipation via radiation shielding and fire walls. A blast is not a static load, however it could be idealized as a uniformly applied impact loading. There is evidence that under such an idealized loading condition the impact strength of the concrete is significantly greater than the static compressive strength (Neville 1971). This phenomena would explain the ability of concrete to absorb strain energy (Neville 1971) and thereby serve as an effective blast energy sink. Preliminary calculation and investigation indicate that a hydrogen explosion would probably cause severe damage to concrete walls and other structures within approximately a 30 foot radius of the blast center. This estimation does not account for energy dissipation by venting, etc.

For this analysis, it is conservatively assumed that if an explosion occurs then the RHR heat exchangers are disabled. The Technical Specifications will allow the plant to operate with both trains of the RHR system disabled for only 72 hours before the plant must begin a controlled shutdown. This 72 hours is the time in which the RHR system would be unavailable should an accident sequence be initiated. Therefore to calculate the unavailability of the RHR heat exchangers, the frequency of explosions ($6.25\text{E-}09/\text{hr}$) is multiplied by the time of unavailability (72 hours). Thus the additional unavailability of the RHR heat exchangers is $4.5\text{E-}07$. This additional unavailability is added to the unavailability of the RHR heat exchangers as addressed in the representative plant risk assessments. For Oconee, the terms that best represent the RHR heat exchanger unavailability are terms D and E. For Grand Gulf, the terms that best represent the RHR heat exchanger unavailability are terms SCVA and SCVB.

Additional radiation release due to damage to the Auxiliary building is not included in this analysis.

The resolution for this issue is assumed to reduce the unavailabilities of RHR heat exchangers to the original values given in the respective risk assessments.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Piping and Use of Highly Combustible Gases in Vital Areas (106)

2. Affected Plants (N):

PWRs: Operating 47
BWRs: Operating with Hydrogen Water Chemistry System

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

	<u>$\bar{T}$ (yr)</u>
PWR: Operating	27.7
BWR: Operating	23 (Dresden 2)

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

PWR: (19,900 man-rem)(1.0E-10/py) = 2.0E-06 man-rem/py
BWR: (19,900 man-rem)(5.9E-12/py) = 1.2E-07 man-rem/py

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

<u>Best Estimate</u>	<u>Error Bounds (man-rem)</u>	
<u>(man-rem)</u>	<u>Upper</u>	<u>Lower</u>
2.6E-03	807	0

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

Installation, operation and maintenance of the excess flow valves will not involve work in a radiation zone. The valves should be installed prior to the hydrogen line entering either the Auxiliary buildings. Therefore, there is no occupational dose increase due to this Sir, and D=Do=G=0.

3.0 SAFETY ISSUE COSTS

The cost estimations do not include cleanup and repair of the portion of the Auxiliary Building which might be damaged as a result of the hydrogen explosion.

The costs including NRC costs are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Piping and Use of Highly Combustible Gases In Vital Areas (106)

2. Affected Plants (N):

PWRs: Operating 47
BWRs: Operating 1

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

	<u>$\bar{T}$ (yr)</u>
PWR: Operating	27.7
BWR: Operating	23 (Dresden 2)

Industry Costs (Steps 4 through 12)

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

PWR: $(\$1.65E+09)(1.0E-10/\text{py}) = \$0.17/\text{py}$
BWR: $(\$1.65E+09)(5.9E-12/\text{py}) = \$0.0097/\text{py}$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$220	\$6.7E+07	0

6. Per-Plant Industry Resources for SIR Implementation:

It is assumed that the excess flow valves are installed during scheduled down time. Further the resource breakdown is presented on a per valve basis. The difference between PWR and BWR resources required depends on the number of valves installed.

Labor

Hardware design and review	2 days
Procurement	1 day
Preinstallation check	0.5 hr/valve
Installation	
Estimate from craft services:	
1 man-day/valve (welder)	
1 man-day/valve (fitter)	
Post-installation check	1.5 hr/valve
Documentation	0.5 day
	<u>3.5 days + 2.25 hr/valve</u>

Equipment

Install a valve in the main branches of hydrogen lines entering safety related structures.

Additional Power

Assume all valves are installed during scheduled downtime.

PWR: Equipment 2 valves *
Labor 8 days (3.5 days + 2(2.25 days/valve))
Add. Power 0

* Install valves in Chemical-Volume Control System and Waste Gas Decay System.

BWR: Equipment 1 valve *
Labor 5.75 days (3.5 days + 1(2.25 days/valve))
Add. Power 0

* Install valve in Hydrogen Water Chemistry System.

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

PWR: Labor	(8 days)(wk/5 days)(\$2270/wk)	= \$ 3632
Equipment	(2 valves)(\$870/valve)	= 1740
Add Power		= 0
	Total:	= <u>\$ 5372</u>

BWR: Labor	(5.75 days)(wk/5 days)(2270/wk)	= \$ 2611
Equipment	(1 valve)(\$870/valve)	= \$ 870
Add. Power		= 0
	Total:	= <u>\$ 3481</u>

Note: The cost of a valve is the average of two vendor quotes for an excess flow valve for a 2 inch hydrogen line.

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

PWR	NI = (47 plants)(\$5372/plant)	= \$ 2.5E+05
BWR	NI = (1 plant)(\$3481/plant)	= <u>\$ 3.5E+03</u>
	Total:	= <u>\$ 2.5E+05</u>

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

Per vendor estimate: semiannual check of valve to assure the shaft is not "frozen." Replacement of diaphragm as needed (frequency depends on valve environment). For purposes of this estimate, the diaphragm is assumed to be replaced every seven years at a labor requirement of 0.5 day.

Semiannual check	2 hr/valve-check
Diaphragm replacement	(average life/7 yr)(4 hr/valve)

PWR: 2 valves
Semiannual check (2 hr/valve-check)(2 valves)(2 checks/yr)
= 8 hr/yr
Diaphragm replacement (27.7yr/7 yr)(4 hr/valve)(2 valves)
= 31.66 hr for plant life or 1.14 hr/yr

Annual total = 9.14 hr/yr

BWR: 1 valve
Semiannual check (2 hr/valve-check)(1 valve)(2 checks/yr)
= 4 hr/yr
Diaphragm replacement (23yr/7 yr)(4 hr/valve)(1 valve)
= 13.14 hr for plant life or 0.57 hr/yr

Annual total = 4.57 hr/yr

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

$$I_o(\text{PWR}) = (9.14 \text{ hr/yr})(\text{wk}/40 \text{ hr})(\$2270/\text{man week}) = \$519$$

$$I_0(\text{BWR}) = (4.57 \text{ hr/yr})(\text{wk}/40 \text{ hr})(\$2270/\text{man week}) = \$259$$

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

$$NTI_0 = \$6.8E+05$$

12. Total Industry Cost (SI):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$9.3E+05	\$1.3E+06	\$5.7E+05

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

- Formulate guideline and documentation requirements - 2 man-weeks
 - Respond to industry questions - 0.75 man-weeks
 - Finalize implementation and enforcement - 1.25 man-weeks
- Total - 4 man-weeks

14. Total NRC Cost for SIR Development (C_n)

$$C_n = (4 \text{ man-weeks})(\$2270/\text{man-week}) = \$9,080$$

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

- Review plant implementation, operation and maintenance schedule - 1 day
 - Documentation - 1 day
 - Enforcement of Implementation - 1 day
- Total: 0.6 man-week/plant

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

$$C = (0.6 \text{ man-week/plant})(\$2270/\text{man-week}) = \$1362/\text{plant}$$

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

$$NC = (\$1362/\text{plant})(48 \text{ plants}) = \$6.5E+04$$

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

Review and record appropriate documentation of semiannual tests.

$$(0.5 \text{ day/plant test})(2 \text{ tests/yr}) = 1 \text{ day/py}$$

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

$$C_o = (1 \text{ day/py})(\text{man-week}/5 \text{ days})(\$2270/\text{man-week}) = \$454/\text{py}$$

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

$$N\bar{C}_o = (\$454/\text{py})((47)(27.7) + (1)(23)) = \$6.0E+05$$

21. Total NRC Cost (S_N)

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$6.7E+05	\$9.8E+05	\$3.7E+05

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 107, Main Transformer Failures

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: The concern of this issue is the generic safety implications associated with main transformer failures at U.S. light water reactors (LWRs). Safety issue resolution involves licensee reviews and evaluations of main transformer problems and enhancement of fire protection capabilities where necessary.

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

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 2.6E+03

OCCUPATIONAL DOSES:

SIR Implementation = 0
SIR Operation/Maintenance = 0
Total of Above = 0
Accident Avoidance = 17

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation = 4.3
SIR Operation/Maintenance = -0.4
Total of Above = 3.9
Accident Avoidance = 0.9

NRC COSTS:

SIR Development = 0.08
SIR Implementation = 0.74
SIR Operation/Maintenance Review = 3.4
Total of Above = 4.2

MAIN TRANSFORMER FAILURES

ISSUE 107

1.0 SAFETY ISSUE DESCRIPTION

The principal concern of this issue is the potential safety implications associated with main transformer failures at U.S. light water reactors (LWRs). Concern for this issue arose when the North Anna Power Stations had seven main transformer failures in 26-months. Five of these resulted in reactor trips. Of the seven failures, three included rupture of the transformer tank with two fires occurring. One of the fires spread beyond the transformer bay to the turbine bay.

Safety-related loads are supplied from buses that can be supplied from any one of the following at the licensee's choice: (1) auxiliary transformer; (2) startup transformer (or reserve auxiliary transformer); or (3) backup power supply (e.g. diesel generators). A main transformer failure will result in a loss of load or unbalanced load on the generator. A generator/turbine trip would result and power would not be available to the auxiliary transformer for station power. Station power would be obtained from the grid through the startup transformer or from backup power sources. Switchyards have redundant and duplicate systems to provide sufficient relaying and circuit breakers so that a failure would not be expected to cause a loss of offsite power. In the event of a failure in the switchyard to provide power to the startup transformer, power can be backfed through the main transformer to the auxiliary transformer.

An assessment of the main transformer failures at North Anna concluded that there was a possibility of generic implications arising out of plant-specific characteristics (Dalton, Kresser, Savage, and Selan, 1982). ANO-1 also experienced main transformer failures that resulted in three oil spills from tank ruptures and one fire. Some of the generic issues addressed were transformer fires, transformer maintenance and operational procedures, excessive shipping and handling, cascading effects, electrical/mechanical damage and explosions. Generic issues regarding transformer fires included addressing the fire protection system, overhead conductors and buses, cable trays, storage of flammable material near potential fire hazards, and oil filled transformers in general. Some of the issues that were identified included that the oil from a ruptured transformer will float on the water used by fire protection system such that the fire will move in the direction of drainage, the fire may propagate to overhead cables and buses, and the need for access to adjacent locations (such as building roofs) when fighting the fire.

PROPOSED RESOLUTION

For purposes of this analysis, this proposed safety issue resolution (SIR) is assumed to involve the following actions:

1. Licensees should evaluate their main transformer to ensure that the offsite power is protected. Design requirements should be established for routing and separation of offsite power source feeds to protect against loss from a transformer fire.

2. Fire protection system for the main transformer should be reviewed for adequacy and revised as necessary to assure that potential fire is prevented from spreading to other plant areas. The review should address the deluge system, drainage system, fire barriers, and fire fighting equipment and procedures.
3. Maintenance and operating procedures for the main transformer should be reviewed for adequacy and revised as necessary.
4. Modify, if necessary, the drainage system to provide drains for each transformer so that liquids flow away from the turbine building, power lines and safety-related cables to or within the reactor and related safety equipment. Modifications could include adding drains, building dikes and sloping transformer yard away from buildings and other transformers.
5. Modify, if necessary, the fire-fighting equipment and procedures. This may include longer hoses, increased ease of access to building roofs, mobility of fire-fighting equipment, and training for personnel.
6. Move, if necessary, the power lines to the safety-related buses so that they would not be affected by a fire in the transformer bays.

AFFECTED PLANTS

The resolution of this safety issue is assumed to affect all operating and planned LWRs as listed in NUREG/CR-2800 (Andrews, et al., 1983).

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with the issue resolution are estimated in this section. Results are summarized in Tables 1 and 2, respectively. The analyses are conducted for a representative PWR and a representative BWR. The Oconee-3 PRA (Andrews, et al., 1983, Appendix A) risk equations have been assumed to be representative of all PWRs for this analysis. The Grand Gulf-1 PRA (see Andrews, et al., 1983, Appendix B) was used to derive estimates of core-melt frequency reduction and occupational dose increase for BWRs.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Main Transformer Failures, Issue 107

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

	<u>N</u>	<u>$\bar{T}$ (yr)</u>
All PWRs	90	28.8
All BWRs	44	27.4
All plants	134	28.3

3. Plants Selected for Analysis:

A PWR and a BWR are assumed to be representative of each type of reactor. The safety analyses are conducted for Oconee-3 for PWRs and Grand Gulf 1 for BWRs.

4. Parameters affected by SIR:

For the representative PWR, the primary affected parameter is:

T_2 Loss of power conversion system transient caused by other than loss of offsite power

For the representative BWR, the primary affected parameter is:

T_{23} Any transient other than loss of offsite power which requires an emergency reactor shutdown.

5. Base-Case Values for Affected Parameters:

For the representative PWR, the base-case frequency is:

$$T_2 = 3/\text{py}$$

For the representative BWR, the base-case frequency is:

$$T = 7.0$$

23

6. Affected Accident Sequences and Base-Case Frequencies:

Oconee-3:

T_2 MLU	τ (PWR-3) =	6.0E-7
	β (PWR-5) =	8.8E-9
	ϵ (PWR-7) =	6.0E-7
T_2 MQH	τ (PWR-3) =	5.5E-6
	β (PWR-5) =	8.0E-8
	ϵ (PWR-7) =	5.5E-6
T_2 MQFH	τ (PWR-2) =	2.5E-6
	β (PWR-4) =	3.7E-8
	ϵ (PWR-6) =	2.5E-6
T_2 MLUO	τ (PWR-3) =	4.1E-6
	β (PWR-5) =	5.9E-8
	ϵ (PWR-7) =	4.1E-6
T_2 KMU	τ (PWR-3) =	3.9E-6
	β (PWR-5) =	5.7E-8
	ϵ (PWR-7) =	3.9E-6
T_2 MQD	τ (PWR-3) =	7.5E-7
	β (PWR-5) =	1.1E-8
	ϵ (PWR-7) =	7.5E-7

Grand Gulf:

T_{23} PQI	α (BWR-1) =	3.7E-8
	δ (BWR-2) =	3.7E-6
T_{23} PQE	τ (BWR-3) =	2.7E-7
	δ (BWR-4) =	2.7E-7
T_{23} QW	δ (BWR-2) =	1.2E-5
T_{23} C	δ (BWR-2) =	5.4E-6

7. Affected Release Categories and Base-Case Frequencies:

Oconee:

Grand Gulf:

<u>Category</u>	<u>Frequency, py^{-1}</u>	<u>Category</u>	<u>Frequency, py^{-1}</u>
PWR-2	2.5E-6	BWR-1	3.7E-8
PWR-3	1.5E-5	BWR-2	2.1E-5
PWR-4	3.7E-8	BWR-3	2.7E-7
PWR-5	2.2E-7	BWR-4	2.7E-7
PWR-6	2.5E-6		
PWR-7	1.5E-5		

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

$$\bar{F}_{PWR} = 3.776E-5/\text{py} \quad \bar{F}_{BWR} = 2.206E-5/\text{py}$$

Note: These values were calculated using the Grand Gulf-1 and Oconee-3 PRA mini-computer codes. The values contain excess significant figures in order to compute the difference between the base-case and adjusted-case.

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

$$W_{PWR} = 98.830 \text{ man-rem/py} \quad W_{BWR} = 153.909 \text{ man-rem/py}$$

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

As discussed in Attachment 1, SIR is assumed to reduce the frequency of transients at PWRs and BWRs by enhancing the reliability of main transformers and preventing potential fires that may result from main transformer failures from spreading to other vital areas of the plant. To model the effects of SIR on plant risk, the transient frequencies given in NUREG/CR-2800 (Andrews, et al., 1983) were modified to reduce the transient frequencies by the amount equivalent to the frequency of outages caused by main transformer failures.

For PWRs, the adjusted case value is:

$$T_2 = 3/\text{py} - 0.023/\text{py} = 2.977/\text{py}$$

For BWRs, the adjusted case value is:

$$T_{23} = 7/\text{py} - 0.023/\text{py} = 6.977/\text{py}$$

11,12. Steps Leading to Calculation of Adjusted-Case Affected Accident Sequence Frequencies and Adjusted-Case Frequencies for Affected Release Categories:

The Oconee-3 and Grand Gulf-1 mini-computer codes were used to calculate the adjusted-case affected core-melt frequencies and public risk values and the changes in core-melt frequency and public risk associated with SIR.

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

$$\bar{F}^*_{PWR} = 3.747E-05/\text{py} \quad \bar{F}^*_{BWR} = 2.199E-05/\text{py}$$

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

$$W^*_{PWR} = 98.072 \text{ man-rem/py} \quad W^*_{BWR} = 153.404 \text{ man-rem/py}$$

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

$$\Delta\bar{F}_{PWR} = 2.89E-7/\text{py} \quad \Delta\bar{F}_{BWR} = 7.25E-8/\text{py}$$

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

$$\Delta W_{PWR} = 0.76 \text{ man-rem/py}$$

$$\Delta W_{BWR} = 0.51 \text{ man-rem/py}$$

17. Total Public Risk Reduction, ΔW (Total):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
2.6E+03	9.6E+6	0

ATTACHMENT 1 (to Table 1)

Resolution of Issue 107 uses the Oconee-3 and Grand Gulf-1 PRAs as described by Andrews, et al., (1983) as the basis for evaluation of this issue. Several PRAs were reviewed to determine whether main transformer failures were addressed in any of the dominant cut sets. Because failure rates for transformers are relatively low (on the order of 10^{-6} per hour), their failures are not a significant cause of transients. As a result, main transformer failures are integrated into a category of transients that result from loss of network load. The affected parameters, T_2 for Oconee and T_{23} for Grand Gulf, were then adjusted to determine the public risk and core-melt frequency reductions associated with Safety Issue Resolution (SIR).

It is postulated that SIR will enhance the reliability of the main transformers and thus reduce the frequency of transients associated with main transformer failures. The frequency of occurrence of reactor transients provided in NUREG/CR-3862 (Mackowiak, et al., 1985) was used to estimate the reduction in transient frequencies associated with SIR. This document categorizes reactor transients into 41 categories for PWRs and 37 categories for BWRs. One of the BWR categories is for transients caused by loss of auxiliary power, characterized as a loss of incoming power to a plant as a result of onsite failures (such as failure of an auxiliary transformer). This category closely resembles failure of the main transformer. The transient frequency associated with this category is given as 0.02 event/py.

Licensee Event Reports (LERs) reviewed contained three of the seven main transformer failures at the North Anna Power Station. No other failures of main transformers were identified in the LERs. The IEEE reliability data for liquid filled transformers (347 to 550 kVA) at nuclear power generating stations are shown below. The sum of the failure rate for all failure modes as given by IEEE is 2.67 per 10^6 hours. The corresponds to an annual frequency of 0.023 failures per year for main power generators or unit transformers. This value is not significantly different than the transient frequency given above and will be used as the basis for reductions in main transformer failures that are postulated to result from SIR.

<u>Failure Modes</u>	<u>Failure Rate/10^6 Hours</u>		
	<u>Low</u>	<u>Recommended</u>	<u>High</u>
Single Phase Liquid Filled			
All Modes	0.74	1.62	2.67
- Catastrophic	0.53	1.16	1.91
- Degraded	0.094	0.21	0.34
- Incipient	0.12	0.25	0.42

<u>Failure Modes</u>	<u>Failure Rate/10⁶ Hours</u>		
	<u>Low</u>	<u>Recommended</u>	<u>High</u>
Three Phase Liquid Filled			
All Modes	0.78	1.35	2.61
- Catastrophic	0.43	0.74	1.44
- Degraded	0.17	0.29	0.56
- Incipient	0.18	0.32	0.61

A second aspect of main transformer failures addressed here is the potential for fires. Fires are of concern because of their potential to damage or otherwise degrade the performance of one or more safety systems. For example, fires could result in spurious actuation of valves, generate false instrument readings in the control room, and produce mechanical and thermal damage to safety-related components. Main transformer fires could potentially spread to overhead electric power cables that supply AC power to safety buses and, depending upon their proximity to the main transformer, could damage the reactor or turbine buildings. As a result, a single fire could cause malfunctions of various components that receive electric power from the fire-damaged cables. The proposed SIR attempts to prevent a main transformer fire from spreading to other vital areas of the plant, including electric power and control cables.

The effects of transformer fires on public risk are difficult to quantify because of the limited treatment of fires given to date in PRA studies. The detailed Oconee-3 PRA (Sugnet, et al., 1984) was reviewed to develop insights on the potential risks associated with main transformer fires. The analysis of fires at Oconee-3 included an attempt to identify critical locations where fires could result in an initiating event and, at the same time, cause failure of redundant safety-related components. The main transformers were not among the critical locations and thus analysis of main transformer fires was not performed. Presumably, this was for the following reasons:

- Main transformer fires that spread to power cables would result in a loss of AC power to safety-related systems. These systems are provided with backup DC power supplies and emergency diesel generators that would likely be unaffected by the fire and thus the plant would be capable of recovering from the loss of normal AC power. Offsite power could also be switched to the auxiliary transformers in the event of a fire involving the main transformers. Thus, three separate sources of AC power would be available in the event that a fire disables the main transformer and damages power cables. Combining the probabilities of failure for these backup power sources and the probability of occurrence of a fire provides evidence that a fire-induced common mode failure of all electric power is extremely low.
- Main transformers are typically provided with fire detection and extinguishment systems so it is expected that a fire in this area would be of relatively short duration. The purpose of the SIR proposed here is to improve the fire extinguishment system and emergency procedures such that all plants are capable of providing a rapid and effective response to fires.

- Fires in other areas of the plant, such as the cable spreading room, control room, and electrical equipment room, appear to create situations in which a plant is more vulnerable to common cause failures than fires involving the main transformer.

For these reasons, the risks associated with main-transformer-fire-induced common cause failures of electric power supply systems is anticipated to be very low. Further, the risk reduction associated with the portions of the proposed SIR involving enhanced fire protection systems is not quantified here.

In order to evaluate the adjusted case core-melt frequency as a result of the SIR implementation, it was assumed that the transient initiating event frequencies would be reduced, at maximum, by the annual frequency of the transformer failures. The annual frequency of transformer failures was computed previously as 0.023 event/yr, based on IEEE reliability data.

Affected accident sequences for Oconee-3 include all sequences involving initiating event T_2 and those for Grand Gulf-1 include all sequences involving the initiating event T_{23} . These accident sequences and their base-case frequencies are presented in Table 1.

The adjusted-case, affected core-melt frequency for Oconee-3 is calculated by replacing the base-case frequency for initiating event T_2 of 3/py by the adjusted-case T event frequency of $(3/\text{py} - 0.023/\text{py}) = 2.977/\text{py}$. The adjusted-case transient frequency was input to the Oconee-3 minicomputer code to calculate the adjusted-case core melt frequency and public risk. The adjusted-case, affected core-melt frequency for the representative BWR is calculated by replacing the base-case frequency for initiator T_{23} of 7/py by the adjusted-case T event frequency of 6.977/py. The Grand Gulf mini-computer code was used to calculate the reductions in core-melt frequency and public risk. The results of these calculations are presented in Table 1.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Main Transformer Failures, Issue 107

2. Affected Plants (N):

All 134 plants (71 operating and 63 planned) are assumed to be affected. This includes 90 PWRs and 44 BWRs.

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

	<u>$\bar{T}$ (yr)</u>
All PWRs	28.8
All BWRs	27.4
All plants	28.3

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

Using 19,860 man-rem for D_R , then

PWR: $\Delta(\bar{F}D_R) = (19,860 \text{ man-rem})(2.89\text{E-}7/\text{py}) = 5.7\text{E-}3 \text{ man-rem/py}$
 BWR: $\Delta(\bar{F}D_R) = (19,860 \text{ man-rem})(7.25\text{E-}8/\text{py}) = 1.4\text{E-}3 \text{ man-rem/py}$

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

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
1.7E+1	1.5E+4	0

6 to 12. Steps Leading to Total Occupational Exposures for SIR Implementation, Operation, and Maintenance

SIR is assumed to not involve any labor in radiation zones. This is because the main transformers are not located in a building in which radioactive materials are used or stored and thus the radiation dose rates would be zero. SIR does not require any entries into containment or into the reactor building. As a result, total occupational exposures for SIR implementation, operation, and maintenance are 0.

3.0 SAFETY ISSUE COSTS

The industry and NRC costs associated with resolution of this safety issue 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:

Main Transformer Failures, Issue 107

2. Affected Plants (N):

All 134 plants (71 operating and 63 planned) are assumed to be affected.

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

	<u>$\bar{T}$ (yr)</u>
Operating:	23.1
Planned:	30
All plants:	28.3

Industry Costs (Steps 4 through 12):

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

PWR: $\Delta(\bar{F}A) = (8.0E-07/\text{py})(\$1.65E+09) = \$287/\text{py}$

BWR: $\Delta(\bar{F}A) = (8.0E-07/\text{py})(\$1.65E+09) = \$88/\text{py}$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$8.5E+05	\$5.0E+08	0

6. Per-Plant Industry Resources for SIR Implementation:

For all operating plants, it is assumed that the NRC would issue a generic letter or bulletin requiring all plants to review the design and installation of main transformers and associated fire protection systems, control circuits, and operating and maintenance procedures. The assumed resource requirements for this review are:

Labor =	2	man-weeks/plant to evaluate fire protection system(s)
	1	man-weeks/plant to review protective circuitry
	4	man-weeks/plant to review operating and maintenance procedures
	2	man-weeks/plant to revise operating and maintenance procedures
	<u>2</u>	<u>man-weeks/plant to revise staff training</u>
	<u>11</u>	<u>man-weeks/plant for SIR Implementation</u>

As a result of the reviews conducted at all plants, it is assumed that 10% of the plants would require modifications to the fire protection system and rerouting of cables around the main transformer areas. The assumed resource requirements for the plants requiring modifications are:

Labor =	3	man-weeks/plant to design modifications
	2	man-weeks/plant to plan installation and testing
	2	man-weeks/plant to revise procedures
	<u>2</u>	<u>man-weeks/plant for acceptance testing</u>
	<u>9</u>	<u>man-weeks/plant for SIR Implementation</u>

Additional hardware requirements for those plants requiring modifications include: an additional drain, gravel and concrete to slope the area around the transformers and construct dikes; additional power lines to route power to the buildings; additional breakers to protect equipment connected to the auxiliary transformer; and longer fire hoses. The hardware and installation labor costs for the plants requiring modifications are itemized below:

Dike (250-ft long, 4-ft high)	.	.	.	.	.\$ 3,750
Concrete and gravel	.	.	.	.	. 15,820
Power lines (1,000 ft)	.	.	.	.	. 5,000
Poles (10 @ \$1085 each)	.	.	.	.	. 11,000
Breakers (2 @ \$2,500 each)	.	.	.	.	. 5,000
Fire hose/storage cabinet (100-ft)	.	.	.	.	. 500
Subtotal	.	.	.	.	. 41,000
Escalation (1982 to 1988)	.	.	.	.	. 1.18
Total (1988 dollars)	.	.	.	.	. <u>48,000</u>

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

All plants (review and evaluation)			
Labor =	(11 man-wks)(\$2270/man-wk)	=	\$ 25,000
10% of plants (hardware modifications)			
Labor =	(9 man-wks)(\$2270/man-wk)	=	20,400
Hardware		=	<u>48,000</u>
Total:		=	<u>68,400</u>

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

NI =	(134 plants)(\$25,000/plant) + (14 plants)(\$68,400/plant)
	= \$4.3E+6

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

All plants:

Labor = 0.2 man-wk/py for periodic review of main transformer procedures, operations, and maintenance

Plants requiring hardware modifications:

Labor = 1.0 man-wk/py for periodic maintenance/inspection of drains and new diked areas; removal of trash from drains; etc.

Improvements to the reliability of main transformers and improvements to the fire protection systems could potentially result in avoided costs of replacing a transformer damaged by a fire. This avoided cost was estimated as follows. First, SIR improves the reliability of main transformers an amount equivalent to 0.023 failures per reactor-year. A review of LER data indicated that 14 main transformer failures (7 at ANO-1 and 7 at North Anna) resulted in 3 fires for a conditional probability of 0.2 that a main transformer failure results in a fire. Thus, the potential exists for avoidance of 0.005 main transformer fires/py. These avoided costs apply only to the 10% of the plants that were assumed to identify deficiencies in their fire protection systems. The remaining 90% of the plants are assumed to be adequately prepared to prevent serious damage to the transformers that would result from a fire. The estimated replacement costs for a large transformer is \$14,000, including installation.

Avoided replacement power costs associated with reducing the number of reactor trips per year caused by main transformer failures were also addressed. As above, SIR is postulated to reduce the frequency of transformer failures by 0.023 failures/py. A review of the LER data showed that of the 14 main transformer failures at ANO-1 and North Anna, 5 resulted in reactor trips. This is a conditional probability of 0.36 that a main transformer failure results in a reactor trip. Assuming each shutdown lasts three days, the avoided replacement power days are equal to 0.025 days/py. Using an average replacement power cost of \$3E+5/day (Andrews, et al., 1983), the avoided annual costs are about \$7.5E+3/py.

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

For all plants:

$$I_o = (0.2 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$450/\text{py}$$

For plants requiring hardware modifications:

$$I_o = (1.0 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$2270/\text{py}$$

Avoided costs for transformer replacement:

$$I_o = -(0.005 \text{ transformer fires/py})(\$14,000/\text{transformer}) \\ = -\$70/\text{py}$$

Avoided replacement power costs:

$$I_o = -\$7,500/\text{py}$$

Note: Negative sign (-) indicates avoided costs.

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

$$\begin{aligned} \text{NTI}_o &= (134 \text{ plants})(\$450/\text{py})(28.3 \text{ py}) + (14 \text{ plants})(\$2270/\text{py})(28.3 \text{ py}) \\ &\quad - (14 \text{ plants})(28.3 \text{ yr})(\$70/\text{py}) - (14 \text{ plants})(28.3 \text{ yr})(\$7,500/\text{py}) \\ &= -\$3.7\text{E}+5 \end{aligned}$$

12. Total Industry Cost (S_T):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$3.9E+06	\$6.1E+06	\$1.7E+06

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development

The NRC costs for developing the SIR include four man-weeks to issue a generic letter or bulletin to the licensees (includes technical, legal, and administrative staff support) 6 man-months to review licensee responses to the letter, assess the differences between plant designs, and research potential implementation measures (assumed to be provided by a contractor), and 4 man-wks of NRC technical staff labor to monitor the contractor. SIR development also includes issuance of revised design guidance to the licensees related to adequate main transformer designs and procedures. It is estimated that an additional 6 man-wks of NRC technical, legal, and administrative staff labor are needed to develop, approve, and issue the revised guidance.

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

$$\begin{array}{rcl} \text{Labor} & (14 \text{ man-weeks})(\$2,270/\text{man-week}) & = \$3.2\text{E}+04 \\ \text{Contract Support} & & + \underline{5.0\text{E}+04} \\ C_o & & = \underline{8.2\text{E}+04} \end{array}$$

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

NRC labor to support SIR implementation consists of reviewing utility plans to comply with the revised guidance plus an onsite inspection by resident inspectors to review the plans. The labor requirements are:

Review and approval of license's plans	.	2 man-wks/plant
Onsite inspection	.	0.4 man-wks/plant
Total	.	<u>2.4 man-wks/plant</u>

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

$$C = (2.4 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$5.5\text{E}+03/\text{plant}$$

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

$$\text{NC} = (134 \text{ plants})(\$5.5\text{E}+03/\text{plant}) = \$7.4\text{E}+05$$

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

NRC labor to review SIR operation and maintenance is assumed to be primarily integrated with other NRC inspection activities. However, additional labor is assumed to be needed for enhanced reviews of main transformer testing/maintenance programs, operability of the fire protection system, and the effectiveness of revised hardware configurations. The NRC labor requirements for these enhanced reviews are estimated at about 2 man-days per plant per year.

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

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

20. Total NRC Cost for Review of SIR Operation and Maintenance ($\overline{N\overline{TC}}_o$):

$$\overline{N\overline{TC}}_o = (134 \text{ plants})(28.3 \text{ yr})(\$908/\text{py}) = \$3.4\text{E}+06$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$4.2E+06	\$5.9E+06	\$2.5E+06

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 110, Equipment Protective Devices on Engineered Safety Features

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: Some protective devices intended to trip active engineered safety features (ESF) components, under indications of equipment faults, are overridden by ESF actuation signals. The safety issue resolution will allow the tripping of the active ESF components and subsequent corrective actions to prevent damage to ESF components.

AFFECTED PLANTS: All plants (BWRs - 44; PWRs - 90)

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 1.1E+4

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	310
Total of Above =	310
Accident Avoidance =	72

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	48
SIR Operation/Maintenance =	18
Total of Above =	66
Accident Avoidance =	6

NRC COSTS:

SIR Development =	0.2
SIR Implementation =	1.2
SIR Operation/Maintenance Review =	8.6
Total of Above =	10

EQUIPMENT-PROTECTIVE DEVICES ON ENGINEERED SAFETY FEATURES

ISSUE 110

1.0 SAFETY ISSUE DESCRIPTION

Some (but not all) protective devices that are intended to trip engineered safety features (ESFs) components under indications of equipment faults are overridden by ESF actuation signals. Examples can be found in the protective trips of emergency diesel generators. The rationale has been that in a genuine demand, it is better to risk self-destruction of the ESF rather than tolerate spurious trips in critical ESF challenges. However, PRAs are showing that, in truly critical challenges of ESFs, there is often a long time before a point of no return is reached. This time period allows ample opportunity to diagnose the fault and override the protective trip if necessary. PRAs are also showing the importance of support system failures (e.g., common-cause failure of jacket cooling water to emergency diesel generators). It appears that it would be safer to trip the diesel generator off than allow self-destruction due to support system unavailability. The trip off could enable repair and restart of the diesel generators without equipment damage. Water cooling of containment fan cooler motors and provisions to override torque limitations on MOVs may be other examples. A related problem may be the location of reset buttons for some equipment protective trips. It has been suggested that the reset for the over-current or over-heat trip of containment fan cooler motors may, in some plants, be located in containment where the reset button would be inaccessible in a genuine demand situation. It may be advisable to consider the question of whether equipment protective trips of ESF components should be bypassed by ESF actuation signals or not, and whether the reset capabilities are appropriate in light of what has been learned from PRAs about the dominant challenges to active ESF reliability.

PROPOSED SAFETY ISSUE RESOLUTION

The safety issue resolution (SIR) allows tripping of active ESF components at or after ESF actuation. Corrective actions in response to the trip signal are then taken to prevent damage to the ESF components without increasing the severity of plant transients.

AFFECTED PLANTS

This safety issue affects all 90 PWRs and all 44 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 Worksheet

1. Title and Identification Number of Safety Issue:

Equipment-Protective Devices on Engineered Safety Features (110).

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

All PWRs and BWRs are assumed to be affected.

	<u>N</u>	<u>$\bar{T}$ (yr)</u>
PWRs	90	28.8
BWRs	44	27.4
All	134	28.3

3. Plants Selected for Analysis:

Representative PWR: Oconee 3

Representative BWR: Grand Gulf 1

4. Parameters Affected by SIR:

Oconee - D, E, CONST1, CONST2, B1, (B3), and F1

Grand Gulf - DIESEL1, DIESEL2, DIESEL3, H, R, L, LA2, LB2, LC, PA27, PB27, SSA, SSB, and SSC.

See Attachment 1 for discussion.

5. Base-Case Values for Affected Parameters:

Oconee:

D	=	2.357E-2	F1	=	1.87E-3
E	=	2.357E-2	D•E	=	5.159E-4
CONST1	=	2.158E-4	D•X	=	2.13E-4
CONST2	=	6.461E-4	E•W	=	2.13E-4
B1	=	3.547E-2	B•D	=	6.455E-5
(B3)	=	1.196E-3	E•C	=	6.455E-5

Grand Gulf:

DIESEL1	=	2.940E-2	LB2	=	1.447E-2
DIESEL2	=	2.940E-2	LC	=	2.197E-2
DIESEL3	=	2.940E-2	PA27	=	8.100E-4
H	=	2.167E-2	PB27	=	8.100E-4
R	=	5.080E-2	SSA	=	2.847E-2
L	=	2.177E-2	SSB	=	2.847E-2
LA2	=	1.447E-2	SSC	=	1.447E-2

6. Affected Accident Sequences and Base-Case Frequencies:

The PC-based Oconee and Grand Gulf PRA computer codes were used for this analysis. Key interim results are as follows:

Oconee: (RY⁻¹)

T ₂ MLU	=	1.27E-6	T ₂ MLUO	=	1.08E-5
T ₁ MLU	=	2.08E-7	S ₂ D	=	7.45E-7
T ₁ (B3)MLU	=	5.16E-6	S ₃ D	=	1.20E-6
T ₂ MQH	=	1.50E-6	T ₁ MLUO	=	7.21E-6
S ₃ H	=	1.36E-6	T ₃ MLUO	=	1.47E-6
S ₁ D	=	4.81E-6	T ₂ MQD	=	1.29E-6

Grand Gulf: (RY⁻¹)

T ₁ PQI	=	1.37E-6	SI	=	2.63E-6
T ₂₃ PQI	=	2.11E-6	T ₁ QW	=	5.79E-6
T ₁ PQE	=	2.29E-7	T ₂₃ QW	=	8.26E-6
T ₂₃ PQE	=	9.06E-7	T ₁ QUV	=	1.78E-6

7. Affected Release Categories and Base-Case Frequencies:

Oconee: (RY⁻¹)

PWR-1	=	6.12E-8
PWR-3	=	1.87E-5
PWR-5	=	2.96E-7
PWR-7	=	2.25E-5

Grand Gulf: (RY⁻¹)

BWR-1	=	6.79E-8
BWR-2	=	2.05E-5
BWR-3	=	1.53E-6
BWR-4	=	1.75E-6

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

$$\bar{F}_{PWR} = 4.16E-5/RY$$

$$\bar{F}_{BWR} = 2.38E-5/RY$$

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

$$W_{PWR} = 101.68 \text{ person-rem/RY}$$

$$W_{BWR} = 152.52 \text{ person-rem/RY}$$

10. Adjusted-Case Values for Affected Parameters:

Ocone:

D	=	2.355E-2	F1	=	1.780E-3
E	=	2.355E-2	D•E	=	5.149E-4
CONST1	=	2.155E-4	D•X	=	2.129E-4
CONST2	=	6.447E-4	E•W	=	2.129E-4
B1	=	3.545E-2	B•D	=	6.449E-5
(B3)	=	1.145E-3	E•C	=	6.449E-5

Grand Gulf:

DIESEL1	=	2.850E-2	LB2	=	1.445E-2
DIESEL2	=	2.850E-2	LC	=	2.195E-2
DIESEL3	=	2.850E-2	PA27	=	7.700E-4
H	=	2.165E-2	PB27	=	7.700E-4
R	=	5.075E-2	SSA	=	2.845E-2
L	=	2.175E-2	SSB	=	2.845E-2
LA2	=	1.445E-2	SSC	=	1.445E-2

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Ocone: (RY⁻¹)

T ₂ MLU	=	1.26E-6	T ₂ MLUO	=	1.03E-5
T ₁ MLU	=	2.07E-7	S ₂ D	=	7.44E-7
T ₁ (B3)MLU	=	4.94E-6	S ₃ D	=	1.20E-6
T ₂ MQH	=	1.50E-6	T ₁ MLUO	=	6.87E-6
S ₃ H	=	1.36E-6	T ₃ MLUO	=	1.40E-6
S ₁ D	=	4.81E-6	T ₂ MQD	=	1.29E-6

Grand Gulf: (RY⁻¹)

T ₁ PQI	=	1.35E-6	SI	=	2.62E-6
T ₂₃ PQI	=	2.11E-6	T ₁ QW	=	5.65E-6
T ₁ PQE	=	2.22E-7	T ₂₃ QW	=	8.25E-6
T ₂₃ PQE	=	9.03E-7	T ₁ QUV	=	1.73E-6

12. Affected Release Categories and Adjusted-Case Frequencies:

Ocone: (RY⁻¹)

PWR-1	=	6.11E-8
PWR-3	=	1.81E-5
PWR-5	=	2.87E-7
PWR-7	=	2.19E-5

Grand Gulf: (RY⁻¹)

BWR-1	=	6.75E-8
BWR-2	=	2.03E-5
BWR-3	=	1.50E-6
BWR-4	=	1.71E-6

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

$$\bar{F}^*_{PWR} = 4.03E-5/R_Y$$

$$\bar{F}^*_{BWR} = 2.35E-5/R_Y$$

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

$$W^*_{PWR} = 98.30 \text{ person-rem/RY} \quad W^*_{BWR} = 152.96 \text{ person-rem/RY}$$

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

$$\Delta\bar{F}_{PWR} = 1.27E-6/\text{RY} \quad \Delta\bar{F}_{BWR} = 2.66E-7/\text{RY}$$

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

$$\Delta W_{PWR} = 3.37 \text{ person-rem/RY} \quad \Delta W_{BWR} = 1.56 \text{ person-rem/RY}$$

17. Total Public Risk Reduction, (ΔW)_{Total}:

Best Estimate (man-rem)	Error Bounds (man-rem)	
	Upper	Lower
1.1E+4	1.4E+7	0

ATTACHMENT 1

Risk parameters affected by the SIR consist of those associated with the active ESFs, including both primary side and secondary side systems and containment systems. The PWR and BWR risk parameters are evaluated separately.

PWR Risk Parameters

Using the Oconee plant as the reference PWR (Andrews, et al., 1983 and Kolb, et al., 1983), the active ESF components include diesel generators and pumps on all ESF systems. It is uncertain whether the emergency feedwater (EFW) systems belongs to the ESF. However, in this analysis, EFW is included in the ESF consideration. The affected parameters are D, E, CONST1, CONST2, B1, (B3), and F1.

Among these parameters, several have to be redefined for the base-case failure probability calculations. These are explained in detail for each parameter later.

To estimate the adjusted-case failure probabilities, it was assumed that the risk parameters given in the Oconee PRA (Andrews, et al., 1983) reflect the fact that protective trips of the active ESF components are overridden in an accident when the components are actuated. In addition, it is assumed that the failure probabilities of the active ESFs are reduced by about 10%. This number is based on a simple event tree analysis as follows.

For the case of the trip override for a given active ESF component, the simplified event tree is shown in Figure 1a. Based on ESF trip data from January 1, 1984 to June 30, 1984 (Inside NRC, May 27, 1985), there were a total of 501 ESF trips. Of these, 365 trips (73 percent) were "invalid and unnecessary." The rest were valid trips. This is shown in the Figure. In this analysis, it was assumed that 50% of the ESF components will fail prior to accomplishing their intended function if not overridden and 50% will successfully accomplish their intended function before they fail.

For the case in which the ESF trip is not overridden and repair is allowed, a third level is added to the event tree, as shown in Figure 1b. The third level is related to the possibility that sufficient time may be available to affect repairs before the demand for the ESF equipment is critical. For this analysis, without actual data, it was assumed that 50% of the faults can be repaired and returned for continued operation. The rest (50 percent) will fail. As shown in the Figure, the success rates for the override and no override cases are 0.865 and 0.905, respectively. This represents approximately a 5% increase in the successful actuation of ESF components.

Figure 1. Event Tree for ESF Override/No Override

This increase in reliability would be offset somewhat by an increase in the number of spurious trips of ESF components caused by their protective devices. Therefore, the competing negative effects of SIR would result in an increase in the number of ESF component trips, resulting a decrease in reliability. However, it is believed that the 5% increase in reliability that was calculated to result from allowing component repairs after they are tripped is not an upper bound. The actual increase in reliability may be higher. For this reason, the 5% increase in reliability that is used for calculation purposes is judged to include the competing negative effects (i.e., spurious trips) if SIR.

Parameters D, E

The base-case failure probabilities for both D and E need modification. The pumps in events D and E fall into the standby pump category. To model the base-case, the failure mode "does not continue to run after start" must be included in the pump failure rate. The pump failure probabilities for "does not start" were taken from the Nuclear Computerized Library for Assessing Reactor Risks (NUCLARR; Gertman, et al., 1988). In the NUCLARR data base, several failure rate data points were given for failure of a normally-standby electric pump to continue running. The value used in this analysis was obtained by taking the weighted average over all listed operating hours. The weighting factor was calculated by dividing each pumps listed operating hours by the total operating hours for all pumps in this class. The weighted average was calculated by summing the products of the weighting factors and the listed pump failure rates. The pump failure probability for this analysis was taken as the probability of failure of the pump to operate over a 24-hr period. The total failure probability for all

contributing failure modes for the electric pumps in events D and E are as follows:

<u>Contributor</u>	<u>Failure Probability</u>	
	<u>Modified Base-Case</u>	<u>Adjusted-Case</u>
Hardware	1.0E-3	1.0E-3
Control Circuitry	1.8E-3	1.8E-3
Test Outage	<u>1.9E-3</u>	<u>1.9E-3</u>
Does Not Start (sum of above)	4.7E-3	4.7E-3
Does Not Run After Start	4.73E-4	4.49E-4
<hr/>		
Total	5.17E-3	5.15E-4

The adjusted-case probability for the pump was obtained by reducing the contributing values for "pump dose not continue to run after start" by 10% and keeping the other values unchanged. The results are given in the third column of the above table.

The base-case failure probabilities for events D and E are the sum of the modified base-case failure probabilities for the pump and the original PRA probabilities for other components:

$$2(0.0001) + 2(0.0032) + 0.0096 + 0.0002 + 0.001 + 0.00517 + 0.001 = 0.02357$$

The adjusted-case failure probabilities for events D and E were calculated by substituting the adjusted-case failure probability for the pump into the above equation:

$$2(0.0001) + 2(0.0032) + 0.0096 + 0.0002 + 0.001 + 0.00515 + 0.001 = 0.02355$$

Excess significant figures were maintained in the above value for calculation purposes.

Parameter CONST1

The base-case failure probability for CONST1 has to be redefined. There are two electric pumps and one turbine-driven pump for CONST1. The concern for the pumps is the failure to continue to operate given a successful start. Therefore, the pump failure probability for the base-case should be revised similarly to that done for events D and E. This was done using pump failure data in the NUCLARR data base for both types of pumps, as follows:

<u>Component Description</u>	<u>Fault Identifiers</u>	<u>Failure Contributors</u>	<u>Modified Base-Case Q/component</u>	<u>Adjusted-Case Q/component</u>
Electric pump	EFP-A EFP-B	Hardware	1.0E-3	1.0E-3
		Control	1.8E-3	1.8E-3
		circuitry		
		Maintenance	<u>5.8E-3</u>	<u>5.8E-3</u>
		Does not Start	8.6E-3	8.6E-3
		(sum of above)		
		Does not Con- tinue to Run After Start	4.7E-4	4.47E-4
<hr/>				
		Q Total	9.07E-3	9.047E-3
Turbine Pump	EFP-TD	Hardware	9.1E-2	9.1E-2
		Control	<u>5.8E-3</u>	<u>5.8E-3</u>
		circuitry		
		Does not Start	9.7E-2	9.7E-2
		(sum of above)		
		Does not Con- tinue to Run After Start	1.1E-4	1.045E-4
		<hr/>		
		Q Total	9.711E-2	9.710E-2

To calculate the adjusted-case value for CONST1, the failure probabilities for EFP-A, EFP-B, and EFP-TD need to be modified. The contributing probabilities for "does not run after start" were reduced by 10%. The results were shown in the last column of the above table. The value of CONST1 for the modified base-case and adjusted-case are 2.158E-4 and 2.155E-4, respectively.

Parameter CONST2

With the same adjustments for EFP-A, EFP-B, and EFP-TD as were done for CONST1, the probabilities of the modified base-case and adjusted-case CONST2 are 6.461E-4 and 6.447E-4, respectively.

Parameter B1

The base-case failure probability for B1 needs modification to incorporate failure of HPIS pump C to continue to run after start. As was done in events D and E, the mode of "failure to run after start" is added to other failure modes given in NUREG/CR-2800 to calculate the modified base-case failure probability. The value used for this failure mode was taken as the weighted-average of the data points given in the NUCLARR data base. The modified base-case and adjusted-case failure probabilities for the HPIS pump is as follows:

Contributor	Failure Probability	
	Modified Base-Case	Adjusted-Case
Hardware	1.0E-3	1.0E-3
Control circuitry	1.1E-3	1.1E-3
Lube oil becoming viscous	1.0E-2	1.0E-2
Service water not valved in	1.0E-3	1.0E-3
Maintenance outage	2.1E-3	2.1E-3
Test outage	<u>1.9E-3</u>	<u>1.9E-3</u>
Does not start (sum of above)	1.71E-2	1.71E-2
Does not continue to run after start	4.73E-4	4.49E-4
<hr/>		
Total	1.757E-2	1.755E-2

The modified base-case value for B1 is:

$$2(0.0096) + 3(0.0002) + 2(0.0001) + 0.01757 - 0.0021 = 0.03547$$

The adjusted-case value is:

$$2(0.0096) + 3(0.0002) + 2(0.0001) + 0.01755 - 0.0021 = 0.03545$$

Parameter (B3)

In the Oconee plant, the emergency power is provided by two hydroelectric generators. In this analysis, the failure probabilities for diesel generators were updated using data in NSAC-108 (Wyckoff 1986). These values were also used for the failure probability for the hydroelectric generator.

The failure modes for emergency diesel generators consist of "does not start" and "does not continue to run." In this safety issue, the failure mode "does not continue to run" is of concern. From NSAC-108, the unreliability of diesel generators was estimated to be 0.005/demand for the mode "does not start" and 0.018/demand for the "load-run faults." The total unreliability was given as 0.022/demand. These values will be used for the modified base-case failure probabilities for emergency diesel generators. For the adjusted-case, the "load-run" failure probability was reduced by 10%. Therefore:

	<u>Modified Base-Case</u>	<u>Adjusted-Case</u>
Failure to Start	0.005	0.005
Load-Run Faults	<u>0.018</u>	<u>0.0171</u>
Total:	<u>0.023</u>	<u>0.0221</u>

The maintenance outage is assumed to be unchanged at 5.8E-3 as in the Oconee PRA. Therefore, the "equivalent" value for (B3) is the sum of (1) both generators fail to continue to run, (2) either generator fails to continue to run while the other is down for maintenance, and (3) both emergency DC batteries needed for generator

startup fail (the value of $4E-4$ from the Oconee PRA is used for the batteries). Therefore, the modified base-case value of event (B3) is:

$$(0.023)^2 + 2(0.023)(0.0058) + 4E-4 = 1.196E-3$$

The adjusted case value of (B3) is:

$$(0.0221)^2 + 2(0.0221)(0.0058) + 4E-4 = 1.145E-3$$

Parameter F1

The base-case failure probability for F1 is $1.87E-3$ (see Electric pump in parameters D and E). The adjusted-case failure probability is $(0.95 \times 0.00187) = 1.78E-3$.

Parameter D•E

The modified base-case probability for D•E is:

$$(0.02357)^2 - (0.0063)^2 = 5.159E-4$$

where the value of 0.0063 is for the total maintenance contribution. The adjusted-case value is:

$$(0.02355)^2 - (0.0063)^2 = 5.149E-4$$

Parameters D•X and E•W

The modified base-case value is:

$$(0.02357)(0.0096) - (0.0063)(0.0021) = 2.130E-4$$

The adjusted-case value is:

$$(0.02355)(0.0096) - (0.0063)(0.0021) = 2.129E-4$$

Parameters B•D and E•C

The modified base-case value is:

$$(0.0033)(0.02357) - (0.0063)(0.0021) = 6.455E-5$$

The adjusted-case value is:

$$(0.0033)(0.02355) - (0.0063)(0.0021) = 6.449E-5$$

BWR RISK PARAMETERS

For BWRs, Grand Gulf is used as the reference plant (Andrews, et al., 1983 and Hatch 1981). The affected parameters in terms of the active ESF components are DIESEL1, DIESEL2, DIESEL3, H, R, L, LA2, LB2, LC, PA27, PB27, SSA, SSB, and SSC.

Parameters DIESEL1, DIESEL2, and DIESEL3

The base-case failure probabilities for these three events need modification similarly to event (B3) for PWRs. Maintenance outage is unchanged from the Grand Gulf PRA value of $6.4E-3$. The failure probability is the sum of the failure modes "does not start," "load-run faults," and "maintenance outage." The adjusted-case probability for the mode "load-run faults" was calculated by reducing the base-case probability by 5%. The modified base-case and adjusted-case values for DIESEL1, DIESEL2, and DIESEL3 are shown below:

	<u>Modified Base-Case</u>	<u>Adjusted-Case</u>
Failure to Start	0.005	0.005
Load-run Faults	0.018	0.01671
Maintenance Outage	<u>0.0064</u>	<u>0.0064</u>
Total	0.0294	0.0285

Parameter H

The base-case failure probability for event H needs to be modified. The contribution from the pump component needs to be redefined to include the failure mode "does not continue to run" as was done for PWR events D and E. This is done as follows:

<u>Failure Contributors</u>	<u>Modified Base-Case Q/component</u>	<u>Adjusted-Case Q/component</u>
Hardware	$1.0E-3$	$1.0E-3$
Control	$1.0E-3$	$1.0E-3$
circuitry		
Maintenance	<u>$5.8E-3$</u>	<u>$5.8E-3$</u>
Does not Start	$7.8E-3$	$7.8E-3$
(sum of above)		
Does not Continue to Run After Start	$4.7E-4$	$4.47E-4$
<hr/>		
Q Total	$8.27E-3$	$8.25E-3$

The adjusted-case probability was obtained by reducing the contributing value for "does not continue to run" by 5%. The result was given in the third column of the above table.

The modified base-case failure probability for event H is:

$$3(0.0001) + 0.0059 + 0.0072 + 0.00827 = 2.167E-2$$

The adjusted-case failure probability for event H is:

$$3(0.0001) + 0.0059 + 0.0072 + 0.00825 = 2.165E-2$$

Parameter R

The base-case failure probability for event R needs to be modified to include the failure mode "does not continue to run." From NUCLARR, the weighted-average failure probability for normally standby turbine-driven pumps for this failure mode was $1.1\text{E-}4$ (see event CONST1 for PWRs). The redefined base case values for a normally standby electric pump was $8.27\text{E-}3$, as shown in event H. The adjusted case values are obtained by multiplying these values by 0.95. Therefore, the adjusted-case failure probabilities are $1.045\text{E-}4$ and $8.25\text{E-}3$ for the turbine-driven and electric pumps, respectively.

The modified base-case value for event R is:

$$5(0.0001) + 2(0.0002) + 4(0.0059) + 2(0.0072) + 0.0013 + 0.0022 + 0.00011 + 0.00827 = 5.078\text{E-}2$$

The adjusted-case value for event R is:

$$5(0.0001) + 2(0.0002) + 4(0.0059) + 2(0.0072) + 0.0013 + 0.0022 + 0.000105 + 0.00825 = 5.075\text{E-}2$$

Parameter L

The base-case failure probability for event L needs to be modified because of the pump (see event H). The modified value for the pump is $8.27\text{E-}3$ and the adjusted-case value is 95% of this value or $8.25\text{E-}3$. Therefore, the modified base-case value for event L is:

$$2(0.0001) + 0.0059 + 0.0072 + 0.0002 + 0.00827 = 2.177\text{E-}2$$

The adjusted-case value for event L is:

$$2(0.0001) + 0.0059 + 0.0072 + 0.0002 + 0.00825 = 2.175\text{E-}2$$

Parameters LA2, LB2

The base-case failure probability for events LA2 and LB2 also needs to be modified because of the pump. The modified base-case value for the pump is $8.27\text{E-}3$ and the adjusted-case value is $8.25\text{E-}3$ (see event H). Therefore, the modified base-case value for events LA2 and LB2 are:

$$0.0001 + 0.0059 + 0.0002 + 0.00827 = 1.447\text{E-}2$$

The adjusted-case value for event L is:

$$0.0001 + 0.0059 + 0.0002 + 0.00825 = 1.445\text{E-}2$$

Parameter LC

The base-case failure probability for event LC needs to be modified because of the pump. The modified base-case value for the pump is $8.27\text{E-}3$ and the adjusted-case value is 95% of this value or $8.25\text{E-}3$. Therefore, the modified base-case value for event LC is:

$$2(0.0001) + 0.0059 + 0.0072 + 2(0.0002) + 0.00827 = 2.197\text{E-}2$$

The adjusted-case value for event L is:

$$2(0.0001) + 0.0059 + 0.0072 + 2(0.0002) + 0.00825 = 2.195\text{E-}2$$

Parameters PA27 and PB27

The base-case value is unchanged from the Grand Gulf PRA value of $8.1\text{E-}4$. The adjusted case value is $0.95 \times 8.1\text{E-}4 = 7.7\text{E-}4$.

Parameters SSA and SSB

The base-case failure probabilities for events SSA and SSB need to be modified because of the pump (see event H). The modified value for the pump is $8.27\text{E-}3$ and the adjusted-case value is 95% of this value or $8.23\text{E-}3$. Therefore, the modified base-case value for events SSA and SSB are:

$$0.0001 + 2(0.0059) + 0.0083 + 0.00827 = 2.847\text{E-}2$$

The adjusted-case value for event L is:

$$0.0001 + 2(0.0059) + 0.0083 + 0.00825 = 2.845\text{E-}2$$

Parameter SSC

Similar to events SSA and SSB, the modified base-case failure probability for event SSC is:

$$0.0001 + 0.0059 + 0.0002 + 0.00827 = 1.447\text{E-}2$$

The adjusted-case value for event SSC is:

$$0.0001 + 0.0059 + 0.0002 + 0.00825 = 1.445\text{E-}2$$

PARAMETERS T₂₃PQI and SI

VGB2•SSA and VGA2•SSB

The modified base-case value is:

$$(0.024)(0.02847) - 2(3)(0.0058)^2 = 4.814\text{E-}4$$

The adjusted-case value is:

$$(0.024)(0.02845) - 2(3)(0.0058)^2 = 4.810\text{E-}4$$

VGA2•LB2 and VGB2•LA2

The modified base-case value is:

$$(0.024)(0.01447) - [2(0.0058)]^2 = 2.127\text{E-}4$$

The adjusted-case value is:

$$(0.024)(0.01445) - [2(0.0058)]^2 = 2.122\text{E-}4$$

SSA•SSB

The modified base-case value is:

$$(0.02847)^2 - [3(0.0058)]^2 = 5.078\text{E-}4$$

The adjusted-case value is:

$$(0.02845)^2 - [3(0.0058)]^2 = 5.066\text{E-}4$$

VGB1•SSA and VGA1•SSB

The modified base-case value is:

$$(0.01447)(0.02847) - 2(3)(0.0058)^2 = 2.252\text{E-}4$$

The adjusted-case value is:

$$(0.01445)(0.02845) - 2(3)(0.0058)^2 = 2.249\text{E-}4$$

LB2•SSA and LA2•SSB

The modified base-case value is:

$$(0.01447)(0.02847) - 2(3)(0.0058)^2 = 2.101\text{E-}4$$

The adjusted-case value is:

$$(0.01445)(0.02845) - 2(3)(0.0058)^2 = 2.093\text{E-}4$$

VGA1•LB2 and VGB1•LA2

The modified base-case value is:

$$(0.015)(0.01447) - [2(0.0058)]^2 = 8.249\text{E-}5$$

The adjusted-case value is:

$$(0.015)(0.01445) - [2(0.0058)]^2 = 8.219\text{E-}5$$

LA2•LB2

The modified base-case value is:

$$(0.01447)^2 - [2(0.0058)]^2 = 7.482\text{E-}5$$

The adjusted-case value is:

$$(0.01445)^2 - [2(0.0058)]^2 = 7.424\text{E-}5$$

PARAMETERS T_1PQE and $T_{23}PQE$

H•R

The modified base-case value is:

$$(0.02167)(0.05078) - 3(7)(0.0058)^2 = 3.940E-4$$

The adjusted-case value is:

$$(0.02165)(0.05075) - 3(7)(0.0058)^2 = 3.923E-4$$

SSA•SSB

The modified base-case value is:

$$(0.02847)^2 - [2(0.0058)]^2 = 6.760E-4$$

The adjusted-case value is:

$$(0.02845)^2 - [2(0.0058)]^2 = 6.748E-4$$

VGB1•SSA and VGB1•SSB

The modified base-case value is:

$$(0.015)(0.02847) - [2(0.0058)]^2 = 2.925E-4$$

The adjusted-case value is:

$$(0.015)(0.02845) - [2(0.0058)]^2 = 2.922E-4$$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Equipment-Protective Devices on Engineered Safety Features (110)

2. Affected Plants (N):

All PWRs and BWRs are assumed to be affected.

	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	47	43	90
BWRs	24	20	44

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

	<u>$\bar{T}$</u>		
	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	27.7	30	28.8
BWRs	25.2	30	27.4

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

PWR = (19,860 person-rem)(1.27E-6/RY) = 2.52E-2 person-rem/RY

BWR = (19,860 person-rem)(2.70E-7/RY) = 5.36E-3 person-rem/RY

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

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
7.2E+1	1.6E+4	0

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

The SIR implementation involves procedure changes only. It does not involve utility labor in radiation zones. Therefore, the occupational dose increase for SIR implementation is zero.

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

From NUREG/CR-2098 (Atwood 1983), the total number of ESF pump failures in the category "do not continue to run after start" is 478, based on failures reported in LERs from 1972 to 1980. Assuming that 37 PWRs and 16 BWRs were operating during that time period, the overall pump failure frequency is as follows:

$$\frac{478 \text{ events}}{(37+16) \text{ plants} \times 9 \text{ years}} = 1 \text{ event/RY}$$

As discussed in Attachment 1, 50% of the failures in "do not continue to run" category were assumed to have no safety significance (i.e., 0.5

event/Ry). It was assumed that 1 man-hr labor is required for inspection of each trip signal to determine if the components will continue to operate safely. Therefore, the labor requirement in radiation zones (0.05 R/hr) for pumps is (0.5 event/Ry) (1 man-hr/event) = 0.5 man-hr/Ry.

From NUREG/CR-2270 (Steverson and Atwood 1983), the total number of MOV failures reported in LERs from 1976 to 1980 was 422. It was assumed that all of these are ESF components (i.e., including MOVs in emergency feedwater systems in the ESF estimation). Assuming 15 PWRs and 4 BWRs were in operation during this time period, the resulting MOV failure frequency is estimated to be:

$$\frac{422 \text{ events}}{(15+4) \text{ plants} \times 5 \text{ years}} = 4.4 \text{ events/Ry}$$

It was assumed that half of these events are caused by NC MOVs (i.e., 2.2 events/Ry). As discussed in Attachment 1, 50% of the NC MOV failure events are of safety significance (i.e., 1.1 events/Ry). Assuming 1 man-hr per event to inspect the component, the total labor requirement for MOVs is estimated to be (1.1 event/Ry) (1 man-hr/event) = 1.1 man-hr/Ry.

From NSAC-108, a total of 6 emergency diesel generator failures were reported to result from real (unplanned) demands during the years 1983 to 1985. A total of 67 plants were listed in the NSAC report as providing data to the study. Therefore, the resulting diesel generator failure frequency is estimated to be:

$$\frac{6 \text{ events}}{67 \text{ plants} \times 3 \text{ years}} = 0.03 \text{ event/Ry}$$

Assuming 1 man-hr per event in radiation zones for inspection, the total labor requirements for diesel generators is (0.03 events/Ry) (1 man-hr/event) = 0.03 man-hr/Ry.

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

Pump trips	= (0.05 R/hr)(0.5 man-hr/Ry)	= 0.025 man-rem/Ry
NC MOV trips	= (0.05 R/hr)(1.1 man-hr/Ry)	= 0.055 man-rem/Ry
Diesel generator trips	= (0.05 R/hr)(0.03 man-hr/Ry)	= 0.0015 man-rem/Ry
Total		= 0.082 man-rem/Ry

11. Total Occupational Dose Increase for SIR Operation/Maintenance, (NTD_o):

Operation/maintenance doses are incurred at all plants.

$$\begin{aligned} \text{NTD}_o &= (90 \text{ PWRs})(28.8 \text{ yr})(0.082 \text{ man-rem/Ry}) \\ &\quad + (44 \text{ BWRs})(27.4 \text{ yr})(0.082 \text{ man-rem/Ry}) \\ &= 3.1\text{E}+2 \text{ man-rem} \end{aligned}$$

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
3.1E+2	9.3E+2	1.0E+2

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Equipment-Protective Devices on Engineered Safety Features (110)

2. Affected Plants (N):

All PWRs and BWRs are assumed to be affected.

	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	47	43	90
BWRs	24	20	44

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

	<u>$\bar{T}$</u>		
	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	27.7	30	28.8
BWRs	25.2	30	27.4

Industry Costs (Steps 4 through 12):

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

$$\Delta\bar{F}A \text{ (PWR)} = (\$1.65E+9) (1.27E-6/RY) = \$ 2096/RY$$

$$\Delta\bar{F}A \text{ (BWR)} = (\$1.65E+9) (2.70E-7/RY) = \$ 445/RY$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$6.0E+6	\$1.4E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

The SIR implementation involves significant procedure changes. A labor requirement of 10 man-wks per affected ESF component is assumed. These resources are needed for design evaluations and safety analyses to determine the types of faults that can be recovered successfully and under what plant conditions the repairs may be affected. Resources are also needed to train operators and repair crews and for quality assurance. For PWRs, a total of 15 ESF components are affected (3 diesel generators, 7 pumps, and 5 NC MOVs). For BWRs, a total of 18 components are affected (3 diesels, 10 pumps, and 5 NC MOVs). Therefore, the total per-plant industry

labor requirements for SIR implementation are 150 man-wks for PWRs and 180 man-wks for BWRs.

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

$$I \text{ (PWR)} = (150 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$3.4\text{E}+5/\text{plant}$$

$$I \text{ (BWR)} = (180 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$4.0\text{E}+5/\text{plant}$$

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

$$\begin{aligned} NI &= (90 \text{ PWRs})(\$3.4\text{E}+5/\text{plant}) + (44 \text{ BWRs})(\$4.0\text{E}+5/\text{plant}) \\ &= \$4.8\text{E}+7 \end{aligned}$$

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

Labor requirement for ESF pumps	=	0.5 man-hr/Ry
Labor requirement for NC MOVs	=	1.1 man-hr/Ry
Labor requirement for diesel generators	=	<u>0.03 man-hr/Ry</u>
Total:		<u>1.6 man-hr/Ry</u>

See Table 2 for discussion.

In addition, due to the complexity of the procedure changes required by this SIR, it is estimated that 1 man-wk/Ry is needed to review and validate procedures for all affected SIRs and an additional 1 man-wk/Ry is needed to carefully document ESF component operability and failure trends and review manufacturer's data on the ESF components.

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

$$\begin{aligned} I_o &= (\$2270/\text{man-wk})/[1.6 \text{ man-hr/Ry}](40 \text{ man-hr/man-wk}) \\ &\quad + (\$2270/\text{man-wk})(2 \text{ man-wk/Ry}) \\ &= \$4630/\text{Ry} \end{aligned}$$

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

$$\begin{aligned} NTI_o &= (134 \text{ plants})(28.3 \text{ yr})(\$4630/\text{Ry}) \\ &= \$1.8\text{E}+7 \end{aligned}$$

12. Total Industry Cost (S_i):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$6.6E+7	\$9.1E+7	\$4.0E+7

NRC Cost (Steps 13 through 21)

13-20. Steps Related to NRC Costs for SIR Development, Support of SIR Implementation, and Review of SIR Operation and Maintenance

13. NRC Resources for SIR Development:

The NRC will review the generic issue, the operational procedure changes submitted by the licensees, and make recommendations regarding the changes. A resource requirement of 0.2 man-yr is allocated for these activities. The NRC is also assumed to initiate a research project to evaluate pump, valve, and diesel generator manufacturer's specifications, design requirements, and repair times and then compare these data with the core damage initiation times in published PRAs and SARs. Based on these data, conclusions could be developed regarding the survivability and ability to repair ESF components within the allowable times. This study is estimated to require approximately 2 man-yr of contractor labor and an additional 0.2 man-yr of NRC staff labor.

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

$$\begin{aligned} C_D &= (0.2 \text{ man-yr})(\$1.0E+5/\text{man-yr}) + (2 \text{ man-yr})(\$1.0E+5/\text{man-yr}) \\ &+ (0.2 \text{ man-yr})(\$1.0E+5/\text{man-yr}) \\ &= \$2.4E+5 \end{aligned}$$

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

It is estimated that approximately 4 man-wk/plant of NRC labor are required to review and evaluate the licensee's design, safety analyses, and QA documentation for the procedure changes.

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

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

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

$$\begin{aligned} NC &= (134 \text{ plants})(\$9.08E+3/\text{plant}) \\ &= \$1.22E+6 \end{aligned}$$

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

NRC labor for this activity is anticipated to be minimal. It is estimated that 1.0 man-wk/Ry will be sufficient for NRC review of each procedure change and the ESF component data developed by the licensees.

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

$$C_o = (1.0 \text{ man-wk/Ry})(\$2270/\text{man-wk}) = \$2.27E+3/\text{Ry}$$

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

$$\begin{aligned} \bar{N}TC_o &= (134 \text{ plants})(28.3 \text{ yr})(\$2.27E+3/\text{Ry}) \\ &= \$8.84E+6 \end{aligned}$$

21. Total NRC Cost (S_N):

Best Estimate

\$1.0E+7

Upper Bound

\$1.4E+7

Lower Bound

\$5.7E+6

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 115, Enhancement of the Reliability of Westinghouse Solid State Protection System (SSPS).

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: For many years, a great deal of discussion and study has been expended on the Anticipated Transient Without Scram (ATWS) issue and on the benefits of various ATWS fixes. As a consequence, the ATWS rule for Westinghouse plants requires the implementation of a diverse ATWS mitigation system, Auxiliary Mitigating Systems Actuation Circuitry (AMSAC). The function prescribed for AMSAC are turbine trip and the initiation of auxiliary feedwater, independent of the reactor trip system. With respect to Westinghouse plants with the solid state protection system design, recent failures of the undervoltage driver have raised concerns with regards to the susceptibility of the design to common mode failures and/or random failures of redundant components. It is proposed that incorporation of additional diversity for the undervoltage driver function could significantly reduce the potential for an ATWS problem.

AFFECTED PLANTS: BWR: Operating = 0 Planned = 0
PWR: Operating = 19 Planned = 0

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 2.5E+05

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	9.7E+02

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	0.38
SIR Operation/Maintenance =	0
Total of Above =	0.38
Accident Avoidance =	96.0

NRC COSTS:

SIR Development =	0.038
SIR Implementation Support =	0.19
SIR Operation/Maintenance Review =	0
Total of Above =	0.23

ENHANCEMENT OF THE RELIABILITY OF WESTINGHOUSE
SOLID STATE PROTECTION SYSTEM (SSPS)

ISSUE 115

1.0 SAFETY ISSUE DESCRIPTION

For many years, a great deal of discussion and study has been expended on the ATWS issue and on the benefits of various ATWS fixes. As a consequence, the ATWS rule for Westinghouse plants requires the implementation of a diverse ATWS mitigation system, or auxiliary mitigating systems actuation circuitry (AMSAC). The functions prescribed for AMSAC are turbine trip and the initiation of auxiliary feedwater, independent of the reactor trip system. For other than Westinghouse plants, the ATWS rule includes the requirement for a diverse scram system. The diverse scram system was excluded from the ATWS rule for Westinghouse plants due to a marginally favorable value/impact.

With respect to Westinghouse plants with the solid state protection system design, recent failures of the undervoltage driver have raised concerns with regards to the susceptibility of the design to common mode failures and/or random failures of redundant components.

It is proposed that relatively inexpensive modifications to the Westinghouse solid state protection system design could significantly reduce the potential for an ATWS event by incorporating additional diversity for the undervoltage (UV) driver function. This is to avoid the potential of ATWS events due to common mode failure or random failures of the protection system.

AFFECTED PLANTS

This issue deals only with certain Westinghouse design plants. These are the Westinghouse plants with Solid State Protection System (SSPS). These plants represent 19 of the 38 operating Westinghouse plants.

2.0 SAFETY ISSUE RISK AND DOSE

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

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Enhancement of the Reliability of Westinghouse Solid State Protection System (SSPS) (115)

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

	$\frac{N}{19}$	$\frac{\bar{T} \text{ (yr)}}{25.5}$
PWR		

3. Plants Selected for Analysis:

No "specific" Westinghouse (W) design PWR has been selected as the representative plant. Instead, generic event trees (related to W PWRs) modeling plant response to an ATWS event have been used to quantify the ATWS induced core-melt frequencies.

4. Parameters Affected by SIR:

The parameter affected in this analysis is the reliability of the electrical (solid state) portion of the reactor protection system (RPS). This is discussed further in Attachment 1.

5. Base-Case Values for Affected Parameters:

The base-case failure probability of the RPS (electrical portion) is given to be $6.8E-05$.

6. Affected Accident Sequences and Base-Case Frequencies:

Not applicable. Refer to the event trees presented in Attachment 1.

7. Affected Release Categories and Base-Case Frequencies:

In order to be conservative in this analysis, an average of release via PWR release categories 1, 2, and 3 has been selected. This is estimated to be $5.2E+06$ man-rem/event.

8. Base-Case Affected Core-Melt Frequency (F):

The sum of the core-melt frequencies for the current system (base-case) is estimated to be $(3.5E-05/\text{py} + 8.3E-05/\text{py} + 4.6E-06/\text{py} + 1.02E-05/\text{py}) = 1.33E-04/\text{py}$. This is the sum of the core-melt frequencies derived in event trees presented in Attachment 1.

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

The base-case public risk is estimated to be $(1.33\text{E-}04/\text{py})(5.2\text{E}+06 \text{ man-rem})$ or 691 man-rem/py.

10. Adjusted-Case, Values for Affected Parameters:

It is estimated that the failure probability of the RPS (electrical portion) will decrease to $1.46\text{E-}05$ following the modification. This value is derived in Attachment 1.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Not applicable. Refer to event trees presented in Attachment 1.

12. Affected Release Categories and Adjusted-Case Frequencies:

Same as step 7.

13. Adjusted-Case, Affected Core-Melt Frequency (F*):

The sum of the core-melt frequencies following system modification is estimated to be $(7.9\text{E-}06/\text{py} + 1.9\text{E-}05/\text{py} + 1.4\text{E-}06/\text{py} + 3.3\text{E-}06/\text{py}) = 3.16\text{E-}05/\text{PY}$.

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

The adjusted-case public risk is estimated to be $(3.16\text{E-}05/\text{py})(5.2\text{E}+06 \text{ man-rem})$ or 164 man-rem.

15. Reduction in Core-Melt Frequency (ΔF):

Reduction in core-melt frequency is estimated to be $(1.33\text{E-}04/\text{py} - 3.16\text{E-}05/\text{py})$ or $1.01\text{E-}04/\text{py}$.

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

Per-plant public risk reduction is estimated to be $(1.01\text{E-}04/\text{py})(5.2\text{E}+06 \text{ man-rem})$ or 525 man-rem/py.

17. Total Public Risk Reduction. (ΔW)_{Total}

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
2.5E+05	9.3E+06	0

ATTACHMENT 1

The analysis was based on Reliability block diagrams for the Westinghouse Solid State Protection System (SSPS). These diagrams were provided to the NRC staff as a part of the Westinghouse Owners Group response to NRC questions during the review of WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation Systems," and cannot be provided here because they are proprietary. Diversity exists in this design in two basic forms. The first is from the standpoint of measured parameters and sensors that initiate a reactor trip and the second is the diverse trip features of the reactor trip breakers (shunt and undervoltage trip coils). For the analog channels, comparators are the major component that are common to each channel. For the logic cabinet, input relays and the universal (logic) cards are common for each trip function, with the undervoltage driver common to all trip functions. For the reactor trip breakers, the remaining components (primarily mechanical) are common to all trip functions.

Estimates of the system failures due to common mode failures can be made based on the failure rates of components, the time interval between surveillance tests and an assumed Beta factor. Beta factor is the measure of contribution of common-cause failures to overall system failure. Table A.1 summarizes the estimates for common mode failures of the protection system on the bases of a Beta factor of 0.01 and a monthly test interval. A Beta factor of 0.01 is used to be consistent with that used for logic channels as noted in Attachment A (Scram Reliability) of Enclosure D (Recommendations of the ATWS Task Force) to SECY-83-293 (Amendment to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events). Currently, Technical Specifications require testing of breakers and logic every 62 days on a staggered test basis (one train or the other is tested every 31 days such that the time interval for finding common mode failures would be monthly). Based on the review of WCAP-10271, the NRC has approved quarterly testing of analog channels. Since the majority of the trip functions consist of 3 or 4 channels, quarterly tests on a staggered test basis for a 3-channel system results in one channel being tested monthly. Thus, a monthly test interval is also used for analog channels.

The channel comparators are the major contributor to the common mode failure unavailability since they have the largest hourly failure rate. However, if the hourly failure rate for the undervoltage drive is estimated based on five known failures to date and an estimate of 90 reactor-years for Westinghouse plants which have the solid state protection system, the common mode failure unavailability of the undervoltage driver (see Table A.2) becomes the dominant contributor.

This analysis would suggest that, if the undervoltage driver function included diversity, the common mode failure vulnerability could be reduced by a factor of about three. Further, if it is assumed that the hourly failure rate of comparators encompasses both failures to trip as well as failures in the tripped state (safe failures), one might conclude that the estimate of common mode failure of comparators is conservative by as much as a factor of two. The same would not be true for the experience-based failure rate of the undervoltage driver, since the five known failures were all in the non-safe state. This assumption would make the addition of diversity for the undervoltage driver

function a potential benefit of a factor of six in the reduction in ATWS probability due to protection system failures.

TABLE A.1

<u>Components</u>	<u>Common Mode ^(a) Failure Unavailability (E-5)</u>
Channel Comparators ($\lambda = 2.9\text{E-}6/\text{hr}$)	1.1
Logic Input Relays ($\lambda = 8.7\text{E-}8/\text{hr}$)	0.032
Universal Logic Cards ($\lambda = 7.7\text{E-}7/\text{hr}$)	0.29
Undervoltage Driver ($\lambda = 1.95\text{E-}7/\text{hr}$)	0.073
Breaker Mechanical Components ($\lambda = 8.33\text{E-}8/\text{hr}$)	<u>0.031</u>
Total:	<u>1.53</u>

(a) $U = (b\lambda T)/2$ (Average Unavailability due to common mode failure)

TABLE A.2

Undervoltage Driver Failures	5
Reactor Year (Estimated) SSPS Plants	90
Failure Rate	0.056/yr ($6.3\text{E-}6/\text{hr}$)
Common Mode Failure Probability ^(a)	2.3E-5
All other components ($1.53 - 0.073$)E-5	<u>1.46E-5</u>
Total:	5.76E-5

(a) $U = (b\lambda T)/2$ (Average Unavailability due to common mode failure)

The SSPS, in addition to initiating reactor trip, is used to initiate engineered safeguard systems. While these functions of the protection system utilize many of the same components as the reactor trip system (comparators, logic input relays, and universal logic cards), it differs from the reactor trip system in its final output configuration. Instead of an undervoltage drive which turns off 48V DC to the actuated components, a relay driver is provided which supplies 48V DC to energize a master relay, which in turn energizes slave relays that provide contacts to actuate engineered safeguard components. Thus, a relay driver and associated relays could be used to duplicate the function of the undervoltage drive and thereby provide diversity. This would eliminate common mode failures of the undervoltage driver as the dominant contributor to probability of an ATWS event due to protection system failures.

Diversity for the undervoltage driver function would also significantly reduce the probability of an ATWS due to random failures of the undervoltage drive as well as due to testing.

An estimate of the unavailability of the reactor trip system due to both common mode and random failures of the undervoltage driver with and without the proposed modifications are provided in subsection A.1. Subsection A.2 presents

the estimates in the change in the probability of adverse consequences (core-damage) due to an ATWS based on the data provided in subsection A.1. This evaluation is made using the developed generic event trees for an ATWS event. The event trees were obtained from AMENDMENT TO 10 CFR 50 RELATED TO ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) EVENTS, SECY-83-293.

A.1 UNAVAILABILITY ANALYSIS

Random failure of redundant components with staggered testing are estimated based on the following equation (McCormick, 1981).

$$P(f) = 1 - e^{-\lambda t}$$

Where $P(f)$ is the failure probability, λ is the failure rate of the component, and t is the test interval (time).

With staggered testing, $P(f)$ for one component is λt and for the other component is $\lambda(t + T/2)$ during one-half of the test interval. Thus the average unavailability (U) of both components being failed at the same time is:

$$U_{avg} = \frac{1}{T} \int_0^{T/2} (\lambda t) \lambda (t + \frac{T}{2}) dt$$

$$= 5/24 \lambda^2 T^2$$

Assuming $\lambda = 0.056/\text{yr}$ or $0.047/\text{month}$ and $T = 2$ months will yield an average unavailability of $1.8E-05$.

The unavailability due to testing is estimated based on the following:

$$U = \lambda T_T$$

Where T_T is the duration of test, assumed to be 2 hours in this case. Therefore, the unavailability due to testing is $(6.3E-06)(2)$ or $1.26E-05$.

The total unavailability for the present system configuration and the diverse undervoltage (UV) driver is:

	<u>Present System</u>	<u>Diverse UV Driver</u>
Common Mode Failures	3.76E-05	1.46E-05
Random failures	1.80E-05	(a)
Testing	<u>1.26E-05</u>	<u>(a)</u>
Total:	<u>6.80E-05</u>	<u>1.46E-05</u>

(a) Only dominant contributors are considered.

As can be seen, there is a factor of $(6.80\text{E-}05/1.46\text{E-}05)$ or approximately 5 improvement in reduction of failure probability.

A.2 CORE DAMAGE (CD) ANALYSIS

The generic event trees for determining the change in core-damage (or core-melt) frequency are developed and are presented in Figures A.1 through A.8. The current system CD frequency and the modified system's CD frequency are estimated and results are presented.

As can be seen, the same generic event tree has been used for all the cases. The event trees differ from each other by virtue of the failure probability data used. Following is a brief discussion about the some of the event tree headings.

Number of transients (AT): Four significant transients per year have been assumed. This was based on a review of EPRI data. A review of this data and the Utility Group PRA revealed that approximately 70% of the transients involved turbine trip for which bypass to the condenser is available. These transients are both turbine trip initiators and those that would lead to a turbine trip. The remaining 30% of the transients are conservatively treated as loss of feedwater transients (or non-turbine trip transients).

MTC Overpressure: The moderator temperature coefficient (MTC) is a measure of the reduction in core reactivity as the water temperature increases (i.e., a negative temperature coefficient). It is usually expressed as PCM/F, which is the incremental change in reactivity per °F of moderator temperature increase divided by $1.0\text{E}+05$. As the reactor temperature increases during an ATWS, the MTC affects the efficiency of the moderator, causing a reduction in reactor power. For non-turbine trip transients, MTC was assumed to be an unfavorable value such that 10% of the time an ATWS would result in an unacceptable plant conditions. For turbine trip transients, an unfavorable MTC value was assumed to exist only 1% of the time since the pressure transient for this type of ATWS is relatively mild. For more rigorous treatment of MTC, refer to Westinghouse Owners Group Evaluation of this subject (see NUREG-0460).

AFWS Reliability: Auxiliary feedwater effectiveness is tied to the fraction of time for unfavorable MTC. With an unfavorable MTC, auxiliary feedwater alone can not result in successful coping with an ATWS. Conversely, auxiliary feedwater actuation in a timely manner is required to limit RCS pressure for most ATWS transients even with a favorable MTC. Based on AFW reliability studies performed for Westinghouse plants (NUREG-0460), the unavailability of all two or three trains of AFW was assumed to be about 0.001, if the initiation signal is given. This is assuming that all the affected Westinghouse plants have upgraded their plants for automatic actuation of AFW (NUREG-0737). For more detailed analysis of this subject, refer to SECY-83-292.

High Pressure Injection (HPI): Safety injection (SI) is actuated by (1) a manual signal from the operator, (2) a low pressurizer pressure signal, or (3) a high containment pressure signal. If any of these signals is present and if the reactor coolant system pressure is less than the shutoff head of the SI pumps, borated water can be pumped into the RCS by high pressure injection (HPI). Also, borated water can be added by the Chemical Volume and Control System (CVCS) and is available in the event that HPI fails. However, operator action is required

for successful boration by CVCS. An unavailability of 0.01 for HPI which corresponds to automatic actuation of SIS was assumed. This is consistent with Utility Group Study unavailability estimates.

As can be seen, Figures A.3 through A.6 present the effectiveness of the Diverse UV Driver on mitigating turbine and non-turbine trip transients. The reduction in core-damage (core-melt) frequency due to the "new" diverse UV driver is estimated to be $(8.3\text{E-}05/\text{py} + 3.5\text{E-}05/\text{py}) - (1.9\text{E-}05/\text{py} + 7.9\text{E-}06/\text{py})$ or $9.11\text{E-}05/\text{py}$.

Figures A.5 through A.8 represent the effectiveness of Diverse Initiation of AFW and Turbine Trip on mitigating turbine and non-turbine trip transients. As before, the reduction in core-melt frequency is estimated to be $(1.02\text{E-}05/\text{py} + 4.6\text{E-}06/\text{py}) - (3.3\text{E-}06/\text{py} + 1.4\text{E-}06/\text{py})$ or $1.01\text{E-}05/\text{py}$.

The total reduction in the core-melt frequency is, therefore, estimated to be $(9.11\text{E-}5/\text{py} + 1.01\text{E-}05/\text{py})$ or $1.01\text{E-}04/\text{py}$.

The public risk reduction is then $(1.01\text{E-}04/\text{py})(5.2\text{E}+06 \text{ man-rem})$ or 525 man-rem/py.

Figure A.1: Westinghouse Base Case Turbine Trip Transients
 Current Estimate SSPS Plants

Figure A.2: Westinghouse Base Case Non-Turbine Trip Transients
Current Estimate SSPS Plants

Figure A.4: Westinghouse Adjusted Case With Diverse U.V. Driver
Non-Turbine Trip Transients
Current Estimate SSPS Plants

Figure A.5: Westinghouse AMSAC (Option 1a) - Base-Case
Diverse Initiation of AFW and Turbine Trip
Turbine Trip Transients
Current Estimate SSPS Plants

Figure A.6: Westinghouse AMSAC (Option 1b) - Base-Case
 Diverse Initiation of AFW and Turbine Trip
 Isolation Transients (Non-Turbine Trip Transients)
 Current Estimate SSPS Plants

AT	RPS Elect	RPS Mech	MTC Over- pressure	AFWS Reliability	HPI	Unacceptable Consequences Probability
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Figure A.7: Westinghouse AMSAC (Option 1a) - Adjusted-Case
Diverse Initiation of AFW and Turbine Trip With
Diverse U.V. Driver
Turbine Trip Transients
Current Estimate SSPS Plants

AT	RPS Elect	RPS Mech	MTC Over- pressure	AFWS Reliability	HPI	Unacceptable Consequences Probability
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Figure A.8: Westinghouse AMSAC (Option 1a) - Adjusted-Case

Diverse Initiation of AFW and Turbine Trip With Diverse U.V. Driver
Isolation Transients (Non-Turbine Trip Transients)
Current Estimate SSPS Plants

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:
Enhancement of Reliability of Westinghouse Solid State Protection System (SSPS) (115)

2. Affected Plants (N):
19 operating Westinghouse design plants.

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

	$\bar{T}$ (yr)
PWR: Operating	25.5

4. Per-Plant Occupational Dose Reduction Due to Accident Avoidance $\Delta(FD_R)$:
PWR: (19,900 man-rem)(1.01E-04/py) = 1.94 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem)	
	Upper	Lower
9.7E+02	7.1E+03	0

6. Per-Plant Utility Labor in Radiation Zones for SIR Implementation:
Based on discussions with plant engineers, modifications to the RPS system do not require any work in the radiation zone area. Therefore, occupational exposure is assumed to be zero. Steps 7-12 are, therefore, omitted.

3.0 SAFETY ISSUE COSTS

The NRC and industry costs for SIR development, implementation, operation, and maintenance are presented in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Enhancement of Reliability of the Westinghouse Solid State Protection System (SSPS) (115).

2. Affected Plants (N):

19 operating Westinghouse design plants.

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

PWR: Operating $\frac{\bar{T} \text{ (yr)}}{25.5}$

Industry Costs (Steps 4 through 12)

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

PWR: $(\$1.65E+09)(1.01E-04/\text{py}) = \$1.67E+05/\text{py}$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$8.1E+07	\$6.2E+08	0

6. Per-Plant Industry Resources for SIR Implementation:

Cost is estimated directly in Step 7.

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

Based upon discussions with plant engineers the following costs have been identified:

1. Engineering analysis of the problem is estimated to take about 2 man-weeks. This is to document the procedures and changes done to the RPS. At \$2270/man-wk, this is estimated to cost \$4540.
2. Relays and other hardware are assumed to cost \$3000.
3. Installation is assumed to require 1 man-wk. This is estimated to cost \$2270. Since this modification can be completed during normal outage time, no replacement power cost has been included.
4. Possible changes to the technical specifications is assumed to require 4 man-wks. At \$2270/man-wk, this is estimated to cost \$9080.

The total cost of above requirements is estimated to be about \$20,000 per plant.

It is recognized that new modifications to the RPS system could result in spurious trips. These "unexpected" trips could result in economic loss to the industry due to the replacement power cost. If this is so, the overall cost of this resolution will be significant. This fact was, however, not considered in the cost analysis.

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

$$NI = (\$20,000/\text{plant})(19 \text{ plants}) = \$380,000$$

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

The affected plants are assumed not to require any additional operation/maintenance beyond that normally required for the plant. Therefore, this cost is zero. Steps 10-11 are, therefore, omitted.

12. Total Industry Cost (SI):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$3.8E+05	\$5.7E+05	\$1.9E+05

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Cost is estimated directly in step 14.

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

It is estimated that total NRC labor requirement for SIR development is 8 man-wks. At \$2270/man-wk, this is estimated to be \$18,160. The contract for SIR Research Program is assumed to be \$20,000. Therefore, total NRC cost for SIR development is:

$$(\$18,160) + (\$20,000) = \$3,8E+04.$$

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

Cost is estimated directly in step 16.

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

NRC costs in addressing licensing codes and standards and review of the technical specifications are estimated to be \$1.0E+04. This estimate includes a required 4 man-wk/plant of NRC labor at \$2270/man-wk.

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

$$NC = (\$1.0E+04/\text{plant})(19 \text{ plants}) = \$1.9E+05$$

18-20. Per-Plant NRC Labor for Review of SIR Operation and Maintenance:

No additional operation/maintenance review by NRC for the affected PWRs is assumed to be required beyond that normally required for the vessel.

21. Total NRC Cost (SN)

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$2.3E+05	\$3.2E+05	\$1.3E+05

REFERENCES (for Issue 115)

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.

McCormick, N. J. 1981. *Reliability and Risk Analysis Methods and Nuclear Power Applications*. Academic Press.

Memo from R. Bernero to T. Speis. "Enhancement of Reliability of Westinghouse Solid State Protection System." 1985. U.S. Nuclear Regulatory Commission. Washington. D.C.

Memo from W. J. Dircks to the Commissioners. "Amendment to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events." 1983. SECY-83-293. U.S. Nuclear Regulatory Commission. Washington. D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 117, Allowable Outage Times for Diverse, Simultaneous Equipment Outages

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: This generic safety issue addresses the potential for having simultaneous maintenance and/or testing outages for components that appear in the same accident sequence. The proposed resolution involves development of mini-computer based software that includes event trees and detailed fault trees of each nuclear power plant that would calculate core-melt frequencies based on prevailing plant configurations. The resulting "instantaneous" core-melt frequency would be used to make decisions regarding plant shutdown, component repair priorities, and allowable outage times for components.

AFFECTED PLANTS: All Plants (BWRs - 44; PWRs - 90)

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 1.2E+4

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	49

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	42
SIR Operation/Maintenance =	-8.6
Total of Above =	33
Accident Avoidance =	4.0

NRC COSTS:

SIR Development =	0.21
SIR Implementation =	0.30
SIR Operation/Maintenance Review =	0
Total of Above =	0.51

ALLOWABLE OUTAGE TIMES FOR DIVERSE, SIMULTANEOUS EQUIPMENT OUTAGES

ISSUE 117

1.0 SAFETY ISSUE DESCRIPTION

This issue addresses potential safety concerns related to outages of diverse components at nuclear power plants. Since it is not uncommon for a power plant to have several components out of service at the same time, the potential exists for operation of a plant in a dangerous configuration in which two or more components that appear in the same accident sequence are out of service. This issue focuses on outages for diverse components that are not necessarily in the same safety system, such as simultaneous outage of valves in the high pressure injection system (HPIS) and low-pressure/ containment spray injection system (LP/CSIS). In general, plant technical specifications address simultaneous outages of redundant components and are thus insufficient to address many of these potentially dangerous plant configurations. The analysis described herein provides estimates of the potential public risk reduction, occupational doses, and costs associated with implementing a system designed to prevent these plant configurations from occurring.

PROPOSED RESOLUTION

The proposed safety issue resolution (SIR) involves development of mini-computer based software for control room operators to use to track and evaluate prevailing plant configurations. The software would include plant-specific event trees and detailed fault trees. Operators would manually enter the prevailing plant configuration, such as valve alignments, switch settings, and equipment outages, and the software would calculate the plant's core-melt frequency. Being that the results would be based on the prevailing plant configurations, the core-melt frequency would be an "instantaneous" value, i.e., not a value that has been averaged over time.

After the instantaneous core-melt frequency has been calculated, the results would be compared with a set of critical values to assist the operators to make decisions regarding subsequent actions. The British have developed a comparable system for their Advanced Gas Reactors (AGRs) and have decided not display the core-melt frequency estimate. Instead, the computer compares the instantaneous core-melt frequency with two thresholds. Below an acceptable core-melt frequency, there is no limit on the amount of time the plant can continue to operate in the prevailing configuration. Above a certain critical core-melt frequency, the plant must shutdown because the current configuration represents one in which diverse safety systems may not be available to respond to one or more initiating events. In between, there is a finite allowable outage time within which the prevailing plant configuration is allowed to persist while the plant remains in service (Rowson 1985). In addition to providing input to decisions regarding plant shutdowns, the software could be interrogated to establish priorities for repairing components, and to aid in decisions on maintenance or surveillance priorities.

AFFECTED PLANTS

Resolution of this safety issue is assumed to affect all nuclear plants that are in operation as well as those under construction.

2.0 SAFETY ISSUE RISK AND DOSE

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

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issues:

Allowable Outage Times for Diverse, Simultaneous Equipment Outages (117)

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

All PWRs and BWRs are assumed to be affected.

	<u>N</u>	<u>$\bar{T}$ (yr)</u>
PWRs	90	28.8
BWRs	44	27.4
All	134	28.3

3. Plants Selected for Analysis:

Representative PWR: Oconee 3

Representative BWR: Grand Gulf 1

4. Parameters Affected by SIR:

The dominant minimal cut sets for Oconee 3 and Grand Gulf accident sequences were reviewed to identify core-melt sequences that contained two or more parameters in the same minimal cut sets that included equipment failures due to test/maintenance outages. All these affected parameters are listed in Attachment 1.

5. Base-Case Values for Affected Parameters:

The base-case values of the affected parameters are the same as those shown in Appendix A of NUREG/CR-2800 (Andrews, et al., 1983) and are listed in Attachment 1.

6. Affected Accident Sequences and Base-Case Frequencies:

Oconee-3:

	α (PWR-1) = 1.99E-8/py
	τ (PWR-3) = 3.98E-7/py
S_2D	β (PWR-5) = 1.45E-8/py
	ϵ (PWR-7) = 1.59E-6/py
	τ (PWR-3) = 7.10E-7/py
S_3D	β (PWR-5) = 1.04E-8/py
	ϵ (PWR-7) = 7.10E-7/py

T_2MQD τ (PWR-3) = 7.65E-7/py
 β (PWR-5) = 1.12E-8/py
 ϵ (PWR-7) = 7.65E-7/py

Grand Gulf:

T_1PQI α (BWR-1) = 1.47E-8
 δ (BWR-2) = 1.47E-6

 $T_{23}PQI$ α (BWR-1) = 3.69E-8
 δ (BWR-2) = 3.69E-6

 T_1PQE τ (BWR-3) = 1.20E-7
 δ (BWR-4) = 1.20E-7

 $T_{23}PQE$ τ (BWR-3) = 2.71E-7
 δ (BWR-4) = 2.71E-7

 SI α (BWR-1) = 4.60E-8
 δ (BWR-2) = 4.60E-6

 T_1QW δ (BWR-2) = 6.57E-6

 $T_{23}QW$ δ (BWR-2) = 1.21E-5

 T_1QUV τ (BWR-3) = 9.20E-7
 δ (BWR-4) = 9.20E-7

7. Affected Release Categories and Base-Case Frequencies:

<u>Oconee-3:</u>		<u>Grand Gulf:</u>	
PWR-1	1.99E-8	BWR-1	9.76E-8
PWR-3	1.87E-6	BWR-2	2.84E-5
PWR-5	3.61E-8	BWR-3	1.31E-6
PWR-7	3.07E-6	BWR-4	1.31E-6

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

$\bar{F}_{PWR} = 5.00E-6/py$ $\bar{F}_{BWR} = 3.11E-5/py$

See Attachment 1.

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

$W_{PWR} = 10.3 \text{ person-rem/py}$ $W_{BWR} = 209.7 \text{ person-rem/py}$

10. Adjusted-Case Values for Affected Parameters:

As discussed in Attachment 1, SIR is assumed to result in removal of the contributions of test/maintenance outage terms from certain component failures. The minimal cut sets affected by SIR include those that contain test/maintenance outage terms for two or more components in the same core-melt sequence. The test and maintenance outage contributions to component

(or parameter) unavailabilities were removed from the second (and succeeding) parameter in each affected core-melt sequence. See Attachment 1.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Oconee-3:

S_2D	α (PWR-1) = 1.95E-8/py
	τ (PWR-3) = 3.90E-7/py
	β (PWR-5) = 1.42E-8/py
	ϵ (PWR-7) = 1.56E-7/py
S_3D	τ (PWR-3) = 6.65E-7/py
	β (PWR-5) = 9.71E-9/py
	ϵ (PWR-7) = 6.65E-7/py
T_2MQD	τ (PWR-3) = 7.15E-7/py
	β (PWR-5) = 1.04E-8/py
	ϵ (PWR-7) = 7.15E-7/py

Grand Gulf:

T_1PQI	α (BWR-1) = 9.37E-9
	δ (BWR-2) = 9.37E-7
$T_{23}PQI$	α (BWR-1) = 3.62E-8
	δ (BWR-2) = 3.62E-6
T_1PQE	τ (BWR-3) = 9.65E-8
	δ (BWR-4) = 9.85E-8
$T_{23}PQE$	τ (BWR-3) = 2.61E-7
	δ (BWR-4) = 2.61E-7
SI	α (BWR-1) = 4.52E-8
	δ (BWR-2) = 4.52E-6
T_1QW	δ (BWR-2) = 6.40E-6
$T_{23}QW$	δ (BWR-2) = 1.18E-5
T_1QUV	τ (BWR-3) = 7.60E-7
	δ (BWR-4) = 7.60E-7

12. Affected Release Categories and Adjusted-Case Frequencies:

Oconee-3:

Grand Gulf:

Category	Frequency (py ⁻¹)	Category	Frequency (py ⁻¹)
PWR-1	1.95E-8	BWR-1	9.08E-8
PWR-3	1.77E-6	BWR-2	2.73E-5
PWR-5	3.43E-8	BWR-3	1.12E-6
PWR-7	2.94E-6	BWR-4	1.12E-6

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

$$\bar{F}_{PWR}^* = 4.76E-6/\text{py}$$

$$\bar{F}_{BWR}^* = 2.96E-5/\text{py}$$

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

$$W_{PWR}^* = 9.71 \text{ person-rem/py}$$

$$W_{BWR}^* = 200.6 \text{ person-rem/py}$$

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

$$\Delta\bar{F}_{PWR} = 2.4E-7/\text{py}$$

$$\Delta\bar{F}_{BWR} = 1.5E-6/\text{py}$$

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

$$\Delta W_{PWR} = 0.55 \text{ person-rem/py}$$

$$\Delta W_{BWR} = 9.1 \text{ person-rem/py}$$

17. Total Public Risk Reduction, (ΔW)_{Total}:

Best Estimate
(man-rem)

Error Bounds (man-rem)
Upper Lower

1.2E+4

8.4E+6

0

ATTACHMENT 1 (To Table 1)

This generic safety issue addresses the potential for having simultaneous maintenance and/or testing outages for components that appear in the same accident sequence. Since it is not uncommon for several components at nuclear power plants to be out of service, potentially dangerous conditions may exist in which several diverse components that appear in the same accident sequence are out of service. This issue is more broad than two redundant components being out of service at the same time, which are typically addressed in plant technical specifications and limiting conditions for operation. The proposed resolution involves development of mini-computer based software that includes event trees, core melt sequences, and minimal cut sets for each nuclear power plant. The software would calculate core-melt frequencies for prevailing plant configurations (e.g., valve alignments, switch settings, components out of service). The resulting "instantaneous" core-melt frequency would be used to make decisions regarding plant shutdown, component repair priorities, and allowable outage times for components.

The primary purpose of the software would be to prevent simultaneous outages of diverse equipment whose failures are included in the same accident sequences. Plant technical specifications are generally focused on ensuring the operability of safety systems by ensuring that at least one out of a redundant set of components or equipment trains is available. However, technical specifications do not necessarily consider the operability of other safety systems when addressing allowable outage times for components in a specified safety system. For example, technical specifications and limiting conditions for operation (LCOs) for the high pressure injection system (HPIS) at a PWR, which generally address allowable outage times for HPIS components, do not necessarily consider whether or not the low pressure/containment spray injection system (LP/CSIS) is operable. Thus, plant risk will be improved as a result of improving component unavailabilities by eliminating equipment failures due to test/maintenance outages from the second (and subsequent) terms of the dominant accident sequences.

The first step taken to model the effects of SIR on public risk reduction was to review in detail each minimal cut set given for Oconee 3 and Grand Gulf in NUREG/CR-2800, Appendix A (Andrews, et al., 1983). The purpose of the review was to identify core-melt sequences that included two or more cut set elements that contained test/maintenance outage terms. These are the cut sets that will be affected by this issue. Only accident sequences S_2D , S_3D , and T_2MQD contained cut sets with elements having 2 or more maintenance outages. For Grand Gulf, the affected accident sequences were more extensive. All of the accident sequences except for $T_{23}C$ were determined to be affected.

The cut sets in the affected accident sequences were then modified to approximate the effects of SIR implementation. The modifications removed the multiple maintenance outage sequence as a possible cause of failure of two successive components. To do this, we define two cut set elements, A and B, as follows:

$$A = A_F + A_M$$

$$B = B_F + B_M$$

where A_M and B_M are the test and maintenance unavailabilities of terms A and B and A_F and B_F are the sum of all other failures (including hardware, failure, operator error, control circuitry faults, etc.)

The probability of simultaneous failure of A and B can be written as follows:

$$\begin{aligned} A \cdot B &= (A_F + A_M) \cdot (B_F + B_M) \\ &= A_F B_F + A_F B_M + A_M B_F + A_M B_M \end{aligned}$$

The product is thus the sum of four terms. Three of the terms include either the probability of failure of A, the probability of failure of B, or both. The fourth term is the product of test/maintenance unavailabilities of both components. Removal of this latter product from the expansion of $A \cdot B$ approximates the effect of SIR. The hardware, test, and maintenance unavailabilities for the minimal cut set elements affected by this issue are shown on the next page of this attachment.

An example of the method described above will be performed to illustrate the calculations. The cut set used in this example is $S_2 \cdot C1 \cdot B1$. Using the subscript notations "F" and "M" defined above, the following expressions were developed for C1 and B1.

$$C1 = C1_F + C1_M = 0.0077 + 0.0021 = 0.0098$$

$$B1 = B1_F + B1_M = 0.0268 + 0.0082 = 0.035$$

Given that the value of S_2 is $4E-4/\text{py}$, the base-case value for cut set $S \cdot C1 \cdot B1 = 1.4E-7/\text{py}$.

Unavailabilities of Minimal Cut Set Elements Affected by this Issue

<u>Boolean Term</u>	<u>Total Unavailability</u>	<u>Contribution of Maintenance</u>	<u>Test</u>
Oconee-3			
B	0.0033	0.0021	
C	0.0033	0.0021	
D	0.023	0.0063	0.0019
E	0.023	0.0063	0.0019
A1	0.0098	0.0021	
B1	0.035	0.0063	0.0019
C1	0.0098	0.0021	
CH1	0.0050	0.0021	
CH2	0.0050	0.0021	
CH3	0.0050	0.0021	
CH4	0.0050	0.0021	
Grand Gulf			
DIESEL1	0.036	0.0064	
DIESEL2	0.036	0.0064	
DIESEL3	0.036	0.0064	
VGA2	0.024	0.0174	
VGB2	0.024	0.0174	
VGA1	0.015	0.0116	
VGB1	0.015	0.0116	
SA	0.014	0.0116	
SB	0.014	0.0116	
SSA	0.021	0.0174	
SSB	0.021	0.0174	
SSC	0.014	0.0116	
LA2	0.014	0.0116	
LB2	0.014	0.0116	
LC	0.022	0.0174	
LB1	0.013	0.0116	
V1	0.008	0.0058	
V2	0.008	0.0058	
HACT	0.0112		0.00023
RACT	0.0112		0.00023
LRACT	0.0112		0.00023
BCACT	0.0112		0.00023
SAACC	0.0112		0.00023
SBACC	0.0112		0.00023
SAC	0.0112		0.00023
SBC	0.0112		0.00023
SCC	0.0112		0.00023
SCVA	0.032	0.023	
SCVB	0.032	0.023	
H	0.021	0.0174	
L	0.021	0.0174	
R	0.051	0.041	

The next step was to remove the cut set involving the double-maintenance outage term or:

$$\begin{aligned}
 (S_2 \cdot C1 \cdot B1)_{\text{Adjusted}} &= (S_2 \cdot C1 \cdot B1)_{\text{Base Case}} - (S_2 \cdot C1_M \cdot B1_M) \\
 &= 1.4E-7 - (4E-4)(0.0021)(0.0082) \\
 &= 1.3E-7/\text{py}
 \end{aligned}$$

This process was repeated for all cut sets in the accident sequences that contained two or more elements with test/maintenance unavailabilities. The adjusted-case cut set values in these two accident sequences were then summed to develop adjusted case values for the affected accident sequences. The effects of SIR on core-melt frequency and public risk were then estimated using the methods and data given in NUREG/CR-2800.

Double maintenance outage terms were removed from several accident sequences in the original Oconee and Grand Gulf PRAs and thus had already been removed from the base-case. Oconee minimal cut sets that included elements E · C and B · D and Grand Gulf cut sets with the elements shown on pp. B.27 and B.28 of NUREG/CR-2800 were not affected by this issue, unless the affected cut sets contained unavailabilities due to simultaneous testing of two components or due to testing of one component and maintenance of another. These combinations of test and maintenance unavailability had not been removed from the base-case. For these cut sets, the procedure described above was followed to remove the double-test and test-maintenance combinations from the affected cut sets.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Allowable Outage Times for Diverse, Simultaneous Equipment Outages (117)

2. Affected Plants (N):

All PWRs and BWRs are assumed to be affected.

	<u>N</u>
PWRs	90
BWRs	44
All	134

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

	<u>$\bar{T}$ (yr)</u>
PWRs	28.8
BWRs	27.4
All	28.3

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

PWR = (19,860 person-rem) (2.4E-7/py) = 0.0048 person-rem/py
 BWR = (19,860 person-rem) (1.5E-6/py) = 0.030 person-rem/py

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

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
49	6.3E+3	0

6-12. Steps Related to Occupation Doses for SIR Implementation and Operation/Maintenance:

SIR involves development of a mini-computer based software system to be installed in the control rooms of nuclear power plants. These activities would not significantly affect the occupational dose commitments at power plants, i.e., implementation and operation of the software system would involve no additional work in radiation zones. SIR does not affect the frequency or duration of test/maintenance outages, only their timing. Therefore, the occupational dose increase for SIR implementation, operation, and maintenance would be 0.

3.0 SAFETY ISSUE COSTS

This SIR involves development and implementation of a mini-computer based probabilistic risk assessment (PRA) of each nuclear power plant that is capable of performing analyses of the "instantaneous" core-melt frequency of a plant based on prevailing conditions. It is assumed that the costs to perform a pilot study on a selected plant and for developing a regulatory position and reviewing licensing changes are the responsibility of the NRC. The costs to develop and implement the computer software at all plants are assumed to be borne by the industry.

NRC costs are assumed to include research and development (R&D) costs to develop and demonstrate the software for a selected plant. Because some work related to this pilot study is currently being performed (i.e., the SARA computer code currently under development), the NRC will realize some increased efficiency in developing the software addressed by this SIR. Portions of the SARA code would be directly applicable, such as coding of the minimal cut sets, accident sequences, and containment failure probabilities. Additional efforts by a contractor would be needed for further development, such as display of the instantaneous core-melt frequency and comparison with critical thresholds; enhancing the ability of the user to remove components from service; development of algorithms to perform the assessments of allowable outage times; and development of algorithms to aid the user in setting component repair priorities. It is estimated that approximately 2 man-yrs of contractor labor would be needed to develop and demonstrate the pilot software system, given the advanced stage of the SARA code.

This SIR will also be assisted by the ongoing research related to the Severe Accident Policy Statement of the NRC. The current draft calls for further severe accident safety analyses of operating plants as well as PRAs for new plants. The NRC expects the studies from operating plants would be similar in scope and level of detail to those conducted under the Interim Reliability Evaluation Program (IREP), which would be sufficient to serve as the basis of the computer software. In addition, specification of the critical core-melt frequency thresholds could potentially be obtained from the results of this ongoing research. As a result, the costs of the PRAs and research needed to specify the core-melt frequency thresholds is not included in the costs of this SIR.

Assuming the pilot software system is successful and the NRC decides it is a valuable tool for the industry, a rulemaking would be initiated to require the system to be implemented at all nuclear power plants. Approximately 1 man-month of NRC labor is assumed to be required to develop a regulatory position and that an additional 1 man-wk per plant is required to review the licensing changes at each plant. Furthermore, it is assumed that approximately 0.5 man-wk per plant per year of NRC labor would be needed to periodically review the operation of the software system.

Industry costs will consist of software development, verification, and validation, purchase and installation of a mini-computer, and operator training. Industry resources for software development are estimated at approximately 2 man-yrs. This estimate considers the efficiencies to be obtained from successful development of the pilot software. It also considers that an IREP-level severe accident assessment will be performed for each plant. All that is necessary is

to encode the component unavailabilities and failure modes, minimal cut sets, and containment failure probabilities determined by the plant-specific PRA as well as other plant-specific effects. Software verification and validation (V&V) is anticipated to be significant, being that important decisions would be made based on the results provided. Therefore, it is assumed that V&V costs amount to 40% of the initial development costs, or about 0.8 man-yr of labor. This V&V is above and beyond the V&V efforts that would be performed for the PRA in any event.

Industry costs to purchase and install a mini-computer in the reactor control rooms are also needed. A relatively powerful mini-computer would be needed to store and operate the software. An example system is costed out below that is believed to be adequate to run the software envisioned in this SIR:

<u>Product Description</u>	<u>Price (\$)</u>
Microcomputer 640K main memory, keyboard, disk drives	2,500
RGB color monitor	350
Color (or enhanced) graphics adapter	100
10 Mbyte hard disk	1,000
Multi-function board	400
Dot matrix graphics printer	450
Math coprocessor	200
Software (operating system)	50
Other (cabinet, stand, surge protector, dust cover, etc.)	<u>200</u>
Total:	<u>5,250</u>

In addition to installation and purchase costs for the mini-computer system, utility costs will include the costs for training operators in the use of the software system. It is anticipated that the software can be made user-friendly to the point at which the operators need not be experts in PRA techniques to operate the software. Therefore, the operators would require training in the basics of reliability analysis and in operation of the software system. It is estimated that one-week short courses in reliability engineering/PRA would be needed as well as a second one-week short course in operation of the software. It is assumed that three operators from each power plant would receive this training. Thus, a total of 6 man-weeks of utility labor is needed for this training. In addition, the costs of the instructors and registration for the short courses are included in the utility costs. These costs are assumed to include: burdened labor costs for the instructor; \$100/day for living expenses (the instructor is assumed to travel to the plant); \$500 for travel costs; and 15% profit. This brings the total costs for each 5-day training course to about \$3,500.

The implementation of this SIR offers the potential for industry and NRC costs savings. Because the software would be capable of calculating allowable component outage times and would allow operators to interrogate the system to determine the effects of taking additional components out of service, it is postulated here that temporary emergency exemptions from limiting conditions for operation would almost never be necessary. It is assumed that the costs of one temporary exemption per plant per year would be saved by implementing this SIR. It is estimated that each exemption, on average, requires approximately 1 man-wk of utility labor to perform the necessary analyses and submit the request for a

temporary exemption and about 0.5 man-wks of NRC labor to review and evaluate the request. These costs are subtracted from the total costs of operation and maintenance of this SIR.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Allowable Outage Times for Diverse, Simultaneous Equipment Outages (117)

2. Affected Plants (N):

All PWRs and BWRs are assumed to be affected.

	<u>N</u>
PWRs	90
BWRs	44
All	134

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

	<u>$\bar{T}$ (yr)</u>
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, $\Delta(\bar{F}A)$:

$$\begin{aligned} (\text{PWR}) &= (\$1.65\text{E}+9) (2.4\text{E}-7/\text{py}) = \$4.0\text{E}+2/\text{py} \\ (\text{BWR}) &= (\$1.65\text{E}+9) (1.5\text{E}-6/\text{py}) = \$2.5\text{E}+3/\text{py} \end{aligned}$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$4.0E+6	\$5.0E+8	0

6. Per-Plant Industry Resources for SIR Implementation:

SIR implementation costs consist of software development, verification and validation, computer system purchase and installation, and operator training costs. The resources necessary for these items are discussed in the text and are summarized below.

Software development:	2 man-yrs/plant
Software V&V:	<u>0.8 man-yrs/plant</u>
Total:	<u>2.8 man-yrs/plant</u>

Computer system purchase/installation:	\$5,250/plant
Training	
- staff resources	6 man-wks/plant
- instructor, course fees	\$7,000/plant

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

Per-plant Costs, \$

Software	280,000
Computer purchase	5,250
Training	13,260
Course fees	7,000
Total:	<u>310,000</u> (rounded)

$$I \text{ (all plants)} = \$3.1\text{E}+5/\text{plant}$$

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

$$\begin{aligned} NI &= (90 + 44) (\$3.1\text{E}+5/\text{plant}) \\ &= \$4.2\text{E}+7 \end{aligned}$$

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

Labor requirements are addressed in Step 10.

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

The costs for actual operation of the computer system are expected to result in negligible incremental work for plant operators. Use of the system is anticipated to be integrated into the normal operating routine of the plant. However, an operating cost savings is anticipated to be realized through reductions in the number of temporary emergency exemptions to limiting conditions for operation. This amount to approximately 1 man-wk/py, as discussed in the text.

$$I_o = (1 \text{ man-wk/py}) (\$2,270/\text{man-wk}) = -\$2,270/\text{py}$$

(Note: Negative (-) sign indicates cost savings)

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

$$\begin{aligned} NTI_o &= (\$2,270/\text{py}) [(90 \text{ PWRs})(28.8 \text{ yrs}) + (44 \text{ BWRs})(27.4 \text{ yrs})] \\ &= -\$8.6\text{E}+6 \end{aligned}$$

(Note: Negative (-) sign indicates cost savings)

12. Total Industry Cost (S_t):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$3.3E+7	\$5.4E+7	\$1.2E+7

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

Development of pilot software: 2 man-yrs
Development of regulatory position: 1 man-month

See text for additional discussion.

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

$$C_D = (2 \text{ man-yrs})(\$1.0E+5/\text{man-yr}) + (4 \text{ man-wks})(\$2,270/\text{man-wk}) \\ = \$2.1E+5$$

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

1 man-wk/plant of NRC labor to review licensing changes.

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

$$C = (1 \text{ man-wk/plant})(\$2,270/\text{man-wk}) = \$2,270/\text{plant}$$

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

$$NC = (134 \text{ plants})(\$2,270/\text{plant}) = \$3.0E+5$$

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

NRC resources of 0.5 man-wk/py are estimated to be required for review of software operations. This is offset entirely by a savings of 0.5 man-wk/py associated with reductions in the number of temporary emergency exemptions applied for by utilities.

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

The net per-plant NRC cost for review of SIR operation and maintenance is 0.

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

The total NRC costs for review of SIR operation and maintenance is 0.

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$5.1E+5	\$6.9E+5	3.3E+5

REFERENCES (For Issue 117)

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.

Rowsome, F. H. 1985. *Candidate Generic Safety Issue: Allowable Outage Times for Diverse, Simultaneous Equipment Outages*. Memo to W. Minners, U.S. Nuclear Regulatory Commission, Washington D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 120, On-Line Testability of Protection Systems

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: This issue relates to on-line testability of protection systems and the possibility that some nuclear power plants may not comply with existing regulatory requirements in this area even though plant FSARs indicate otherwise. The proposed resolution recognizes that in some cases there is no practical design modification that will allow on-line testing of these systems. In these cases, the utilities would be required to submit information that describes and justifies deviations from these requirements. Other affected plants would be required to design and implement modifications that would ensure full compliance with these regulatory requirements.

AFFECTED PLANTS: PWR: Operating = 8 Planned = 0
BWR: Operating = 34 Planned = 0

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION	4.8E+03
OCCUPATIONAL DOSES	
SIR Implementation	4.3E+01
SIR Operation/Maintenance	4.6E+02
Total of Above	5.0E+02
Accident Avoidance	17.4

COST RESULTS (\$10⁶):

INDUSTRY COSTS

SIR Implementation	2.9
SIR Operation/Maintenance	10.0
Total of Above	13.0
Accident Avoidance	2.8

NRC COSTS

SIR Development	0.014
SIR Implementation	1.1
SIR Operation/Maintenance Review	5.2
Total of Above	6.3

ON-LINE TESTABILITY OF PROTECTION SYSTEMS

ISSUE 120

1.0 SAFETY ISSUE DESCRIPTION

The issue relates to on-line testability of protection systems and resulted from the NRC staff's review of recent designs. These reviews discovered that the designs do not comply with existing regulatory requirement on testability even though the FSARs indicated otherwise. A fundamental regulatory practice requires that plants be designed with the capability to permit testing of protection systems while the plant is operating at power to assure that the required reliability of protection system components is maintained during the life of the power plants.

The design bases for testing of protection systems include the recommendation of Regulatory Guide 1.22 and IEEE Standard 338. The engineered safety features actuation system (ESFAS) subgroup relays are to be tested at 6-month intervals. The subject plant designs appear not to include appropriate features to facilitate proper testing during power operations as required. Typically, the reactor protection system contains 100 relays for testing. In the plants that do not comply with the regulations, approximately 20 of those can not be tested while the plant is on-line.

PROPOSED RESOLUTION

For this analysis, the proposed resolution to this safety issue is assumed to consist of the following options:

1. Recognize that there are cases where there is no practical system design modification that will permit at-power operation of the actuated equipment without adversely affecting the safety or operability of the plant. Exceptions could be taken that include not testing the automatic initiating logic and associated actuating devices. Actions could include the submittal of information by the licensees to describe and justify any deviations from regulatory requirements and to describe the revision of the plant technical specifications stating the testing required. Test those systems that can be tested without defeating the ESFAS train.
2. Design and implement modifications to allow compliance with the requirements for on-line testing of all systems without defeating the ESFAS train. Each channel of the reactor trip module (RTM) needs to be provided with two-key operated bypass switches, a channel bypass switch and a shutdown by-pass switch. The 2/4 system would then operate in the 2/3 mode during the testing.

AFFECTED PLANTS

Eight operating PWRs and all 34 operating BWRs are assumed to be affected by this issue. The PWRs are San Onofre 2 and 3, Waterford 3, Palo Verde 1, 2 and 3, Callaway 1, and Wolf Creek. None of the planned reactors are assumed to be affected as it is assumed that the design of future plants will include the capability to test all relays and breakers in the reactor protection system.

TABLE 1. Public Risk Reduction Work sheet

1. Title and Identification Number of Safety Issue:

On-Line Testability of Protection Systems (120)

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

Operating PWRs = 8 ($T = 27.7$ yr)	Planned PWRs = none ¹
Operating BWRs = 34 ($T = 25.2$ yr)	Planned BWRs = none ¹

3. Plants Selected for Analysis:

Representative PWR: Oconee-3
Representative BWR: Grand Gulf 1

4. Parameters affected by SIR:

For the representative PWR, the affected parameter is K, Reactor Protection System.

For the representative BWR, the affected parameter is C, Failure To Render The Reactor Subcritical.

5. Base-Case Values for Affected Parameters:

The base-case for this issue assumes the plants are in full compliance with protection system testing requirements. It is assumed that this was the basis for estimation of protection system unavailabilities during preparation of the Oconee-3 and Grand Gulf-1 PRAs. See Attachment 1.

Oconee:	K (base-case frequency) = 3.0E-6/py
Grand Gulf 1:	C (base-case frequency) = 9.2E-7/py

6. Affected Accident Sequences and Base-Case Frequencies:

Oconee 3:	T_2 KMU - τ (PWR - 3)	4.5E-06/py
	T_2 KMU - β (PWR - 5)	6.6E-08/py
	T_2 KMU - ϵ (PWR - 7)	4.5E-06/py

Grand Gulf 1:	T_{23} C - δ (BWR - 2)	6.4E-06/py
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Note: For both Oconee-3 and Grand Gulf-1, the non-dominant minimal cut set contributions to the affected accident sequences are 0.

¹

Not included because this is not a generic issue for PWRs and licensee is expected to include in design.

7. Affected Release Categories and Base-Case Frequencies:

Oconee 3: (PWR - 3) 4.5E-06/py
 (PWR - 5) 6.6E-08/py
 (PWR - 7) 4.5E-06/py

Grand Gulf 1: (BWR - 2) 6.4E-06/py

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

$\bar{F}_{PWR} = 9.1E-06/py$ $\bar{F}_{BWR} = 6.4E-06/py$

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

$W_{PWR} = 24.4 \text{ man-rem/py}$ $W_{BWR} = 30.7 \text{ man-rem/py}$

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

The adjusted case values of affected parameters K (for Oconee-3) and C (for Grand Gulf-1) are those given in NUREG/CR-2800 (Andrews, et al., 1983). See Attachment 1.

Oconee-3 K = 2.6E-5
 Grand Gulf-1 C = 7.7E-7

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Affected accident sequences the adjusted-case frequencies are discussed in Attachment 1. The adjusted-case PWR sequences are:

Oconee 3: $T_2KMU - \tau$ (PWR - 3) 3.9E-06/py
 $T_2KMU - \beta$ (PWR - 5) 5.7E-08/py
 $T_2KMU - \epsilon$ (PWR - 7) 3.9E-06/py

Grand Gulf 1: $T_{23}C - \delta$ (BWR - 2) 5.4E-06/py

12. Affected Release Categories and Adjusted-Case Frequencies:

Oconee 3: (PWR - 3) 3.9E-06/py
 (PWR - 5) 5.7E-08/py
 (PWR - 7) 3.9E-06/py

Grand Gulf 1: (BWR - 2) 5.4E-06/py

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

$\bar{F}^*_{PWR} = 7.9E-06$ $\bar{F}^*_{BWR} = 5.4E-06$

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

$W^*_{PWR} = 21.1 \text{ man-rem/py}$ $W^*_{BWR} = 25.9 \text{ man-rem/py}$

15. Reduction in Core-Melt Frequency (ΔF):

$$\Delta F_{PWR} = 1.2E-06/\text{py}$$

$$\Delta F_{BWR} = 1.0E-06/\text{py}$$

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

$$\Delta W_{PWR} = 3.3 \text{ man-rem/py}$$

$$\Delta W_{BWR} = 4.8 \text{ man-rem/py}$$

17. Total Public Risk Reduction, (ΔW)_{Total}

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
4.8E+03	9.5E+05	0

ATTACHMENT 1 (to Table 1)

This analysis uses the Oconee-3 Probabilistic Risk Assessment (PRA) as a basis for evaluation of PWRs and the Grand Gulf-1 PRA is used to represent BWRs. Estimates of the public risk and the core-melt frequency reductions presented in this analysis are based on the results in NUREG/CR-2800.

The reductions in public risk and core-melt frequency were estimated based on reductions in failure rates for relays in the reactor protection system (RPS) contribution that would result from implementation of the SIR. The Engineered Safety Feature Actuation System (ESFAS) consists of sensors, logic and actuation signals to circuits which monitor selected plant parameters and provide an actuating signal to safety systems. Group relays actuate the engineered safety features which generally consist of solenoid operated valves, motor operated valves or pump motors. For this analysis, the relays are assumed to be part of the control circuitry.

For this analysis, it was assumed that the results of the Oconee-3 PRA were derived assuming the test interval for all relays in the RPS is 6 months and that the plant is in full compliance with on-line testing requirements. Thus, the base-case represents the situation in which a fraction of the relays can be tested only during refueling outages or other extended shutdowns. An 18-month test interval for these relays was assumed. As a result, the adjusted-case is represented by the original results for Oconee-3 as presented in NUREG/CR-2800, Appendix A. Similarly, the original results for Grand Gulf-1 (NUREG/CR-2800, Appendix B) represent the adjusted-case for BWRs.

For Oconee-3, the affected parameter, "K", appears in only 1 accident sequence, T₂KMU. The adjusted-case frequency of this accident sequence is dominated by one minimal cut set, as shown below:

<u>Cut Set</u>	<u>Base-Case Event Frequency Or Probability</u>	<u>Base-Case Cut-set Frequency</u>
T ₂	3/py	7.8E-06/py
K	2.6E-5	
M	1.0	
HPMAN1	0.1	

The events in this cut set are defined as follows:

<u>Event</u>	<u>Description</u>
T ₂	Loss of power conversion system transient caused by other than a loss of offsite power.
HPMAN1	Operator failure to start the HPIS during an ATWS sequence (extremely high stress).
K	Failure of reactor protection system due to primarily test and maintenance faults.
M	Interruption of PCS.

The RPS fault tree shown in NUREG/CR-2800 provides an unavailability value, Q, of 2.6E-5/demand for the reactor protection system. This value represents the adjusted-case value of K. Resolution of the safety issue involves assuring that all relays are tested at 6-month intervals rather than the current practice at the affected plants of testing a fraction of the relays only during plant shutdowns. To calculate the base-case value of Q for a change in test frequency from 6 to 18 months, relay unavailability data from ANO-2 for the two frequencies of testing is used. It was assumed that the testing of 100 relays instead of 80 relays will increase the unavailability of 1 of 4 Reactor Trip Modules (RTMs) by 25%. The relay unavailability from ANO-2 data based on the testing interval is:

$$\begin{aligned} Q &= 7.2E-4/\text{demand} \\ Q \text{ (6-months)} &= 7.2E-4/\text{demand} \\ Q \text{ (18-months)} &= 2.2E-3/\text{demand} \end{aligned}$$

Recalculating the base-case frequency using the value of K derived from ANO-2 data results in a value of 2.96E-5 for an eighteen month testing period. The adjusted-case value of K was 2.6E-5 for the six month testing period. These values were substituted into the Oconee-3 risk equations to develop estimates of the change in core-melt frequency and public risks associated with SIR.

The dominant cut sets for Grand Gulf 1 which involve reactor transients including grid disturbances, along with their original, base-case frequencies are shown below:

<u>Base-Case Cut-set Element</u>	<u>Base-Case Event Frequency/ Probability</u>	<u>Cut-Set Frequency</u>
T ₂₃	7.0/py	5.4E-6/py
C	7.7E-7/demand	

The events in these cut sets are defined as follows:

<u>Event</u>	<u>Description</u>
T ₂₃	Any transient other than a loss of offsite power which requires an emergency reactor shutdown.
C	Failure to render the reactor subcritical.

To obtain the base-case frequency for the hypothetical BWR, it was assumed that the change in C can be scaled from the fault tree results calculated for the hypothetical PWR. It was assumed that the value in the Grand Gulf PRA is for a testing frequency of 6-months. Using this assumption, the base-case value of C for an 18-month frequency of testing is 9.2E-7. The adjusted-case value of C is shown above and represents the original value for Grand Gulf-1 as given in NUREG/CR-2800, Appendix B. These values were substituted into the Grand Gulf-1 risk equations to estimate the changes in core-melt frequency and public risk for affected BWRs.

TABLE 2 - Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

On-Line Testability of Protection Systems (120)

2. Affected Plants (N):

Eight PWRs and thirty-four BWRs are assumed to be affected.

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

Operating PWRs $\bar{T} = 27.7$ yr

Operating BWRs $\bar{T} = 25.2$ yr

All operating plants $\bar{T} = 26.9$ yr

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

PWR: $\Delta(\bar{F}D_R) = (19,860 \text{ man-rem}) (1.2\text{E-}6/\text{py}) = 2.4\text{E-}02 \text{ man-rem/py}$

BWR: $\Delta(\bar{F}D_R) = (19,860 \text{ man-rem}) (1.0\text{E-}6/\text{py}) = 2.0\text{E-}02 \text{ man-rem/py}$

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

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
17.4	894.0	0

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

The plants affected by this issue are assumed to be divided into two groups. The first group (Group 1) consists of plants at which no design modifications are possible that would permit testing of the RPS at full power. The second group (Group 2) consists of plants that could possibly implement design modifications that would permit this testing. The affected plants are assumed to be divided equally into these two groups.

For Group 1 plants, SIR is estimated to involve 1 man-wk of utility labor in a radiation zone to inspect the non-testable relays and review the system design. Group 2 plants would be subjected to this review and, in addition, would require an estimated 10 man-wks to install the design modifications and 4 man-wks to test the modified system.

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

It was assumed that testing is performed outside containment so the dose rate is 2.5 mrem/hr. It was assumed that the work involves a 75% utilization factor.

Group 1:

$$D = (0.0025 \text{ R/hr})(40 \text{ man-hr/plant})/(0.75) = 0.13 \text{ man-rem/plant}$$

Group 2:

$$D = (0.0025 \text{ R/hr})(600 \text{ man-hr/plant})/(0.75) = 2.0 \text{ man-rem/plant}$$

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

$$\begin{aligned} ND &= (21 \text{ plants})(0.13 \text{ man-rem/plant}) \\ &+ (21 \text{ plants})(2.0 \text{ man-rem/plant}) = 42.6 \text{ man-rem} \end{aligned}$$

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

Group 1: SIR operation and maintenance would presumably require additional inspection activities during plant outages associated with assuring the operability of the relays in the RPS. It is estimated that an additional 4 man-hrs per relay (i.e., those 20 relays that cannot be tested at power) per six-months would be required for this activity. Thus, a total of 160 man-hrs/py would be required for Group 1 plants. Assuming a 75% utilization factor for labor in radiation zones, the total annual utility labor is estimated at about 210 man-hrs/py.

Group 2: It is assumed that for the affected LWRs in Group 2, operation requires 2 man-hr/six-months to test the additional 20 relays and breakers (because the design modifications should increase the efficiency of inspecting and testing relays relative to Group 1 plants). Using a 75% utilization factor, this labor requirement becomes 110 man-hr/py in a radiation zone.

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

Group 1:

$$D_o = (0.0025 \text{ R/hr})(210 \text{ man-hr/py}) = 0.53 \text{ man-rem/py}$$

Group 2:

$$D_o = (0.0025 \text{ R/hr})(110 \text{ man-hr/py}) = 0.28 \text{ man-rem/py}$$

11. Total Occupational Dose Increase for SIR Operation and Maintenance (NTD_o):

$$\begin{aligned} NTD_o &= (21 \text{ plants})(26.9 \text{ yr})(0.53 \text{ man-rem/py}) \\ &+ (21 \text{ plants})(26.9 \text{ yr})(0.28 \text{ man-rem/py}) = 4.6E+2 \text{ man-rem} \end{aligned}$$

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
5.0E+02	1.5E+03	1.7E+02

3.0 SAFETY ISSUE COSTS

The industry and NRC costs associated with resolution of this safety issue 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:

On-Line Testability of Protection Systems (120)

2. Affected Plants (N):

Eight PWRs and thirty-four BWRs are assumed to be affected.

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

Operating PWRs $\bar{T} = 27.7$ yr

Operating BWRs $\bar{T} = 25.2$ yr

All operating plants $\bar{T} = 26.9$ yr

Industry Costs (Steps 4 through 12):

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

Using \$1.65E+09 as the cost of reactor cleanup, repair/refurbishment, and replacement power, the industry cost savings for each reactor is

PWR: $\Delta(\bar{F}A) = (1.2E-06/\text{py})(\$1.65E+09) = \$1,980/\text{py}$

BWR: $\Delta(\bar{F}A) = (1.1E-06/\text{py})(\$1.65E+09) = \$1,650/\text{py}$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$2.8E+06	\$7.4E+07	0

6. Per-Plant Industry Resources for SIR Implementation:

As with the occupational dose increase, industry costs are divided into two plant groups. The industry resources for SIR implementation by these two groups are as follows:

Group 1:

· Inspection/review of current plant configuration	- 1 man-wk
· Research design modifications	- 3 man-wks
· Analyze/justify deviations from requirements	- 4 man-wks
· Technical specification change and associated technical/legal/administrative support	- 8 man-wks
Total:	<u>16 man-wks</u>

Group 2:

. Inspection/review of current plant config.	- 1 man-wk
. Design modifications	- 3 man-wks
. Install and test design modifications	- 16 man-wks
. Revise testing procedures	<u>2 man-wks</u>
Total:	<u>21 man-wks</u>

. Hardware costs (estimate)	\$50,000/plant
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7. Per-Plant Industry Cost for SIR Implementation (I):

Group 1:

$$\begin{aligned}\text{Labor} &= (16.0 \text{ man-wk/plant})(\$2270/\text{man-wk}) \\ &= \$3.6\text{E}+04/\text{plant}\end{aligned}$$

Group 2:

$$\begin{aligned}\text{Labor} &= (22.0 \text{ man-wk/plant})(\$2270/\text{man-wk}) \\ &= \$5.0\text{E}+04/\text{plant}\end{aligned}$$

$$\begin{aligned}\text{Hardware} &= \underline{\$5.0\text{E}+4/\text{plant}} \\ \text{Total} &= \$1.0\text{E}+05/\text{plant}\end{aligned}$$

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

$$\begin{aligned}\text{NI} &= (21 \text{ plants})(\$3.6\text{E}+04/\text{plant}) + (21 \text{ plants})(\$1.0\text{E}+5/\text{plant}) \\ &= \$2.9\text{E}+06\end{aligned}$$

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

As discussed in Step 9 of Table 2, utility labor for SIR operation and maintenance was estimated at 210 man-hrs/py for Group 1 plants and 110 man-hrs/py for Group 2 plants.

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

Group 1:

$$\begin{aligned}\text{I}_o &= (210 \text{ man-hrs/py})(\$2270/\text{man-wk})/(40 \text{ man-hrs/man-wk}) \\ &= \$1.2\text{E}+04/\text{py}\end{aligned}$$

Group 2:

$$\begin{aligned}\text{I}_o &= (110 \text{ man-hrs/py})(\$2270/\text{man-wk})/(40 \text{ man-hrs/man-wk}) \\ &= \$6.2\text{E}+03/\text{py}\end{aligned}$$

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

$$\begin{aligned}\text{NTI}_o &= (21 \text{ plants})(\$1.2\text{E}+04/\text{py})(27.2 \text{ yr}) \\ &\quad + (21 \text{ plants})(\$6.2\text{E}+03/\text{py})(27.2 \text{ yr}) \\ &= \$1.0\text{E}+07\end{aligned}$$

12. Total Industry Cost (S_T):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.3E+07	\$1.8E+07	\$7.8E+06

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development

NRC resources for SIR development consist of preparation of a Generic Letter to the affected plants that informs the licensees of the potential problems and requires an inspection/review of the protection system testing capabilities as well as the technical analyses and/or design modifications needed to implement the resolution. In accordance with NUREG/CR-4627 (SEA 1986), the generic letter for a complex technical specification change requires 4 man-wks of NRC technical staff labor and 2 man-wks for management and legal reviews and concurrences for a total of 6 man-wks of NRC labor.

14. Total NRC Cost for SIR Development (C_0):

$$C_0 = (6 \text{ man-wks})(\$2270/\text{man-wk}) = \$1.4\text{E}+04$$

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

Group 1:

Review and approval of license's evaluations, including technical, management, and legal staff. Applies to technical specification changes = 12 man-wks

Group 2:

Review and approval of license's evaluations, including technical, management, and legal staff. Applies to review of proposed design modifications and technical specification changes = 10 man-wks

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

Group 1:

$$C = (12 \text{ man-wks})(\$2270/\text{man-wk}) = \$2.7\text{E}+04/\text{plant}$$

Group 2:

$$C = (10 \text{ man-wks})(\$2270/\text{man-wk}) = \$2.3\text{E}+4/\text{plant}$$

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

$$\begin{aligned} NC &= (21 \text{ plants})(\$2.7\text{E}+04/\text{plant}) + (21 \text{ plants})(\$2.3\text{E}+04/\text{plant}) \\ &= \$1.1\text{E}+06 \end{aligned}$$

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

It is assumed that additional NRC labor will be required at all affected plants to review reports, inspect the plant modifications (if any), and for record-keeping. The estimated NRC resources for these activities are about 2 man-wks/py.

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

$$C_o = (2.0 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$4.6\text{E}+03/\text{py}$$

20. Total NRC Cost for Review of SIR Operation and Maintenance (NTC_o):

$$\text{NTC}_o = (42 \text{ plants})(26.9 \text{ yr})(\$4.6\text{E}+03/\text{py}) = \$5.2\text{E}+06$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$6.3E+06	\$9.0E+06	\$3.6E+06

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DEFICIENCIES IN THE REGULATIONS SUGGESTED BY THE DAVIS-BESSE INCIDENT

ISSUE 123

1.0 INTRODUCTION AND CONCLUSIONS

The objective of this analysis is to determine the priority or further disposition of Generic Issue 123, "Deficiencies in the Regulations Governing Design - Basis Accident and Single Failure Criterion Suggested by the Davis-Besse Incident of June 9, 1985." This analysis was prepared pursuant to a request for prioritization and preliminary prioritization analysis that were submitted by F. Rowsome (1985) of the U.S. Nuclear Regulatory Commission (NRC).

It was postulated by Rowsome (1985) that four categories of issues are not analyzed in the licensing process. These are:

1. Root causes of Design Basis Accidents (DBAs) are not analyzed and they may initiate or exacerbate a plant transient involving an initiating event;
2. Allowable outage times (AOTs) and limiting conditions for operation (LCOs) in plant technical specifications may be inadequate since they are derived from potentially flawed DBA analyses;
3. High-probability common cause failures are not adequately addressed in licensing requirements; and
4. Single human errors that may have a broader effect than active failures are not covered.

Based on a review of the analyses performed in support of several Unresolved Safety Issues and Generic Safety Issues that were completed subsequent to the Rowsome (1985) analysis, and the resolutions of these issues, it was concluded that each of these four subissues under the GSI 123 umbrella have been resolved.

2.0 BACKGROUND

The proposed safety issues deals with the potential inadequacies of nuclear power plant design criteria and safety analyses. Rowsome (1985) postulated that the Design Basis Accident (DBA) analyses that are performed in support of plant licensing may overlook some high frequency core damage scenarios. Licensees are required during licensing to evaluate the plant's response to the DBAs specified in Chapter 15 of NUREG-0800 (NRC 1981) to ensure that the radiation dose criteria in 10 CFR 100 are not exceeded. More specifically, the plant is required to tolerate (1) the initiating event, (2) the concomitant effects of the initiating event, (3) the loss of all non-safety-grade equipment, and (4) any single active failure among the Engineered Safety Features (ESFs) challenged or potentially challenged by the initiating event. However, the root cause of the DBA need not be determined so the possibility exists that the root cause of the DBA may also defeat one or more divisions of ESFs that are designed to respond to the DBA. Based on this argument and supported by descriptions of several similar incidents (including the Davis-Besse incident), Rowsome (1985) stated that plants may be vulnerable to core damage scenarios involving as little as two active failures,

one active failure and one human error, two human errors, or a single common cause failure. Rowsome (1985) further stated that since limiting conditions for operation (LCOs) and allowable outage times (AOTs), which are derived from DBA analyses, may allow plants to operate at power in a condition in which they are only one active failure away from a core damage scenario.

The specific vulnerabilities of concern here are:

1. Support system failures that can precipitate initiating events as well as defeat one or more of the ESFs needed to respond to the initiator. Note that IEEE (1980) addresses selected double failures related to ESF actuation systems.
2. Single human errors that may have more broad effects than any active failures.
3. Active failures or human errors of multiple common cause that have more broad effects than any active failure.

Based on these potential vulnerabilities, it is possible that there exist some high probability paths to core melt (and possibly early containment failures) that remain undiscovered and unreviewed in operating plants. Rowsome (1985) goes on to state that there are likely to be vulnerabilities to the concomitant effects of auxiliary system failures that have not been identified.

3.0 PROPOSED RESOLUTION

The Rowsome (1985) analysis split the aforementioned concerns into two subissues. Issue 123.A deals with two-failure paths leading from normal operation to core-damage and issue 123.B deals with high-probability multiple failures and common cause failures. The proposed resolutions to these subissues are discussed in the following paragraphs.

The proposed resolution to subissue 123.A "Double Failure Vulnerabilities," was to amend the regulations to require that the component failures or human errors that may precipitate a DBA and also defeat one or more divisions of an ESF be identified and analyzed in the licensing process. This would require all plants to re-evaluate their designs and implement modifications, where necessary, including modifications to AOTs and LCOs. Dependency analyses and plant logic models could be developed to identify the potential vulnerabilities related to this subissue.

The proposed resolution to subissue 123.B, "High-Probability Common Cause Vulnerabilities Outside the Reach of the Regulations," deals with regulatory changes to require plants to evaluate certain classes of multiple common cause failures in their DBA analyses. A scoring scheme was proposed as a means of identifying the appropriate vulnerabilities to be examined as DBAs. Such a scheme would require a substantial revision to nuclear power plant licensing requirements. The goal would be to ensure that all high probability common cause failures are identified, appropriately evaluated, and corrected (if necessary) in plant licensing stages.

4.0 DISCUSSION

Based on the previous discussion and the preliminary analysis prepared by Rowsome (1985), there are four specific categories of plant safety issues contained within the Generic Issue 123 umbrella. It was postulated by Rowsome (1985) that each of these four categories of issues are not analyzed in the licensing process. These are: (1) root causes of DBAs are not analyzed and they may initiate or exacerbate a plant transient involving an initiating event; (2) AOTs and LCOs may be inadequate since they are derived from potentially flawed DBA analyses; (3) high-probability common cause failures are not adequately addressed in licensing requirement; and (4) single human errors that may have a broader effect than active failures are not covered.

The Rowsome analysis was prepared in 1985 and thus did not have the benefit of the results of the analyses of several Unresolved Safety Issues (USIs) and Generic Safety Issues (GSIs). At that time, the NRC was in the early stages of resolving USIs A-45 (Decay Heat Removal), A-44 (Station Blackout), A-17 (Systems Interactions), and A-47 (Interactions of Control Systems). In addition, GSI 117, which involved AOTs and LCOs, has been examined. Consequently, a significant amount of work has been performed since 1985 related to the listed USIs and GSIs. Furthermore, the NRC recently issued a requirement to all licensees to prepare a plant-specific PRA (i.e., Individual Plant Examination - IPE). As a result, it appears that most, if not all, of GSI 123, including the proposed resolution, has been addressed in these other activities. Discussions of each of the four categories of issues listed above and the aspects addressed by the other activities are provided in the following sections. A summary of the following discussion is presented in Table 1.

I. Root Causes of DBAs are not Analyzed

This subissue is concerned with the possibility that auxiliary or support system failures may cause a plant transient or initiating event as well as result in failure of one or more safety systems to perform their intended function. One example of this type of event involves failure of the service water system (SSW) which supplies cooling water directly or indirectly to reactor coolant pump (RCP) seals as well as to a number of ESF pumps (e.g., high-head safety injection and containment fan cooler pumps). Failure of the SSW system may result in an RCP seal LOCA and failure of the systems intended to maintain primary coolant inventory and remove decay heat after the RCP seal LOCA occurs. Other examples include failures of the component cooling water (CCW) system, which may result in similar accident sequences as SSW failures, and station blackout.

It appears that this concern has been addressed in a number of recently resolved USIs. USI A-17, "Systems Interactions in Nuclear Power Plants," (NRC 1989a and 1989b) specifically addressed events involving transients and loss of at least one redundant portion of any one of the systems required to respond to the transient. Similarly, A-17 addressed initiating events and similar failures of redundant safety systems. A-17 also examined degradation of safety-systems by non-safety-systems as well as degradation of auxiliary support systems such as SSW, CCW, and AC/DC power. Regarding the latter, the root causes of DBAs would most likely be traced to failures or malfunctions of auxiliary support systems, such as loss or degradation of feedwater control power, RCP seal cooling, or turbine-generator control systems. This demonstrates that A-17 addressed the

vulnerabilities to events involving transients or initiating events and degradation of the safety systems that are designed to respond to the events.

The results of the A-17 studies indicated that the causes and effects of systems interactions were plant-specific in nature due to the differences in plant designs. In addition, it was demonstrated that plant-specific probabilistic risk assessments (PRAs) were effective tools for identifying vulnerabilities to systems interactions. At this time, licensees are being required to perform systematic Individual Plant Examinations (IPEs, NRC 1988a), which could be a PRA-type analysis. One of the subjects to be evaluated in the IPEs is common cause (dependent) failures, of which systems interactions are a subset. The information and insights gained from the A-17 studies has been provided to the utilities to assist in the identification and evaluation of system interactions and other common cause failures. The utilities are expected to propose plant-specific procedure and/or hardware modifications, where appropriate, to reduce their vulnerabilities to such events. Consequently, it appears that the vulnerabilities to the root causes of DBAs are being systematically identified and corrected, if necessary, on a plant-specific basis in the IPE process.

Two additional USIs were identified as having some overlap on this subissue of GSI 123. These are USI A-44, "Station Blackout," and USI A-45, "Shutdown Decay Heat Removal Requirements." USI A-44 (NRC 1988b) addresses the likelihood and duration of losses of offsite power and the redundancy and reliability of onsite emergency AC power sources (e.g., diesel generators), and the effects on plant risk of failures of all AC power sources. Support system failures were important aspects of these analyses, particularly DC instrumentation and control (I&C) power supplies, instrument air supplies, and auxiliary cooling systems such as SSW and CCW. Two key aspects of this issue are losses of DC power which can "blind" operators to actual plant conditions and losses of auxiliary cooling systems that, if not recovered, could cause an RCP seal LOCA as described previously. Again, this issue focused on losses of support systems that can both cause an initiating event or transient as well as defeat the systems that are designed to respond to the event. Resolution of this issue involved, among other things, improving the reliability of offsite AC power systems and strengthening each plant's capability to cope with an extended loss of AC power.

USI A-45 addressed potential improvements to the reliability of shutdown decay heat removal systems which are required to operate after a transient or initiating event. Among the items addressed by this USI were support system failures and single-point vulnerabilities. Although the major hardware modifications proposed as potential resolutions to this issue were not implemented because of an unsatisfactory cost-benefit ratio, part of the proposed resolution involved performance of plant-specific PRAs which has been implemented in the IPE process. It was concluded (NRC 1988c) that resolution of this issue could only be achieved on a plant-specific basis. Therefore, the utilities were directed to identify decay heat removal vulnerabilities and determine if cost-effective solutions to these vulnerabilities could be achieved.

The effectiveness of the resolutions to these USIs are illustrated in the results of the NUREG-1150 PRAs (NRC 1989c). Support system failures, including SSW and CCW failures, are shown to be important contributors to plant core damage frequencies both as initiating events and as contributors to core damage sequences initiated by other causes. Station blackout is also an important initiating event at some plants. This demonstrates the high visibility that has

been given to support systems in the most recent PRAs, much of which has derived from the visibility of these important USIs.

II. AOTs and LCOs May Be Inadequate

This subissue relates to the fact that AOTs and LCOs may be derived from the results of DBA analyses and, if the DBA analyses are inadequate, as Rowsome (1985) indicates, then the AOTs and LCOs may also be inadequate. This issue addressed potential safety concerns related to allowable outages of diverse components at nuclear power plants. Since it is not uncommon for a power plant to have several components out of service at the same time, the potential exists for operation of a plant in a dangerous configuration in which two or more components that appear in the same accident sequence are out of service. This issue focuses on outages for diverse components that are not necessarily in the same safety system, such as simultaneous outage of valves in the high pressure injection system (HPIS) and low-pressure injection system (LPIS). In general, AOTs and LCOs address simultaneous outages of redundant components and may be insufficient to address many of these potentially dangerous plant configurations.

The issue of AOTs and LCOs was addressed in GSI 117, "Allowable Outage Times for Diverse Simultaneous Equipment Outages," (NRC 1983). The proposed safety issue resolution involved development of a methodology for control room operators to use to track and evaluate prevailing plant configurations. The methodology would include plant-specific accident sequences. Operators would manually enter the prevailing plant configuration, such as valve alignments, switch settings, and equipment outages, and the methodology would indicate whether or not the plant is vulnerable to high-probability core-melt sequences. Above a certain critical core-melt frequency, the plant would be required to shutdown because the current configuration represents one in which diverse safety systems may not be available to respond to one or more initiating events. A computer-base PRA software system was proposed as one possible mechanism for accomplishing this resolution.

This subissue of GSI 123 deals with allowable outage times for components and the possibility that a plant may be operating one active component failure away from core damage. A large fraction of the potential core damage frequency and public risk reduction associated with this subissue would be associated with removing vulnerabilities associated with the component outages. The approach to evaluating the change in core damage frequency for GSI 117 was to remove the test/maintenance unavailability from basic events in each cut set that contained multiple test/maintenance outage terms. This would preclude the possibility that a plant could be operating at full power with vital equipment in different ESF systems down for maintenance. This would effectively remove the vulnerabilities associated with AOTs and LCOs on components in different ESF systems and in redundant divisions of each ESF system. The cost-benefit analysis preformed for GSI 117 showed the resolution was not cost-effective and the issue was dropped from further consideration (NRC 1983).

III. High-Probability Common Cause Failures Not Adequately Addressed in Licensing Requirements

USI A-17 addressed, among other things, the potential for common cause events involving systems/components that share physical connections, share spatial configurations, or could cause operator errors that may result from operation disinformation or inhibition of an operator's ability to respond to a malfunction (NRC 1989a). An example that was addressed in the USI A-17 analyses was high-energy line breaks and the possibility that adverse environmental conditions resulting from such an event could induce failures in one or more safety systems designed to respond to the event. This is an example of the spatially-coupled system interaction. Other examples include seismic events, fires, and floods that could affect the operability of equipment/systems located in close proximity to each other, interactions between normal offsite and emergency onsite AC power systems (e.g., sharing common breakers or power distribution buses), and common support systems, cross-connects, and other functional dependencies.

As discussed previously, the NRC concluded that plant-specific analyses were necessary to accurately identify, evaluate, and resolve (where appropriate) vulnerabilities to systems interactions. The plant-specific IPE program initiated by the NRC (1988a) includes an assessment of common cause or dependent failures. Since systems interactions are a subset common cause failures, the resolution of this subissue of GSI 123 is subsumed in the performance of the IPEs.

USI A-47, "Safety Implications of Control Systems," also addressed aspects of this subissue of GSI 123, including single failures or multiple failures which could cause a malfunction in one or more control systems. Such malfunctions may result in an undesirable control system response or provide misleading information to an operator. The analyses in support of A-47 (see NRC 1989d) identified potential control system failures that could cause overpressure, overcooling, overheating, overfilling, or reactivity events. All these events are covered in DBA analyses. Requirements were established that, in general, provide or enhance systems to protect against reactor vessel/steam generator overfill events and to prevent steam generator dryout, enhance procedures and provisions to verify the operability of these systems, and modify selected procedures to respond to small-break LOCAs. This issue is considered to be resolved by the NRC (NRC 1983).

IV. Single Human Errors

This subissue relates to the possibility that a single human error could potentially result in a plant transient or initiating event and defeat one or more divisions of a safety system. No events of this type have been identified in plant operating experience although the Davis-Besse incident is one that involved two human errors and a flawed Steam and Feedwater Rupture Control System (Rowsome 1985). Therefore, it appears unlikely that significant vulnerabilities to single human errors exist in the industry.

Nevertheless, USIs A-17, A-44, A-45, and A-47 addressed various aspects of this subissue, including degradation of operator information which may lead to operator "blindness" and cause incorrect operator actions, and human errors as contributors to system failures. The analyses performed in support of these

issues considered, for the most part, the possibility that single operator actions could defeat one or more divisions of an ESF system. In addition, in situations where operator actions are necessary but the integrity of the information in the control room may be questionable (such as following a station blackout), it was assumed that the operator would not respond correctly. This effectively addresses single human errors that may defeat an ESF that otherwise would be operable.

Single human errors may also initiate a plant transient. Instances can be found in Licensee Event Reports, such as maintenance errors during electrical switchgear work that result in main feedwater isolation or interruption of vital AC power sources, in which single human errors have resulted in plant shutdowns. Maintenance errors on the non-nuclear side of the plant that resulted in turbine-generator trips have also occurred. However, such failures have not to date resulted in the occurrence of a transient and simultaneous failure of ESF systems that are designed to respond to the transient. This is primarily because of the redundancy and diversity of plant systems, particularly ESF systems, that are designed to minimize the effects of single failures by maintaining separation of different divisions of vital plant equipment. This conclusion was supported by the results of a number of recent PRAs, including NSAC-60 (Sugnet et al. 1984) and the PRAs prepared in support of NUREG-1150 (NRC 1989c), in which no accident sequences initiated by single human errors were found to contribute significantly to core damage frequencies.

TABLE 1: Assessment of Issue 123 Vulnerabilities Relative to Other Generic Safety Issues

<u>Issue 123 Vulnerabilities</u>	<u>Related Safety Issues</u>	<u>Aspect Addressed/Resolution</u>
I Root causes of DBAs not analyzed; may initiate and exacerbate a plant transient or limiting event	A-17, "Systems Interactions in Nuclear Power Plants"	<p>Items to be addressed in the individual plant examinations (IPEs) include:</p> <ol style="list-style-type: none"> 1. Transient <u>AND</u> (A) loss of at least one redundant portion of any one of the systems required to respond <u>or</u> (B) degradation of operator information. 2. Initiating event <u>AND</u> (A) <u>or</u> (B) above. 3. Degradation of a safety-system by non-safety system. 4. Degradation of redundant safety systems including auxiliary support functions such as service water, component cooling water, and AC/DC power.
	A-44, "Station Blackout"	<ol style="list-style-type: none"> 1. Enhance reliability of onsite and offsite AC power systems (including normal and emergency power). 2. Enhance abilities of plants to withstand station blackouts of specified durations.

TABLE 1: Assessment of Issue 123 Vulnerabilities Relative to Other Generic Safety Issues (Cont.)

<u>Issue 123 Vulnerabilities</u>	<u>Related Safety Issues</u>	<u>Aspect Addressed/Resolution</u>
	A-45, "Shutdown Decay Heat Removal Requirements"	<ol style="list-style-type: none"> 1. Addressed support system failures as contributions to core melt accidents. 2. Addressed single-point vulnerabilities (fire, flood, sabotage, etc.). 3. No new requirements were established on cost/benefit basis. Part of proposed resolution has been implemented (plant-specific PRA).
II AOTs and LCOs may be inadequate since they are derived from potentially flawed DBA analyses.	117, Allowable outage times for diverse simultaneous equipment outages	Random failures in one division of an active ESF when a redundant division is inoperable due to test/maintenance/repair outages. Priority: DROPPED
III High-probability common cause failures not adequately addressed in licensing requirements.	A-47, "Safety Implications of Control Systems"	Addresses single failures or multiple failures cause a malfunction in one or more control systems resulting in an undesirable control system response or provide misleading information to an operator. New requirements were established.
IV Single human errors that may have broader effect than active failures not covered.		<ol style="list-style-type: none"> 1. See A-17; addressed degradation of operator information.

TABLE 1: Assessment of Issue 123 Vulnerabilities Relative to Other Generic Safety Issues (Cont.)

<u>Issue 123 Vulnerabilities</u>	<u>Related Safety Issues</u>	<u>Aspect Addressed/Resolution</u>
		2. See A-44; possibility of loss of operator information on plant conditions addressed.
		3. See A-47; addressed control system failures which could result in misleading information to operators.
		4. See A-45; addressed human error as contributors to system failures.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 129, Residual Heat Removal System Valve Misalignment During Shutdown Cooling Operation

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: In existing BWRs with jet pumps, the possibility exists for the operator to inadvertently open a valve(s) resulting in a decrease in the water level of the reactor pressure vessel (RPV) due to drainage to the suppression pool (SP) via the RHR system. The implementation of automatic isolation systems and interlocks into the RHR system during shutdown is the proposed safety resolution.

AFFECTED PLANTS: BWR: Operating = 26; Planned = 11

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 1.9E+03

OCCUPATIONAL DOSES:

SIR Implementation =	2.93E+01
SIR Operation/Maintenance =	0
Total of Above =	2.93E+01
Accident Avoidance =	6.01E+02

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	1.1
SIR Operation/Maintenance =	0
Total of Above =	1.1
Accident Avoidance =	49.9

NRC COSTS:

SIR Development =	0.018
SIR Implementation Support =	0.27
SIR Operation/Maintenance Review =	0
Total of Above =	0.27

RHR SYSTEM VALVE MISALIGNMENT DURING
SHUTDOWN COOLING OPERATION ON BWRs

ISSUE 129

1.0 SAFETY ISSUE DESCRIPTION

It has been the common perception that once a plant is in the shutdown cooling mode of operation that the safety of the plant is assumed. This is demonstrated by the fact that PRAs do not model plants in the shutdown cooling operation mode. It has been found, however, that in existing BWRs with jet pumps, the possibility exists for the operator to inadvertently open a valve(s) resulting in a decrease in the water level of the reactor pressure vessel (RPV) due to drainage to the suppression pool (SP) via the RHR system. There are multiple pathways that water can reach the SP. Presently, all of these plants have automatic shutdown cooling (SDC) suction isolation on the two valves that are in series from the RPV to the suction of the RHR pumps. These valves will close on either high RHR suction line flow or low RPV level. Twenty-five percent of the affected BWRs, however, have SDC isolation disabled whenever their plants are in the SDC mode of operation. The closing of these valves will terminate all possible draining paths to the SP from the RPV. In addition to the SDC isolation system, one half of the affected plants also have interlocks that can prevent a possible drainage of the RPV via a 20-inch shutdown cooling suction path.

PROPOSED SAFETY ISSUE RESOLUTION

The proposed safety issue resolution is to do the following (SIR): (1) have the 25% of the plants that do not have Tech Specs for auto isolation prepare and process Tech Specs changes to require them to have auto isolation operable during SDC operation; (2) have the 50% of the plants that do not have interlocks purchase and install interlocks on their SDC suction valves; and (3) have the 75% of the plants that do not Tech Specs for their interlocks (this is 75% of the affected BWR population since 1/2 of the plants did not have interlocks and 1/2 of the plants that did have interlocks did not have Tech Specs governing their use) make Tech Spec changes to require their interlocks to be operable during SDC operation.

The NRC staff has become increasingly aware of the number of events which have resulted in inadvertent draining of the RPV during plant shutdown conditions. Therefore, a four-part proposed resolution has been recommended. The proposed resolution is briefly described below:

- A. **Design Bases:** This action addresses the review and documentation of the design basis of an inadvertent draining of the RPV during shutdown.
- B. **Interim Operation Procedures:** This action addresses temporary interim modifications and operating procedures to reduce the potential for inadvertent draining of the RPV during the RHR mode of cooling.
- C. **System Design Modifications:** This action addresses permanent modification to reduce the potential for inadvertent draining of the RPV during shutdown cooling.

D. **Technical Specification Changes:** This action addresses changes to Technical Specifications which may be required as a result of items 2 and 3 above.

More detailed discussions of the above resolutions are given in Attachment 1.

AFFECTED PLANTS

This safety issue is of concern for all of the operating and planned BWRs are listed in Attachment 1.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose are estimated in this section. The results are summarized in Tables 1 and 4, respectively.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Residual Heat Removal System Valve Misalignment During Shutdown Cooling Operation (129).

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

		<u>N</u>	<u>$\bar{T}$(yrs)</u>
BWRs:	Operating	26	26.3
	Planned	11	30

3. Plants Selected for Analysis:

None

4. Parameters Affected by SIR:

Possible drainage of the reactor pressure vessel (RPV) due to inadvertent opening of valve(s) to the suppression pool (SP) by the operator has not been addressed in any previous BWR PRAs. A new sequence is developed to analyzed the impacts of an inadvertent opening of valve(s) to the SP from the RPV during shutdown cooling.

5. Base-Case Values for Affected Parameters:

See Step 6.

6. Affected Accident Sequences and Base-Case Frequencies:

Weighted average frequency for the core-melt (referred to here as the SDC sequence) due to an inadvertent drainage of the RPV during shutdown cooling (SDC) is estimated to be $3.0E-5$ /py. This value is derived in Attachment 1.

<u>Sequence</u>	<u>Base-Case Frequency</u>
SDC	$3.0E-5$ /py

7. Affected Release Categories and Base-Case Frequencies:

BWR-4 = $3.0E-5$ /py

The BWR-4 release category value of $6.1E+5$ (man-rem) was reduced by a factor of 10 for the affects of the reactor being in the SDC mode of operation.

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

$$\bar{F}_{BWR} = 3.0E-5/\text{py}$$

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

$$W_{BWR} = 1.83 \text{ man-rem/py}$$

10. Adjusted-Case Values for Affected Parameters:

See Attachment 1

11. Affected Accident Sequences and Adjusted-Case Frequencies:

<u>Sequence</u>	<u>Adjusted-Case Frequency (1/py)</u>
SDC	1.8E-7/py

12. Affected Release Categories and Adjusted-Case Frequencies:

$$BWR-4 = 1.8E-7/\text{py}$$

The BWR-4 release category value of 6.1E+5 (man-rem) was reduced by a factor of 10 for the affects of the reactor being in the SDC mode of operation.

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

$$\bar{F}^*_{BWR} = 1.8E-7/\text{py}$$

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

$$W^*_{BWR} = 1.1E-2 \text{ man-rem/py}$$

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

$$\begin{aligned}\Delta\bar{F}_{BWR} &= (3.0E-5/\text{py}) - (1.8E-7/\text{py}) \\ &= 2.98E-5/\text{py}\end{aligned}$$

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

$$\begin{aligned}\Delta W_{BWR} &= (1.83 \text{ man-rem/py}) - (1.1E-2 \text{ man-rem/py}) \\ &= 1.82 \text{ man-rem/py}\end{aligned}$$

17. Total Public Risk Reduction, (ΔW)_{Total}:

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
1.85E+03	5.57E+04	0

ATTACHMENT 1 (To Table 1)

To quantify the risks associated with an inadvertent draining of the RPV, the probability of a core melt resulting from the loss of water inventory in the RPV is determined. An event tree is shown in Figure 1 that gives the sequence of events postulated to occur from the time water inventory is lost from the RPV during the SDC mode of operation to a possible core-melt.

Initiating Event	Interlock	Auto-ISO	Operator	Inventory	Core Melt	Release Category
Operator Inadvertently Opens Valves to S.P. During SDC Mode of Operation	Interlocks Prevent Op.to Open Valve(s)	Auto Isolation on Low RPV Level	Operator Assesses Problem and Closes Valve(s)	Water Inventory Makeup to RPV

FIGURE 1: Postulated Event Tree for Accident Progression to Core-Melt Due to an Inadvertent Draining of the Reactor Pressure Vessel During the Shutdown Cooling Mode of Operation

INITIATING EVENT

The initiating event is an inadvertent reduction in the primary coolant inventory through the RHR system due to operator error while in the RHR shutdown cooling (SDC) mode of operation.

There have been twenty two events in which a reduction in the primary coolant inventory in the reactor pressure vessel has occurred. A list of the affected BWR plants is given in Table 2. These 22 events have occurred during 226 reactor years of operating experience of the affected Boiling Water Reactors (BWRs). Since these events only occur during the SDC mode of operation an

estimate is made of how many times the reactors are in the SDC mode. It is assumed that an average BWR has five transients per year that require the plants to go to the SDC mode. It is also assumed that each BWR has a shutdown each year for refueling. This gives a total of six times per year that each plant is in the SDC mode of operation. These inadvertent draining events are assumed to occur at or shortly following the onset of SDC; therefore, an equivalent demand failure rate of 22 events per 1356 SDC mode events = $1.62\text{E}-2/\text{yr}$ is estimated.

One half of the events are assumed to result in fast draining of the RPV if SDC is not auto isolated. The other half of the events result in a slow draining of the RPV. This information was obtained by reviewing the list of loss of inventory events involving the RHR system given in the NRC memorandum for GSI #129 [1]. Some interpretation using engineering judgement of the events was required. The frequency of either fast or slow draining events is one half of the above value or $8.1\text{E}-3/\text{year}$.

INTERLOCK SYSTEM

There are a number of potential inventory loss paths on a BWR in the SDC mode of operation, many of which can be initiated by a single mispositioned valve. A fast drain flow path exists when the suppression pool suction valve and the SDC suction valve are opened simultaneously. Six events have been documented when the reactor vessel was drained to the suppression pool via this 20-inch SDC suction line path (This information was provided by the NRC and verified by looking at the list of events). RHR pump suction interlocks are assumed to be provided for one-half of the plants. Of the plants that have these RHR suction interlocks, it is assumed that one-half do not have Tech Specs requiring interlock operability during SDC mode of operation. In addition, there is no requirement to test these interlocks. The average probability that the interlocks fail to prevent fast draining of the RPV to the suppression pool is determined by the percentage of plants that do not have interlocks plus the percentage that do have interlocks times the fraction that do not have Tech Specs requiring use of the interlocks during SDC. The failure probability is therefore estimated as $1/2 + (1/2)(1/2) = 0.75$.

AUTO ISOLATION

Automatic isolation is designed to shut the shutdown cooling suction valves in the event of excessive flow in the shutdown cooling suction line or low water level in the reactor vessel. This will terminate both fast and slow draining events into the suppression pool from the RPV. Of the draining events that have occurred, nine of them were terminated by automatic closure of the SDC suction valves. In three incidents the automatic isolation function did not actuate when required. The most likely cause of these failures is that the isolation logic was not required to be operable during SDC; therefore, the assumption is made that 25% of all the plants have SDC isolation disabled during the SDC mode of operation (3 out of 12 possible closure events). The failure probability of this function is, therefore, 0.25 for both fast and slow draining events.

OPERATOR ACTION

The operator can terminate this transient by closing the suppression pool suction valves in the case of fast draining event or closing either of the SDC suction valves for a fast or slow draining event. The operator could also close

the appropriate valves terminating a slow event (e.g., if due to a heat exchanger leak, etc). The probability of failure for the operator to terminate the transient was estimated by noticing that of the 22 draining events that occurred, nine of them were terminated by automatic isolation. Therefore, there are thirteen draining events that occurred in which the core was not uncovered but automatic isolation did not occur (therefore assuming the operators eventually took appropriate actions). For the case of zero failures in thirteen draining events a Chi Square distribution is assumed and is calculated by the formula $1.386/2D$ where D is number of events in which zero failures occurred. This yields an probability of operator failure of 0.053. This is the median value with 50% confidence bounds.

Another estimate of the failure probability for this operator action can be made using the values in a letter report [2] by Alan Swain at Sandia National Laboratories which has been used as the basis for human error estimates in the current NRC NUREG-1150 severe accident work. In this report Swain gives a value for the operator to diagnose the situation within a time T. He also provides a value for the operator to make a critical action as part of a dynamic task done under moderately high stress which is considered to best characterize our sequence of interest. The total failure probability is the sum of these two events. The amount of time the operator has to diagnose the draining event is estimated to be at least 30 minutes. (For a typical large LOCA in which a reactor is under high pressure and at high power level the ECCS must become operable within 15 minutes). The failure probability to diagnose the event within 30 minutes is 0.001. (It is 0.01 within 20 minutes). The failure probability estimated for the operator to perform the dynamic task once diagnosed is approximately 0.05. Therefore, the probability of failure for this function is estimated to be approximately 0.05 using the value in Swains letter. This is very close to the Chi Square approximation of 0.053. A value of 0.05 is therefore used in this analysis.

WATER INVENTORY MAKEUP

At the present time, there are no Technical Specification requirements requiring the availability of any of the safety (as well as non-safety) systems during the SDC mode of operation. It should be noted, however, that in determining the number of SDC events per year it was estimated that there were 5 transients and 1 plant shutdown for refueling. For the shutdown cases of the 5 transients, the safety systems will be available. The low pressure core spray (LPCS) system should come on automatically at Level 1. The failure probability of approximately $1E-3$ is used for the LPCS system. In addition, recovery is possible by using the condenser and other non-turbine driven high pressure systems for fast or slow draining events as well as the control rod drive (CRD) system for slow draining events.

The safety sytems may not be available for the one shutdown case per year. Even if they were available they would have to actuated by operator action. Therefore, requiring any safety or non-safety system be available no credit is taken for water inventory makeup while the plant is shutdown for refueling. The failure probability for this function is therefore 1 out of 6 or 0.17.

BASE-CASE RELEASE CATEGORY

The release category most appropriate to this core melt sequence is BWR 4. The BWR 4 category release in NUREG/CR-2800 [5] assumes that the reactor is at full power (with its corresponding temperature and pressure) when the accident occurs. In the case of this issue the reactor will be at a much lower decay heat level resulting in a much slower core melt process. All the short term fission products will have decayed away. In addition, it will take longer to pressurize the containment and there will be more time for evacuation.

It is assumed that the BWR 4 release category value of $6.1\text{E}+5$ (man-rem) represents an upper bound. Based on the above arguments a factor of 10 decrease in the whole body man-rem is estimated. This is still considered to be conservative. Therefore, a release category value of $6.1\text{E}+4$ man-rem is used in this analysis for both fast and slow draining events.

BASE-CASE CORE MELT FREQUENCY

Applying the probabilities developed above to the event tree shown in Figure 1 results in a frequency of $(8.1\text{E}-3/\text{py})(0.75)(0.25)(.05)(0.17)$ or $1.3\text{E}-5/\text{py}$ for core melt caused by inadvertent fast draining of the RPV during the SDC mode of operation. The frequency for core melt caused by inadvertent slow draining of the RPV during SDC is $(8.1\text{E}-3/\text{py})(1.0)(0.25)(.05)(0.17)$ or $1.7\text{E}-4/\text{py}$. Note that the interlocks prevent fast draining events only. The total base-case core melt frequency is $(1.3\text{E}-5/\text{py} + 1.7\text{E}-4/\text{py})$ or $3.0\text{E}-5/\text{py}$.

ADJUSTED BASE-CASE

The corrective actions that are assumed to take place in the adjusted base-case are the following:

- 1) All plants prepare and process Tech Specs to require auto isolation to be operable during the SDC mode of operation
- 2) All plants prepare and process Tech Specs to require interlocks to be operable during the SDC cooling mode of operation.

The new values for the functions given in the event tree shown in Figure 1 are discussed below.

INITIATING EVENT

The initiating event was assume to remain the same as for the base-case. It should be noted, however, with new Tech Specs there may be fewer initiating events due to increased operator training and awareness. The effects of this change was not estimated.

INTERLOCK SYSTEM

It is assumed that the Tech Specs are in place (and plant changes are made) to include interlocks on the RHR valves and have them working during shutdown. The failure probability of the interlock system is determined by the following steps:

- 1) Assume operators follow the new Tech Specs,
- 2) Therefore, the failure is due primarily to failure of the interlock.

The failure of the interlock is obtained by using the NRC's Accident Sequence Evaluation Program (ASEP) value of $\sim 1\text{E-}3$ failure probability for the electrical portion plus failure of the valve limit switch (on the valves) which make the interlock work. The value for the failure probability of the valve limit switch is $2.4\text{E-}2$ per the Indian Point. Review Study [3] (see bottom of page 3.2.15-2). This leads to an estimated interlock function failure of $\sim 2.5\text{E-}2$ ($2.4\text{E-}2 + 1.0\text{E-}3$).

AUTO ISOLATION

There are two motor operated valves (MOV's) that will receive an automatic isolation signal to close. The conservative assumption is made that both valves will receive the same automatic isolation signal. The actuation and control value of $1.3\text{E-}3$ per demand given in the Station Blackout Study [3] is used for the auto isolation signal failure. The mechanical failure of an MOV is $3.8\text{E-}3$ per demand [4]. Since only one of the valves need to close the mechanical failure contribution is negligible. Therefore, the failure probability of the auto isolation function is estimated to be $1.3\text{E-}3$.

OPERATOR ACTION

The failure of this function for the adjusted base-case is defined as the failure of the operator to see that either the interlock or auto isolation has failed and take manual action.

The human error failure probability is estimated to be somewhat higher than in the base case assuming he has less time to respond (i.e., by the time the interlock or auto isolation should work but doesn't the reactor water level is already dropping). It is assumed that the human error failure probability is 0.10 which is double the previous base case human error value.

WATER INVENTORY MAKEUP

This is assumed to be the same as in the base-case. This function is given a failure probability of 0.17.

ADJUSTED BASE-CASE CORE MELT FREQUENCY

Applying the probabilities developed above to the functions of the event tree shown in Figure 1 gives a frequency of $(8.1\text{E-}3/\text{py})(2.5\text{E-}2)(1.3\text{E-}3)(0.1)(0.17)$ or $4.5\text{E-}9/\text{py}$ for core melt by inadvertent fast draining of the RPV during the SDC mode of operation. The value for slow draining is $(8.1\text{E-}3/\text{py})(1.0)(1.3\text{E-}3)(0.1)(0.17)$ or $1.8\text{E-}7/\text{py}$.

The total adjusted base case core melt frequency is $4.5\text{E-}9/\text{py} + 1.8\text{E-}7/\text{py}$ or $1.8\text{E-}7/\text{py}$.

ADJUSTED BASE-CASE RELEASE CATEGORY

This is same as for the base-case.

TABLE 2: Affected BWR Population

	<u>Start Date</u>	<u>Years Operated To Date</u>	<u>Backfit Yrs. (35-yrs to date)</u>	<u>Forward Fit</u>
<u>BWRs</u>				
Dresden 2	8/70	16	19	--
Dresden 3	10/71	15	20	--
Monticello	7/71	15	20	--
Quad Cities 1	8/72	14	21	--
Quad Cities 2	10/72	14	21	--
*Pilgrim 2	--	--	--	30
<u>BWR 4s</u>				
Browns Ferry 1	8/74	12	23	--
Browns Ferry 2	3/75	11	24	--
Browns Ferry 3	3/77	9	26	--
Vermont Yankee	11/72	14	21	--
Peach Bottom 2	7/74	12	23	--
Peach Bottom 3	12/74	12	23	--
Fitzpatrick	6/77	9	26	--
Cooper	7/74	12	23	--
Hatch 1	12/75	11	24	--
Hatch 2	8/79	7	28	--
Brunswick 1	3/77	9	26	--
Brunswick 2	11/75	11	24	--
Duane Arnold	5/74	12	23	--
*Fermi 2	/87	--	--	30
*Hope Creek 1	11/86	--	--	30
*Hope Creek 2	--	--	--	30
Susquehanna 1	8/83	3	32	--
Susquehanna 2	2/85	1	34	--
*Shoreham	Indef	--	--	30
Limerick 1	2/86	0	35	--
*Limerick 2	/91	--	--	30
<u>BWR 5s</u>				
LaSalle 1	10/82	3	32	--
LaSalle 2	6/84	2	33	--
*Nine Mile Pt 2	3/87	--	--	30
WNP 2	12/85	1	34	--
<u>BWR 6s</u>				
Grand Gulf 1	7/85	1	34	--
*Grand Gulf 2	Indef	--	--	30
*Clinton 1	12/86	--	--	30
*Perry 1	10/86	--	--	30
*Perry 2	Indef	--	--	30
River Bend 1	6/86	0	35	--
Total:		226	684	330

*Not in Operation - To Be Completed

SUMMARY

The values for the functions used to quantify the core-melt sequence in Figure 1 for both the base-case and the adjusted base-case are summarized in Table 3.

TABLE 3: Failure Probability Values of Functions Shown in Figure 1 for Base-Case and Adjusted Base-Case

	Initiating Event	Interlock	Auto ISO	Operator	Inventory
Base-Case					
Fast Drain	8.1E-3/yr	0.75	0.25	0.05	0.17
Slow Drain	8.1E-3/yr	N/A	0.25	0.05	0.17
Adjusted Base-Case					
Fast Drain	8.1E-3/yr	2.5E-2	1.3E-3	0.10	0.17
Slow Drain	8.1E-3/yr	N/A	1.3E-3	0.10	0.17

TABLE 4. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Residual Heat Removal System Valve Misalignment During Shutdown Operation (129)

2. Affected Plants (N):

26 Operating BWRs
11 Planned BWRs

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

26.3 years for the operating plants
30.0 years for the planned plants

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

BWR: $(19,900 \text{ man-rem})(2.98\text{E-}5/\text{py}) = 5.93\text{E-}1 \text{ man-rem/py}$

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

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds</u> <u>Upper</u>	<u>(man-rem)</u> <u>Lower</u>
601	3,630	0

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

Dose estimates given directly in next step.

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

It is assumed that the system modification will require 3 man-wk. This estimate is based on the value used to make similar interlock modifications in GSI #99. The dose exposure inside the containment building (shutdown mode) is given as 25 mR/hr in (Andrews, et al., 1983). Therefore, total occupational dose for this task (per plant) is:

$$D = (2.5\text{E-}02 \text{ R/hr})(3 \text{ man-wk/plant})(40 \text{ man-hr/man-wk})(75\% \text{ worker efficiency}) \\ = 2.25 \text{ man-rem/plant}$$

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

$$ND = (13 \text{ plants})(2.25 \text{ man-rem/plant}) \\ = 29.3 \text{ man-rem}$$

9-11. Steps Related to Occupational Dose Increase for SIR Operation and Maintenance

The affected plants and the modified systems are assumed not to require any additional operation/maintenance beyond that normally required for the vessel. Therefore, $D = 0$.

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
29.3	87.9	9.8

3.0 SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. Results are summarized in Table 5.

TABLE 5. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Residual Heat Removal System Valve Misalignment During Shutdown Operation
(129)

2. Affected Plants (N):

26 Operating BWRs
11 Planned BWRs

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

26.3 years for the operating plants
30.0 years for the planned plants

Industry Costs (Steps 4 through 12)

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

BWR: $(\$1.65E+09)(2.98E-5/py) = \$4.92E+04/py$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$4.99E+07	\$3.01E+08	0

6. Per-Plant Industry Resources for SIR Implementation:

Cost is estimated directly in Step 7.

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

The following cost estimates are based on \$2270/man-wk. The time estimates for implementing the interlocks is assumed to be same as in GSI #99. The changes for the automatic isolation valves from the RPV to the RHR pump suction requires only Tech Spec changes.

A. Review and Documentation of the Design Basis of SDC Suction Valve Interlocks

This task is estimated to require 4 man-wk. The cost associated with this is $(4 \text{ man-wk})(\$2270/\text{man-wk}) = \9080 .

B. Interim Operating Procedures

It is estimated that it will take 2 man-wk to develop interim operating procedures and 3 man-wk for operator training and instructions. This gives a total of 5 man-wk or (5 man-wk) (\$2270/man-wk) = \$11350.

C. System Design Modifications

The cost of equipment associated with installing the interlocks is estimated to be \$2000. This estimate includes the cost of cables, wiring, and alarms. It should, however be noted that in any major plant modification the cost of equipment and design changes is usually over-shadowed by the power replacement cost. The power replacement cost is estimated later in this section. It is estimated that the system modification will require 3 man-wk for alarm installation and equipment testing. The labor cost is therefore (3 man-wk) (\$2270/man-wk) or \$6810. The logic system modification is estimated to cost about \$8800. This estimate is based on 2 man-wk of engineering support at \$2270/week, 1 man-week of craft services at \$2270/week, and \$2000 in instrumentation and supplies. The replacement power cost for one week of plant shutdown is estimated to be (7 days) (\$300,000/day) or \$2.1E+6. Due to the fact that the system design modifications can be accomplished during normal plant outage, i.e., refueling time, the replace power cost is excluded from this analysis.

D. Technical Specifications Changes

The changes made to the Technical Specifications is estimated to require 4 man-wk. The cost is therefore (4 man-wk)(\$2270/man-wk) or \$9080.

Therefore, the total cost of implementing this safety issue per plant is as follows:

<u>TASK</u>	<u>COST (\$)</u>
A	\$ 9.08E+03
B	\$ 1.14E+04
C	\$ 1.76E+04
D	\$ 9.08E+03
<u>Total</u>	\$ 4.72E+04/plant

This is the cost per plant to both install interlocks and provide Tech Specs for their operation. This required for 1/2 of the 37 plants or 19 plants. Tech Specs are needed to address 1/2 of the plants that already have interlocks without any present Tech Specs (9 plants). In addition, 1/4 of plants need Tech Specs for their RHR auto isolation to function during SDC mode of operation (9 plants). Therefore, 19 plants require tasks A, B, C and D or \$4.72E+04/plant; 9 plants require task D for interlocks (\$9.08E+03/plant) and; 9 plants require task D for auto isolation (\$9.08E+03/plant).

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

$$NI = (\$4.72E+04/\text{plant})(19 \text{ plants}) + (\$9.08E+03/\text{plant})(9 \text{ plants}) + (\$9.08E+03/\text{plant})(9 \text{ plants}) = \$1.06E+06$$

9-11. Steps Related to Industry Cost for SIR Operation and Maintenance:

The affected plants are assumed not to require any additional operation/maintenance beyond that normally required for the plant.

Therefore $I_0 = 0$.

12. Total Industry Cost (SI):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.06E+06	\$1.59E+06	\$5.30E+05

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Cost is estimated directly in step 14.

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

From GSI #99, it is estimated that total NRC labor requirement for SIR development is 8 man-wk. This translates to (8 man-wk)(\$2270/man-wk) or \$1.82E+04.

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

Cost is estimated directly in step 16.

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

From observing GSI#99, it is estimated that it takes 3 man-wk/plant to inspect the design modifications and 3 man-wk/plant to review the Technical Specifications and their compliance with the regulatory guides. Therefore the NRC cost/plant for support of the SIR implementation is:

$$\begin{aligned} (3 \text{ man-wk/plant})(\$2270/\text{man-wk}) &= \$6,810/\text{plant for design review} \\ (3 \text{ man-wk/plant})(\$2270/\text{man-wk}) &= \$6,810/\text{plant for Tech Specs only} \end{aligned}$$

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

$$\begin{aligned} NC &= (\$6810/\text{plant})(19 \text{ plants}) + (\$6810/\text{plant})(18 \text{ plants}) \\ &= \$2.52E+05 \end{aligned}$$

18-20. Steps Related to NRC Costs for Review of SIR Operation and Maintenance:

No additional operation/maintenance review by the NRC for the affected PWRs is assumed to be required beyond that normally required for the plan. Therefore, $C = 0$.

21. Total NRC Cost (SN):

Best Estimate

\$2.7E+05

Upper Bound

\$3.96E+05

Lower Bound

\$1.44E+05

REFERENCES (For Issue 129)

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Kolb, G. J., et al. 1982. *Review and Evaluation of the Indian Point Probabilistic Safety Study*. NUREG/CR-2934 (SAND82-2929), Sandia National Laboratories, Albuquerque, NM.

Kolaczowski, A. M., et al. 1983. *Station Blackout Accident Analyses (part of NRC Task Action Plan A-44)*. NUREG/CR-3226 (SAND82-2450), Sandia National Laboratories, Albuquerque, NM.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 132, RHR Pumps Inside Containment

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: This issue addresses potential deficiencies at a small number of plants related to the location of Residual Heat Removal (RHR) pumps inside containment. The RHR system is the preferred system for long-term decay heat removal at nuclear power plants. At most plants, the RHR pumps are located outside containment and are not required to be environmentally qualified. However, in some Westinghouse plants, the RHR pumps are located inside containment. The RHR system at these plants cannot be relied upon for long-term cooling under conditions which would result in harsh containment environments, such as loss-of-coolant accident (LOCAs), steam line breaks, and inadvertent containment spray actuation. Two options are proposed for resolution. The first involves establishment of environmental qualification requirements for the RHR pumps inside containment and certification by the affected utilities that the existing RHR pumps are environmentally qualified. The second option is to install cross-connects to a second unit on the same site to provide backup long-term decay heat removal capabilities without using the RHR system.

AFFECTED PLANTS: This issue affects a total of 8 PWRs.

<u>RISK/DOSE RESULTS (person-rem):</u>	<u>SIR Option 1</u>	<u>SIR Option 2</u>
PUBLIC RISK REDUCTION =	1.3E+03	1.5E+03
OCCUPATIONAL DOSES:		
SIR Implementation =	0	1.5E+03
SIR Operation/Maintenance =	0	4.1E+01
Total of Above =	0	1.5E+03
Accident Avoidance =	1.5E-01	1.1E+01

COST RESULTS (\$10⁶):

INDUSTRY COSTS:		
SIR Implementation =	2.0E+00	1.6E+01
SIR Operation/Maintenance =	2.2E-01	1.7E-01
Total of Above =	2.2E+00	1.6E+01
Accident Avoidance =	1.2E-02	9.1E-01
NRC COSTS:		
SIR Development =	1.2E-01	4.0E-01
SIR Implementation =	6.4E-02	1.1E-01
SIR Operation/Maintenance Review =	5.3E-02	8.4E-02
Total of Above =	2.3E-01	5.9E-01

RHR PUMPS INSIDE CONTAINMENT

ISSUE 132

1.0 SAFETY ISSUE DESCRIPTION

The Residual Heat Removal (RHR) system transfers decay heat from the Reactor Coolant System (RCS) to a secondary cooling water system. It's primary functions are to reduce the reactor coolant temperature at a controlled rate to cold shutdown temperatures during the second phase of normal plant cooldown and to maintain this temperature until the plant is started up. In addition, the RHR system is used to transfer refueling water from the refueling cavity and transfer canal to the refueling water storage tank at the end of refueling operations.

The RHR system takes suction from the hot leg of the reactor coolant system. Reactor coolant flows from the reactor coolant hot leg to the RHR pumps, through the tube side of the RHR heat exchanger, and back to the reactor coolant cold leg. The heat is transferred to the component cooling water (CCW) system circulating through the shell side of the RHR heat exchanger. The RHR system is placed in operation approximately 4 hours after reactor shutdown when the RCS temperature and pressure are approximately 350°F and 425 psig, respectively. The RHR system is capable of cooling the RCS from 350°F to 140°F within about 16 hours.

The RHR system is typically isolated from the reactor coolant system by two motor-operated valves (MOVs) in series and a relief valve. At most PWRs, the RHR pumps and heat exchangers are located outside containment and the only RHR components inside containment are the MOVs, relief valve, and piping system components. The RHR pumps are not required to be environmentally qualified at such plants. However, in some Westinghouse PWRs, the entire RHR system is located inside containment.

Without being environmentally qualified, the RHR pumps inside containment cannot be given any credit in licensing analyses for providing long-term decay heat removal under any harsh containment conditions. Under harsh containment conditions, long-term decay heat removal must be accomplished by the Emergency Core Cooling System (ECCS), Auxiliary Feedwater (AFW) system, and steam generators. The RHR system is the preferred system for long-term decay heat removal.

Because the RHR pumps inside containment are not environmentally qualified, they may not perform adequately under harsh environmental conditions. Consequently, the RHR system at these plants cannot be relied upon for long-term cooling under conditions which would result in harsh containment environments, such as the environment predicted to result from loss-of-coolant accidents (LOCAs), feed-and-bleed cooling, main steam line break (MSLB) accidents, and inadvertent containment spray actuation. In addition, licensing requirements only specify that the licensees demonstrate that the plants can be brought safely to hot standby conditions. There is no requirement that the plants must be brought to cold shutdown despite the fact that it would ultimately have to be brought to this condition. If the RHR system is not available to do so, the plants must rely

on steam generator cooling or ECCS cooling to bring the plant to cold shutdown conditions.

The RHR systems that are located inside containment are front-line systems yet are not classified as safety-grade and are not environmentally qualified. At such plants, separate safety-grade low-pressure injection (LPI) and low-pressure recirculation (LPR) systems are provided. In addition to the LPI/LPR system, long-term decay heat removal may be accomplished using AFW and steam generators. Some plants are also provided with cross-connects to a second unit on the same site that is capable of providing long-term decay heat removal functions. These plants may use the second unit's decay heat removal capabilities in lieu of the RHR system to bring the plant to cold shutdown conditions. Use of this cross-connect capability for recovery from LOCA and transient-induced core-melt accident sequences was factored into the Surry PRA (Bertucio and Julius, 1990).

PROPOSED RESOLUTION

The RHR system and alternative means of decay heat removal need to be investigated to determine if there is a need for additional requirements. Two potential resolutions to this issue are investigated. These are described below:

Option 1: Plants with and without cross-connects to a second unit on the same site are to undertake the analytical work necessary to environmentally qualify their RHR systems for post-accident survivability. Six plants will achieve risk reduction based on analysis of pre-recovery accident sequence frequencies given in the Surry PRA (Bertucio and Julius, 1990). The remaining two plants, which are already provided with cross-connects, will achieve risk reduction based on analysis of post-recovery accident sequence frequencies.

Option 2: Plants without cross-connects are to install them to interconnect the HPI and refueling water storage tanks between units on the same site. Six plants will achieve risk reduction based on the difference between the pre-recovery and post-recovery accident sequence frequencies as given in the Surry PRA. The remaining two plants will achieve no risk reduction because cross-connects have already been installed.

SIR option 1 includes NRC-sponsored efforts to determine if RHR pumps should be environmentally qualified and, if so, what requirements should be established and what conditions should the RHR pumps be required to operate under. These investigations should also include an assessment of the survivability and reliability of alternate decay heat removal systems (ECCS, AFW, and steam generators) during design-basis and non-design-basis accidents, including small LOCAs, MSLB, feed-and-bleed events, and containment spray actuation.

If new environmental qualification requirements are established for RHR pumps inside containment, licensees will be required to investigate the ability of the current non-qualified RHR pumps that are inside containment to survive harsh containment environments. For conservatism, it was assumed that the existing RHR pumps would ultimately be qualified, thus negating the need to replace non-qualified pumps with qualified pumps. This would minimize the costs of the proposed resolution and provide conservative cost-benefit results.

SIR option 2 involves design, installation, and maintenance of a piping system to connect the HPI and refueling water storage tanks between units on the same site. This option would also include NRC efforts to determine the need and safety-related requirements for the cross-connects.

AFFECTED PLANTS

This safety issue affects a total of 8 PWR plants, including Surry 1 and 2, North Anna 1 and 2, Beaver Valley 1 and 2, and South Texas 1 and 2² (Sheron 1985). The average remaining lifetimes of these plants were calculated using data provided by Andrews, et al. (1983).

<u>Plant Name</u>	<u>Backfit Yrs</u>	<u>Forward-fit Yrs</u>
Surry 1	25	
Surry 2	26	
North Anna 1	31	
North Anna 2	33	
Beaver Valley 1	30	
Beaver Valley 2		30
South Texas 1		30
South Texas 2		30
	<u> </u>	<u> </u>
TOTAL:	<u>145</u>	<u>90</u>

A total of 235 yrs of reactor operating lifetimes are remaining at these plants. The average remaining lifetime of the affected plants is 29.4 reactor-yrs (RY).

As discussed above, plants are categorized as either having or not having cross-connects to a second unit on the same site that would be capable of providing RHR capabilities. Plants with cross-connects include Surry 1 and 2. The remaining plants do not have cross-connects. The average remaining lifetimes of plants with cross-connects is about 26 yr. The average remaining lifetimes of plants without cross-connects is about 31 yr.

² Unit 2 has been cancelled but is included in this analysis for comparison purposes with other safety issue prioritization analyses.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this issue are estimated in this section and summarized in Tables 1 and 2. Attachment 1 describes the basis for the risk and dose changes that are estimated in the Tables.

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

RHR Pumps Inside Containment (132).

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

A total of 8 PWRs are affected by this issue.

	$\frac{N}{8}$	$\frac{\bar{T} \text{ (yr)}}{29.4}$
PWRs		

3. Plants Selected for Analysis:

Representative PWR: Surry 1 (see NUREG/CR-4550, Vol. 3).

4. Parameters Affected by SIR:

This issue affects parameter W_3 , failure of the RHR system to cool the reactor, as modeled in NUREG/CR-4450, Vol. 3 (Bertucio and Julius, 1990).

5. Base-Case Values for Affected Parameters:

The base-case value of parameter W_3 assumes that the RHR system is unavailable when demanded to start under harsh environmental conditions. Therefore, the base-case value of W_3 is equal to 1.0 (see Attachment 1). The base-case for Option 2 is represented by the pre-recovery sequence frequencies for the affected accident sequences.

6. Affected Accident Sequences and Base-Case Frequencies:

The affected accident sequences are those sequences in the Surry PRA which include failure of long-term cooling (see Attachment 1). RHR and LPI/LPR systems provide essentially the same function. However, since RHR is not available under harsh containment environments, failure of RHR is not factored into the analysis of sequences that require this function. There are a number of sequences that require LPR (designated event H_1 in the Surry PRA). These sequences would benefit from having RHR available as a backup if RHR were to be environmentally qualified. This would effectively append W_3 to each sequence that includes H_1 .

Option 1:

<u>Sequence</u>	<u>Base-Case Frequency, RY⁻¹</u>	
	<u>Plants With Cross-connects</u>	<u>Plants Without Cross-connects</u>
T ₂ LH ₁	1.40E-08	4.40E-07
T ₃ QH ₁	5.00E-10	1.30E-07
S ₂ H ₁	2.40E-09	1.60E-06
S ₃ O ₆ H ₁	4.60E-09	4.50E-07
TOTAL:	2.15E-08	2.62E-06

Option 2:

<u>Sequence</u>	<u>Base-Case Frequency, RY⁻¹</u>
T ₂ LH ₁	4.40E-07
T ₃ QH ₁	1.30E-07
S ₂ H ₁	1.60E-06
S ₃ W ₃ H ₁	3.70E-07
S ₃ O ₆ H ₁	4.50E-07
TOTAL:	2.99E-06

7. Affected Release Categories and Base-Case Frequencies:

Option 1:

Plants with Cross-connects

$$\begin{array}{lcl} 2.15\text{E-}08/\text{RY} & \gamma \text{ (PWR-3)} & = 1.08\text{E-}08/\text{RY} \\ & \beta \text{ (PWR-5)} & = 1.57\text{E-}10/\text{RY} \\ & \epsilon \text{ (PWR-7)} & = 1.08\text{E-}08/\text{RY} \end{array}$$

Plants without Cross-connects

$$\begin{array}{lcl} 2.62\text{E-}06/\text{RY} & \gamma \text{ (PWR-3)} & = 1.31\text{E-}06/\text{RY} \\ & \beta \text{ (PWR-5)} & = 1.91\text{E-}08/\text{RY} \\ & \epsilon \text{ (PWR-7)} & = 1.31\text{E-}06/\text{RY} \end{array}$$

Option 2:

$$\begin{array}{lcl} 2.99\text{E-}06/\text{RY} & \gamma \text{ (PWR-3)} & = 1.50\text{E-}06/\text{RY} \\ & \beta \text{ (PWR-5)} & = 2.18\text{E-}08/\text{RY} \\ & \epsilon \text{ (PWR-7)} & = 1.50\text{E-}06/\text{RY} \end{array}$$

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

Option 1:

Plants With Cross-connects Plants Without Cross-connects

$\bar{F} = 2.15\text{E-}08/\text{RY}$ $\bar{F} = 2.62\text{E-}06/\text{RY}$

Option 2:

$\bar{F} = 2.99\text{E-}06/\text{RY}$

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

Option 1:

Plants With Cross-connects Plants Without Cross-connects

$W = 5.82\text{E-}02 \text{ person-rem/RY}$ $W = 7.1\text{E+}00 \text{ person-rem/RY}$

Option 2:

$W = 8.1\text{E+}00 \text{ person-rem/RY}$

10. Adjusted-Case Values for Affected Parameters:

For SIR option 1, the adjusted-case value of parameter W_3 represents the unavailability of the RHR system after it has been environmentally qualified. As a result, the value that was derived for this analysis assumes that the reliability of the RHR system is equivalent to that derived in the Surry PRA. The value was estimated in Attachment 1 to be 0.018/demand.

For SIR option 2, the adjusted-case, except for sequence T_2LH_1 , is represented by the accident sequence frequencies given in the Surry 1 PRA after recovery has been factored into the analysis. For sequence T_2LH_1 , the recovery factor given in the Surry 1 PRA was 0.044/demand for failure of the cross-connect. See Attachment 1.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Option 1:

<u>Sequence</u>	<u>Base-Case Frequency, RY^{-1}</u>	
	<u>Plants With Cross-connects</u>	<u>Plants Without Cross-connects</u>
T_2LH_1	2.52E-10	7.92E-09
T_3QH_1	9.00E-12	2.34E-09
S_2H_1	4.32E-11	2.88E-08
$S_3O_6H_1$	8.28E-11	8.10E-09
TOTAL:	3.87E-10	4.72E-08

Option 2:

<u>Sequence</u>	<u>Base-Case Frequency, RY⁻¹</u>
T ₂ LH ₁	1.94E-08
T ₃ QH ₁	5.00E-10
S ₂ H ₁	2.40E-09
S ₃ W ₃ H ₁	5.90E-10
S ₃ O ₀ H ₁	<u>4.60E-09</u>
TOTAL:	<u>2.75E-08</u>

12. Affected Release Categories and Adjusted-Case Frequencies:

Option 1:

Plants with Cross-connects

$$\begin{array}{lcl} 3.87\text{E-}10/\text{RY} & \gamma \text{ (PWR-3)} & = 1.94\text{E-}10/\text{RY} \\ & \beta \text{ (PWR-5)} & = 2.83\text{E-}12/\text{RY} \\ & \epsilon \text{ (PWR-7)} & = 1.94\text{E-}10/\text{RY} \end{array}$$

Plants without Cross-connects

$$\begin{array}{lcl} 4.72\text{E-}08/\text{RY} & \gamma \text{ (PWR-3)} & = 2.36\text{E-}08/\text{RY} \\ & \beta \text{ (PWR-5)} & = 3.45\text{E-}10/\text{RY} \\ & \epsilon \text{ (PWR-7)} & = 2.36\text{E-}08/\text{RY} \end{array}$$

Option 2:

$$\begin{array}{lcl} 2.75\text{E-}08/\text{RY} & \gamma \text{ (PWR-3)} & = 1.38\text{E-}08/\text{RY} \\ & \beta \text{ (PWR-5)} & = 2.01\text{E-}10/\text{RY} \\ & \epsilon \text{ (PWR-7)} & = 1.38\text{E-}08/\text{RY} \end{array}$$

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

Option 1:

Plants With Cross-connects

$$\bar{F}^* = 3.87\text{E-}10/\text{RY}$$

Plants Without Cross-connects

$$\bar{F}^* = 4.72\text{E-}08/\text{RY}$$

Option 2:

$$\bar{F}^* = 2.75\text{E-}08/\text{RY}$$

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

Option 1:

Plants With Cross-connects

$$W^* = 1.05\text{E-}03 \text{ person-rem/Ry}$$

Plants Without Cross-connects

$$W^* = 1.28\text{E-}01 \text{ person-rem/Ry}$$

Option 2:

$$W^* = 7.43E-02 \text{ person-rem/RY}$$

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

Option 1:

Plants With Cross-connects

Plants Without Cross-connects

$$\Delta\bar{F} = 2.11E-08/\text{RY}$$

$$\Delta\bar{F} = 2.57E-06/\text{RY}$$

Option 2:

$$\Delta\bar{F} = 2.96E-06/\text{RY}$$

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

Option 1:

Plants With Cross-connects

Plants Without Cross-connects

$$\Delta W = 5.72E-02 \text{ person-rem/RY}$$

$$\Delta W = 6.97E+00 \text{ person-rem/RY}$$

Option 2:

$$\Delta W = 8.02E+00 \text{ person-rem/RY}$$

17. Total Public Risk Reduction, (ΔW)_{Total}:

	Best Estimate (person-rem)	Error Bounds (person-rem)	
		Upper	Lower
<u>Option 1:</u>	1.3E+03	4.0E+04	0
<u>Option 2:</u>	1.5E+03	4.5E+04	0

ATTACHMENT 1

This safety issue affects the operability of RHR pumps located inside containment under harsh environmental conditions. In most PWRs, these pumps are located outside containment and would therefore not be affected by harsh containment environments. However, at 8 PWRs, the RHR pumps are located inside containment and cannot be relied upon to remove decay heat from the RCS under harsh environmental conditions. The RHR systems that are located inside containment are front-line systems yet are not classified as safety-grade and are not environmentally qualified. At such plants, separate safety-grade low-pressure injection (LPI) and low-pressure recirculation (LPR) systems are provided. In addition to the LPI/LPR system, long-term decay heat removal may be accomplished using AFW and steam generators.

The RHR system is not addressed in the Oconee Probabilistic Risk Assessment (PRA) described by Andrews, et al. (1983) and cannot be used as the basis for this analysis. However, the Surry PRA, NUREG/CR-4550, Vol. 3 (Bertucio and Julius 1990), included RHR system failures as a potential contributing event to a very small LOCA. RHR system failures were not included in analyses of other small LOCAs, including steam generator tube ruptures, or in other initiating events, because they may not be relied upon to operate under harsh containment environments. If the RHR system were environmentally-qualified, it would be possible to use this system to respond to a variety of initiating events. Alternatively, if cross-connects were to be installed between units on the same site, the LPR or RHR function may be accomplished using the second unit's equipment or water sources. The cross-connect would be used in situations in which one unit's LPR system fails on demand. Therefore, two safety issue resolution (SIR) options are examined in this analysis. These are:

Option 1: Plants with and without cross-connects to a second unit on the same site are to undertake the research and analytical work necessary to environmentally qualify their RHR systems for post-accident survivability. Six plants will achieve risk reduction based on analysis of pre-recovery accident sequence frequencies given in the Surry PRA (Bertucio and Julius, 1990). Two plants, which are already provided with cross-connects, will achieve risk reduction based on analysis of post-recovery accident sequence frequencies given in the Surry PRA.

Option 2: Plants without cross-connects are to install them to interconnect the HPI and refueling water storage tanks. Six plants will achieve risk reduction based on the difference between the pre-recovery and post-recovery accident sequence frequencies as given in the Surry PRA. The remaining two plants, which are already provided with cross-connects, will achieve no risk reduction.

The analyses of these two options are discussed in separate sections below.

SIR Option 1: Environmentally Qualify RHR Systems Inside Containment

As discussed previously, an environmentally-qualified RHR system could be available to perform the same function as the Low Pressure Recirculation (LPR) system. LPR failure is designated event H_1 in the Surry PRA. If the RHR system

is qualified to operate in harsh containment environments, it could serve as a backup to the LPR system should it fail. As a result, if SIR is implemented, both the LPR AND RHR systems would be required to fail. The proposed resolution, in effect, would append event W_3 (RHR system failure) to the end of the accident sequences that currently include LPR failure only. For example, accident sequence T_2LH_1 would be modified to $T_2LH_1(W_3)$.

The next step was to identify all of the accident sequences in the Surry PRA that involved event H1 (LPR failure). The unmodified sequences represent the base-case. From this list of sequences, those initiated by large LOCAs (initiating event A) and medium LOCAS (event S_1) were removed, as indicated on p. 4-13 of NUREG-1289 (NRC 1988). The remaining sequences are described below:

Sequence	Description
T_2LH_1	Loss of main feedwater, failure of auxiliary feedwater, successful feed and bleed cooling, failure to provide low pressure recirculation cooling.
T_3QH_1	Turbine trip (with main feedwater available), failure of pressurizer PORV to reclose (produces transient-induced LOCA similar to S_2), failure to provide low pressure recirculation cooling
S_2H_1	Small LOCA, failure to provide low pressure recirculation cooling
$S_3W_3H_1$	Very small LOCA, failure to provide RHR cooling, failure to provide low pressure recirculation cooling.
$S_3O_3H_1$	Very small LOCA, operator fails to depressurize the RCS, failure of high pressure and low pressure recirculation

The base-case frequencies for the affected accident sequences were then taken directly from the Surry PRA. Two sub-cases are analyzed. The first is for plants with cross-connects in which the sequence frequencies after recovery were used. Surry is provided with a cross-connect between Units 1 and 2 that may be utilized if one Unit's LPR system is unavailable. The second case, which was applied to all plants except for Surry, is one in which there are no cross-connects between Units and thus no recovery from LPR failure is possible. The base-case is represented by the accident sequence frequencies before recovery.

The adjusted-case value of W_3 was approximated using information given by Bertucio and Julius (1990). This value was derived by solving two accident sequence formulae given by Bertucio and Julius, as shown below:

$$S_3W_3H_1 = 5.9E-10/RY$$

$$S_2H_1 = 2.4E-09/RY$$

The values for S_2 and S_3 were given as $1.0E-03/RY$ and $1.3E-02/RY$, respectively. Substituting these values into the equations given above results in:

$$\begin{array}{lcl}
 (1.3\text{E}-2)(W_3)(H_1) & = & 5.9\text{E}-10 \quad \therefore (W_3)(H_1) = 4.5\text{E}-08 \\
 (1.0\text{E}-3)(H_1) & = & 2.4\text{E}-09 \quad \therefore H_1 = 2.4\text{E}-06
 \end{array}$$

The value of W_3 was then calculated by substituting the value of H_1 given in the latter equation into the former equation. The resulting adjusted-case value of event W_3 is therefore $1.8\text{E}-02$. This value seems reasonable given that there are no apparent common equipment or components between the RHR and LPR systems (i.e., no common cause failures). These systems appear independent, except for their obvious reliance on instrument air and AC/DC power.

To determine the base-case and adjusted-case core-melt frequencies and public risks, the base-case and adjusted-case values of event W_3 were substituted into the Boolean equations for the affected accident sequences that were given by Bertuccio and Julius (1990). The adjusted-case accident sequence frequencies were determined by multiplying the base-case values for accident sequences involving H_1 by $1.8\text{E}-02$. These Boolean equations and the resulting core-melt frequencies are shown below. Note that sequence $S_3W_3H_1$ is not included in this option because it already assumes that RHR would not be affected by the containment environment produced in this sequence.

PLANTS WITH CROSS-CONNECTS			
Sequence	Base-Case Frequency (RY ⁻¹)	(W_3) Factor	Adjusted-Case Frequency (RY ⁻¹)
T_2LH_1	$1.40\text{E}-08$	$1.80\text{E}-02$	$2.52\text{E}-10$
T_3QH_1	$5.00\text{E}-10$	$1.80\text{E}-02$	$9.00\text{E}-12$
S_2H_1	$2.40\text{E}-09$	$1.80\text{E}-02$	$4.32\text{E}-11$
$S_3O_0H_1$	$4.60\text{E}-09$	$1.80\text{E}-02$	$8.28\text{E}-11$
TOTAL:	$2.15\text{E}-08$		$3.87\text{E}-10$

PLANTS WITHOUT CROSS-CONNECTS			
Sequence	Base-Case Frequency (RY ⁻¹)	(W_3) Factor	Adjusted-Case Frequency (RY ⁻¹)
T_2LH_1	$4.40\text{E}-07$	$1.80\text{E}-02$	$7.92\text{E}-09$
T_3QH_1	$1.30\text{E}-07$	$1.80\text{E}-02$	$2.34\text{E}-09$
S_2H_1	$1.60\text{E}-06$	$1.80\text{E}-02$	$2.88\text{E}-08$
$S_3O_0H_1$	$4.50\text{E}-07$	$1.80\text{E}-02$	$8.10\text{E}-09$
TOTAL:	$2.62\text{E}-06$		$4.72\text{E}-08$

The next step was to calculate the base-case and adjusted-case public risks. This was accomplished by multiplying the base-case and adjusted-case core melt frequencies by appropriate containment failure probabilities and corresponding

public dose consequences. For this analysis, the containment failure modes, release categories, and public dose consequences were taken from Andrews, et al. (1983) for the Oconee PWR. The containment failure modes and release categories for similar sequences given by Andrews, et al. (1980) were used. It was determined that all of the accident sequences affected by this issue would be represented by the same containment failure mode/release category series from Andrews et al. The next step was to multiply the core-melt frequencies given above by the appropriate containment failure probabilities and corresponding public dose estimates, as shown below:

PLANTS WITH CROSS-CONNECTS					
Sequence Frequency, per yr	Containment Failure Mode	Containment Failure Prob.	Release Category	Consequences, person-rem	Public Risk, person-rem/yr
Base-Case					
2.15E-08	γ β ϵ	0.5 0.0073 0.5	PWR-3 PWR-5 PWR-7	5.4E+06 1.0E+06 2.3E+03	5.81E-02 1.57E-04 2.47E-05
TOTAL:					5.82E-02
Adjusted-Case					
3.87E-10	γ β ϵ	0.5 0.0073 0.5	PWR-3 PWR-5 PWR-7	5.4E+06 1.0E+06 2.3E+03	1.04E-03 2.83E-06 4.45E-07
TOTAL:					1.05E-03

PLANTS WITHOUT CROSS-CONNECTS					
Sequence Frequency, per yr	Containment Failure Mode	Containment Failure Prob.	Release Category	Consequences, person-rem	Public Risk, person-rem/yr
Base-Case					
2.62E-06	γ β ϵ	0.5 0.0073 0.5	PWR-3 PWR-5 PWR-7	5.4E+06 1.0E+06 2.3E+03	7.07E+00 1.91E-02 3.01E-03
TOTAL:					7.10E+00
Adjusted-Case					
4.72E-08	γ β ϵ	0.5 0.0073 0.5	PWR-3 PWR-5 PWR-7	5.4E+06 1.0E+06 2.3E+03	1.27E-01 3.44E-04 5.42E-05
TOTAL:					1.28E-01

SIR Option 2: Install Cross-connects to a Second Unit on the Same Site

In this option, cross-connects between two units on the same site are proposed to be installed to provide a backup to the LPR system at each unit. No efforts are necessary to environmentally-qualify the RHR systems inside containment. The base-case accident sequence frequencies are the pre-recovery frequencies given in the Surry PRA. The adjusted-case accident sequence frequencies are the post-recovery frequencies given in the Surry PRA. The risk reduction resulting from this option is applied to a total of six plants (cross-connects have already been installed at the other two units affected by this issue).

The cross-connect option was included in the Surry PRA for recovery of all of the sequences addressed by this issue except sequence T_2LH_1 . The recovery option for this sequence involved recovery of auxiliary feedwater which is not within the scope of this issue. For this sequence, therefore, the reduction in core-melt frequency cannot be represented by the difference between the pre- and post-recovery sequence frequencies given in the Surry PRA. However, use of the cross-connect to the Unit 2 HPI system or RWST would also result in increased availability of long-term cooling. To quantify the effect of the cross-connect on this sequence, the adjusted-case frequency for this sequence was calculated by multiplying the pre-recovery sequence frequency by the recovery factor that was given for the cross-connect to HPI or RWST. The recovery factor given by Bertucio and Julius (1990) was 0.044/demand. The core-melt frequency and public risk calculations for this option are summarized in the following tables.

PLANTS INSTALL CROSS-CONNECTS			
Sequence	Base-Case Frequency RY(-1)	Factor	Adjusted-Case Frequency RY(-1)
T_2LH_1	4.40E-07	4.40E-02	1.94E-08
T_3QH_1	1.30E-07	NA ^(a)	5.00E-10
S_2H_1	1.60E-06	NA	2.40E-09
$S_3W_3H_1$	3.70E-07	NA	5.90E-10
$S_3O_3H_1$	4.50E-07	NA	4.60E-09
TOTAL:	2.99E-06		2.75E-08

- (a) NA = Not applicable. The base-case and adjusted-case frequencies for these sequences were taken directly from Bertucio and Julius (1990).

PLANTS INSTALL CROSS-CONNECTS					
Sequence Frequency, per yr	Containment Failure Mode	Containment Failure Prob.	Release Category	Consequences, person-rem	Public Risk, person-rem/yr
Base-Case					
2.99E-06	γ β ϵ	0.5 0.0073 0.5	PWR-3 PWR-5 PWR-7	5.4E+06 1.0E+06 2.3E+03	8.07E+00 2.18E-02 3.44E-03
TOTAL:					8.10E+00
Adjusted-Case					
2.75E-08	γ β ϵ	0.5 0.0073 0.5	PWR-3 PWR-5 PWR-7	5.4E+06 1.0E+06 2.3E+03	7.41E-01 2.00E-04 3.16E-05
TOTAL:					7.43E-01

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

RHR Pumps Inside Containment (132).

2. Affected Plants (N):

PWRs	$\frac{N}{8}$
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3. Average Remaining Lives of Affected Plants ($\bar{T}$):

PWRs	$\frac{\bar{T}}{29.4 \text{ yr}}$
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4. Per-Plant Occupational Dose Reduction due to Accident Avoidance, $\Delta(\bar{F}D_R)$:

Option 1:

Plants With Cross-connects

$$\Delta(\bar{F}D_R) = (19860 \text{ person-rem})(2.11E-08/\text{RY}) = 4.2E-04 \text{ person-rem/RY}$$

Plants Without Cross-connects

$$\Delta(\bar{F}D_R) = (19860 \text{ person-rem})(2.57E-06/\text{RY}) = 5.1E-02 \text{ person-rem/RY}$$

Option 2:

$$\Delta(\bar{F}D_R) = (19860 \text{ person-rem})(2.96E-06/\text{RY}) = 5.9E-02 \text{ person-rem/RY}$$

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

	Best Estimate (person-rem)	Error Bounds (person-rem)	
		Upper	Lower
<u>Option 1:</u>	9.5E+00	5.8E+01	0
<u>Option 2:</u>	1.1E+01	6.6E+01	0

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

For Option 1, incremental labor in radiation zones and occupational dose increases due to SIR implementation, operation, and maintenance are estimated to be 0. There are no hardware fixes and no plant modifications in radiation zones associated with SIR. Therefore, the occupational dose estimates for this option are zero.

For Option 2, SIR involves installation of a cross-connect between two units which will involve work in radiation zones. The total labor-hrs for installation of the cross-connect is calculated in Table 3 to be 9.1E+04 man-hr. It was assumed that 20% of the installation labor (1.8E+04 man-hr)

is spent inside containment in a 25 mrem/hr radiation field, 20% is spent outside containment in a 2.5 mrem/hr field, and the remaining 60% is spent in non-radiation zone.

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

Option 1: $D = 0$

Option 2: $D = (1.8E+04 \text{ man-hr})(25 \text{ mrem/hr}) + (1.8E+04 \text{ man-hr})$
 (2.5 mrem/hr)
 $= 5.0E+02 \text{ man-rem/site}$

As shown, the installation labor estimates are presented on a per-site basis. This increase in occupational doses is assumed to be applied to a total of 3 sites consisting of a total of 6 plants.

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

Option 1: $ND = 0$

Option 2: $ND = (5.0E+02 \text{ person-rem/site})(3 \text{ sites})$
 $= 1.5E+03 \text{ person-rem}$

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

For Option 1, the occupational dose estimates are zero.

For option 2, a small amount of utility labor in radiation zones will be required to perform routine inspections of the cross-connects. It was estimated that 8 man-hr/Ry will be spent inside containment and 8 man-hr/Ry will be spent outside containment to perform this function.

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

Option 1: $D_o = 0$

Option 2: $D_o = (8 \text{ man-hr/Ry})(25 \text{ mrem/hr}) + (8 \text{ man-hr/Ry})(2.5 \text{ mrem/hr})$
 $= 0.22 \text{ man-rem/Ry}$

11. Total Occupational Dose Increase for SIR Operation and Maintenance (NTD_o):

Option 1: $NTD_o = 0$

Option 2: $NTD_o = (2.2E-01 \text{ person-rem/Ry})(6 \text{ plants})(31 \text{ yr})$
 $= 4.1E+01 \text{ person-rem}$

12. Total Occupational Dose Increase (G):

	Best Estimate <u>(man-rem)</u>	Error Bounds (man-rem) <u>Upper</u>	<u>Lower</u>
<u>Option 1:</u>	0		
<u>Option 2:</u>	1.5E+03	4.6E+03	5.1E+02

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

RHR Pumps Inside Containment (132).

2. Affected Plants (N):

PWRs $\frac{N}{8}$

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

PWRs $\frac{\bar{T}}{29.4 \text{ yr}}$

Industry Costs (Steps 4 through 12):

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, $\Delta(\bar{F}A)$:
Option 1:

Plants With Cross-connects

$$\Delta\bar{F}A = (\$1.65E+09)(2.11E-08/\text{RY}) = \$3.5E+01/\text{RY}$$

Plants Without Cross-connects

$$\Delta\bar{F}A = (\$1.65E+09)(2.57E-06/\text{RY}) = \$4.2E+03/\text{RY}$$

Option 2:

$$\Delta\bar{F}A = (\$1.65E+09)(2.96E-06/\text{RY}) = \$4.9E+03/\text{RY}$$

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

	<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
<u>Option 1:</u>	\$7.9E+05	\$4.8E+06	0
<u>Option 2:</u>	\$9.1E+05	\$5.5E+06	0

6. Per-Plant Industry Resources for SIR Implementation:

For SIR Option 1, industry costs for SIR implementation include the resources necessary to environmentally qualify their RHR systems or to replace unqualified RHR systems with a new system that could meet the

environmental qualification requirements. A conservative approach was used in this analysis which tends to minimize the implementation costs, i.e., no pump or system replacements are assumed to be needed. All affected plants are assumed to be capable of qualifying their existing RHR system. Resource requirements to perform the environmental qualification tests and analyses on existing RHR systems are highly uncertain. For this analysis, a 2 man-yr effort per plant was estimated to complete these tests/analyses. An additional 0.5 man-yr/plant was estimated to be needed to prepare a safety evaluation regarding the environmental qualification of RHR systems that are located inside containment. Therefore, the total industry resources for SIR implementation were estimated to be 2.5 man-yr/plant.

Implementation of SIR Option 2 will require installation of cross-connects between 2 units on the same site. Costs were estimated for installation of 1000 linear ft. of 12-in. pipe, pipe hangers, and 4 valves in each line. A six man-month effort to design the cross-connect and perform the required safety analyses was estimated to be required. Basic cost and labor data were taken from NUREG/CR-4627, Rev. 2 (Sciaccia 1989). The hardware unit costs and installation labor estimates are as follows:

Design: 6 man-months

Piping: Hardware - $\$56/\text{linear-ft(LF)} \times 1000 \text{ LF} = \$56,000$
 Installation labor - $22 \text{ man-hr/LF} \times 1000 \text{ LF} = 22,000 \text{ man-hr}$

Hangers: Hardware - $\$27/\text{LF} \times 1000 \text{ LF} = \$27,000$
 Installation labor - $1.7 \text{ man-hr/LF} \times 1000 \text{ LF} = 1700 \text{ man-hr}$

Valves: Hardware - $\$8305/\text{valve} \times 4 \text{ valves} = \$33,220$
 Installation labor - $61.6 \text{ man-hr/valve} \times 4 \text{ valves} = 250 \text{ man-hr}$

This results in a total installation labor estimate of about $2.4\text{E}+04$ man-hr. According to Sciaccia (1990), these "greenfield" estimates should be adjusted to account for reduced productivity that results from work in congested areas, in radiation zones, and access and handling difficulties. The total labor productivity adjustment factor was calculated to 3.8. Therefore, the total installation labor requirements are estimated to be approximately $9.1\text{E}+04$ man-hr.

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

Option 1:

$$I = (2.5 \text{ man-yr/plant})(\$100,000/\text{man-yr}) = \$2.5\text{E}+05/\text{plant}$$

Option 2:

$$\begin{aligned} I &= \text{Design} + \text{Hardware} + \text{Labor} \\ &= \{[(6 \text{ man-months})/(12 \text{ months/yr})] * \$100,000/\text{man-yr}\} \\ &\quad + \{\$56,000 + 27,000 + 33,220\} \\ &\quad + \{9.1\text{E}+04 \text{ man-hr} \times \$56.75/\text{man-hr}\} \\ &= \$5.2\text{E}+06/\text{site} \end{aligned}$$

As shown, the installation labor estimates are presented on a per-site basis. The SIR implementation costs are assumed to be applied to 3 sites consisting of a total of 6 plants.

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

Option 1:

$$NI = (\$2.5E+05/\text{plant})(8 \text{ plants}) = \$2.0E+06$$

Option 2:

$$NI = (\$5.2E+06/\text{site})(3 \text{ sites}) = \$5.2E+07$$

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

Increased operation and maintenance labor requirements associated with SIR Option 1 were assumed to be required to enhance inspection/testing of RHR system components. An additional 2 man-day/RY were estimated for this activity. This estimate is also applicable to inspection/testing of the cross-connect that would be installed if Option 2 were implemented.

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

Options 1 and 2:

$$I_o = [(2 \text{ man-day/RY})/(5 \text{ man-day/man-wk})] (\$2270/\text{man-wk}) = \$908/\text{RY}$$

In option 1, this cost is applied to a total of 8 plants. In option 2, this cost is applied to only 6 plants.

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

Option 1:

$$NTI_o = (\$908/\text{RY})(8 \text{ plants})(29.4 \text{ yr}) = \$2.2E+05$$

Option 2:

$$NTI_o = (\$908/\text{RY})(6 \text{ plants})(29.4 \text{ yr}) = \$1.7E+05$$

12. Total Industry Cost (S_T):

	<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
<u>Option 1:</u>	\$2.2E+06	\$3.2E+06	\$1.2E+06
<u>Option 2:</u>	\$1.6E+07	\$2.4E+07	\$8.0E+06

13. NRC Resources for SIR Development:

For SIR Option 1, NRC development costs are assumed to include a research program to investigate RHR system survivability under harsh environments, to predict the appropriate environmental conditions that RHR system must be capable of withstanding in order to be qualified, and to investigate other means of decay heat removal (e.g., ECCS, AFW, steam generators) under harsh environments. The environmental qualification requirements would then be issued to the affected plants. The NRC labor requirements for the environmental qualification research program are estimated to be 1 man-yr of contractor labor plus 0.1 man-yr of NRC staff labor to monitor the contractor. The resource requirements that are needed to issue the new requirements to the affected plants are estimated to be 3 man-wk, based on information given by Sciacca (1989).

For SIR Option 2, SIR development resources will include a research program similar to that described for Option 1. However, rather than focusing on environmental qualification requirements, the research program in Option 2 would focus on the design, operation, and safety-related requirements of cross-connects. It is believed that this will be somewhat more complex problem than that of Option 1 so a larger resource estimate is appropriate. It was estimated that the research program would involve approximately 3 man-yr of contractor resources in addition to NRC resources necessary to monitor the contractor and to issue the new requirements. A total of 4 man-yr of NRC resources are estimated for development of this Option.

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

Option 1:

$$\begin{aligned} C_D &= (1.1 \text{ man-yr})(\$100,000/\text{man-yr}) + (3 \text{ man-wk})(\$2270/\text{man-wk}) \\ &= \$1.2\text{E}+05 \end{aligned}$$

Option 2:

$$C_D = (4 \text{ man-yr})(\$100,000/\text{man-yr}) = \$4.0\text{E}+05$$

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

According to Sciacca (1989), NRC labor requirements to review a relatively uncomplicated technical specification change is estimated to be 3.5 man-wk/plant, including 2.5 man-wk of technical staff resources and 1 man-wk for legal reviews and concurrences. This resource requirement was used in this analysis to estimate NRC implementation costs that are required to review and approve the environmental qualification analyses and safety evaluations prepared by the affected plants. This estimate is applicable to Option 1.

For Option 2, a larger effort was deemed necessary to review the cross-connect designs and safety analyses. A total of 2 man-months/plants was estimated for this activity.

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

Option 1:

$$C = (3.5 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$7.9\text{E}+03/\text{plant}$$

Option 2:

$$C = (2 \text{ man-months/plant})(4 \text{ man-wk/man-month})(\$2270/\text{man-wk}) \\ = \$1.8\text{E}+04/\text{plant}$$

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

Option 1:

$$NC = (8 \text{ plants})(\$7945/\text{plant}) = \$6.4\text{E}+04$$

Option 2:

$$NC = (6 \text{ plants})(\$1.8\text{E}+04/\text{plant}) = \$1.1\text{E}+05$$

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

For Option 1, NRC resource requirements for review of SIR operation and maintenance are anticipated to minimal. An additional 0.5 man-day/Ry was estimated to be required for an annual review of RHR system availability and to review the enhanced inspection/testing program. For Option 2, the labor requirements were estimated to be twice as high as that for Option 1. Therefore, NRC labor requirements for review of operation and maintenance were estimated to be 1 man-day/Ry.

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

Option 1:

$$C_o = [(0.5 \text{ man-day/Ry})/(5 \text{ man-day/man-wk})] (\$2270/\text{man-wk}) = \$227/\text{Ry}$$

Option 2:

$$C_o = [(1.0 \text{ man-day/Ry})/(5 \text{ man-day/man-wk})] (\$2270/\text{man-wk}) = \$454/\text{Ry}$$

20. Total NRC Cost for Review of SIR Operation and Maintenance (N $\overline{TC}_o$):

Option 1:

$$N\overline{TC}_o = (\$227/\text{Ry})(8 \text{ plants})(29.4 \text{ Ry/plant}) = \$5.4\text{E}+04$$

Option 2:

$$N\overline{TC}_o = (\$454/\text{Ry})(6 \text{ plants})(31 \text{ Ry/plant}) = \$8.4\text{E}+04$$

21. Total NRC Cost (S_N):

	<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
<u>Option 1:</u>	\$2.3E+05	\$3.1E+05	\$1.6E+05
<u>Option 2:</u>	\$5.9E+05	\$8.0E+05	\$3.8E+05

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 137, Refueling Cavity Seal Failure

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: Failure of inflatable refueling cavity seals may cause water in the refueling cavity to drain to the containment floor. If spent fuel is being transferred to the spent fuel storage pool at the time of failure, the water level in the refueling cavity and fuel transfer tube may drop significantly resulting in high radiation levels and possible fuel damage. Seal failure at some plants may also result in drainage of the spent fuel pool to below safe levels resulting in larger scale fuel damage. The resolution involves replacing the inflatable seals with improved-design seals. In addition, for plants with widely-spaced double seals, safety issue resolution involves installation of a permanent steel seal.

AFFECTED PLANTS: BWR: Operating = 0 Planned = 4
 PWR: Operating = 28 Planned = 13

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 5.7E-4

OCCUPATIONAL DOSES:

SIR Implementation =	66
SIR Operation/Maintenance =	50
Total of Above =	120
Accident-Avoidance =	5.8

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	2.5
SIR Operation/Maintenance =	4.0
Total of Above =	6.5
Accident-Avoidance =	2.2

NRC COSTS:

SIR Development =	0.20
SIR Implementation Support =	0.22
SIR Operation/Maintenance Review =	4.1
Total of Above =	4.5

REFUELING CAVITY SEAL FAILURE

ISSUE 137

1.0 SAFETY ISSUE DESCRIPTION

In August of 1984, while in preparation for refueling, the Haddam Neck Plant experienced a failure of a refueling cavity seal. This failure caused the water level in the refueling cavity to drop to the level of the reactor vessel flange and flooded containment with approximately 200,000 gallons of water. No fuel was being transferred to the spent fuel storage pool at the time the failure occurred. However, if fuel was being transferred, it could have been partially or totally uncovered resulting in high radiation levels, fuel cladding failure, and release of radioactivity. Furthermore, because the refueling cavity is connected by the fuel transfer tube to the spent fuel storage pool, the potential exists for this seal failure to cause partial or total drainage of the storage pool. The fuel transfer tube would have to be open, either intentionally or as a result of failure of the fuel transfer tube seals, for the refueling cavity seal failure incident to result in drainage of the fuel storage pool.

Pneumatic rubber seals are most vulnerable to failure because they are susceptible to puncture, misalignment, rupture, or improper inflation. Other types of seals, such as the permanent stainless steel bellows seals on most Boiling Water Reactors (BWRs), have proven to be more reliable than inflatable rubber seals (Sailor, et al. 1987).

The safety concerns regarding this issue are generic in that loss of water from the refueling system could lead to a sequence of events not previously considered by the NRC staff, i.e., drainage of water from the refueling cavity that results in (1) uncovering fuel assemblies in the transfer tube, and (2) drainage of the spent fuel storage pool to below the top of the stored fuel assemblies. These sequences are not explicitly considered in the Standard Review Plan nor in the NRC guidance on acceptability of facility designs, establishment of technical specifications, and establishment of operating and emergency procedures. As a result of the Haddam Neck incident, the NRC issued IE Bulletin 84-03 (NRC 1984) which required licensees to investigate potential refueling cavity seal failures at their plants. The responses to this Bulletin were a major source of background information for this safety issue.

It is noted here that this issue is a subset of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools." However, Issue 82 is primarily concerned with accidents within the spent fuel pool itself and lesser emphasis was placed on refueling cavity seals. Consequently, the present issue focuses on the risk and dose impacts associated with proposed resolutions to the refueling cavity seal failure issue.

PROPOSED SAFETY ISSUE RESOLUTION

Haddam Neck and NRC staffmembers have extensively studied refueling cavity seals and proposed solutions to the seal failure problem. Several of the proposed resolutions to this issue have been implemented at Haddam Neck.

The proposed resolution to this issue is based on information developed by Haddam Neck staff (letter from W. G. Council to Director of Nuclear Reactor Regulation 1984) and the NRC (Minners 1986). The first item in the proposed resolution to this issue involves reinforcing the refueling cavity seals by inserting steel pins in the flange at the top of the seal. This prevents the seal from slipping through the gap between the reactor vessel flange and biological shield. This is proposed as a potential "fix" for plants that require improved-design inflatable rubber seals. Second, a "U" shaped metal backup seal was temporarily installed at Haddam Neck over each of the two inflatable seals. If the inflatable seal were to fail, the backup seal would prevent the refueling cavity from quickly draining, allowing the operators sufficient time to correct the problem. This was only a temporary "fix"; it is not proposed to be implemented at other plants. Third, a coffer dam was installed to prevent the drainage of the spent fuel pool in the event of a refueling cavity seal failure. This is a proposed "fix" for plants at which spent fuel pool drainage through the refueling cavity could occur. During a later outage at Haddam Neck, the inflatable seals and backup seal were discarded and a permanent steel seal was welded in place. The permanent seal prevents any water in the refueling cavity from leaking down to the containment floor. This is a proposed "fix" for plants which are currently equipped with double inflatable seals. Finally, emergency guidelines were developed to instruct the reactor operators in the event of a seal failure.

The Haddam Neck plant represents a unique case because it has a wide gap between the reactor vessel and biological shield requiring a metal seal ring and double inflatable seals. The changes implemented at Haddam Neck may not apply to plants with single inflatable seals. However the permanent steel seal installed at Haddam Neck that is proposed here is assumed to resolve the issue for plants with double inflatable seals. It is further assumed that a redesigned inflatable seal will effectively eliminate the risk of refueling cavity seal failure in plants that have only a single inflatable cavity seal.

In summary, plants currently equipped with single inflatable seals are assumed to replace them with improved-design inflatable seals. Plants with double inflatable seals are assumed to replace them with permanent steel seals. Plants that can potentially uncover spent fuel pool in the storage pool as a result of a refueling cavity seal failure are assumed to install a coffer dam between the storage pool and the refueling cavity to prevent this event from occurring. Development of emergency guidelines for reactor operators to respond to failures of refueling cavity seals is assumed to be implemented at all plants.

AFFECTED PLANTS

As a result of the Haddam Neck seal failure, the Nuclear Regulatory Commission (NRC) issued IE Bulletin 84-03 which required licensees to inspect their refueling cavity seals. A review of the licensee responses to this Bulletin performed by Sailor, et al. (1987), indicated that 45 plants are equipped with inflatable rubber seals. Of these 45 plants, 38 are operating PWRs, 3 are PWRs under construction, and 4 are operating BWRs. Thirty-six of the 45 affected plants are currently equipped with single inflatable seals and 9 are equipped with double seals. A total of 5 plants indicated in their response to IE Bulletin 84-03 that they could potentially drain their spent fuel storage pool through a failed refueling cavity seal.

2.0 SAFETY ISSUE RISK AND DOSE

Estimates of public risk reduction and occupational dose are included in this section with calculations summarized in Tables 1 and 2, respectively.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Refueling Cavity Seal Failure (137)

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

Responses to the NRC Bulletin 84-03 show that a total of at least 45 plants use inflatable refueling cavity seals.

	<u>N</u>	<u>$\bar{T}$</u>
PWR (planned)	13	30.0 yr
(operating)	28	27.5 yr
BWR (planned)	4	30.0 yr

3. Plants Selected for Analysis:

This accident is not addressed in detail in any plant-specific probabilistic risk assessments. Therefore, the safety issue risk and dose estimates are derived based on a hypothetical PWR and BWR. It is assumed that the frequency of refueling cavity seal failures is equal for both PWRs and BWRs.

4. Parameters Affected by SIR:

The parameters Y and Z are defined as the affected parameters for this safety issue because the two accident sequences described in Section 1.0 have not been addressed in previous PRAs. Parameter "Y" represents the frequency of refueling cavity seal failure results in uncover of spent fuel in the refueling canal. Parameter "Z" represents the frequency of refueling cavity seal failures that results in drainage of the spent fuel pool and uncover of stored spent fuel.

5. Base-Case Values for Affected Parameters:

$$Y = 1.9E-6/\text{py}$$

$$Z = 5.0E-7/\text{py}$$

These parameter values are assumed to apply to both PWRs and BWRs.
(See Attachment 1)

6. Affected Accident Sequences and Base-Case Frequencies:

<u>Sequence</u>	<u>Frequency</u>
Refueling cavity seal failure causing transfer canal drainage:	
Y B (PWR-9 * 0.01)	1.9E-06/py
Y B (BWR-5 * 0.01)	1.9E-06/py
Refueling cavity seal failure causing fuel pool drainage:	
Z B (PWR-9)	5.0E-07/py
Z B (BWR-5)	5.0E-07/py

(See Attachment 1)

7. Affected Release Categories:

<u>Category</u>	<u>Frequency (1/py)</u>	
	<u>Spent Fuel Pool Drainage</u>	<u>Transfer Canal Drainage</u>
PWR-9	3.7E-9	1.4E-8
BWR-5	3.7E-9	1.4E-8

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

This issue does not involve a core-melt accident. However, the base-case, affected accident frequency for both PWRs and BWRs is 2.4E-6/py (see Attachment 1). The release frequency (i.e., accident frequency times the containment failure probability) for both plant types is 1.8E-8/py.

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

	<u>Spent Fuel Pool Drainage</u>	<u>Transfer Canal Drainage</u>	<u>Total</u>
	man-rem/py	man-rem/py	man-rem/py
W (PWR)	4.4E-7	1.7E-8	4.6E-7
W (BWR)	2.8E-7	2.8E-9	2.8E-7

(See Attachment 1)

10-14. Steps related to Adjusted-Case Affected Parameters, Accident Sequences, Core-Melt Frequency, and Public Risk:

SIR is assumed to effectively eliminate the problem of failure of inflatable refueling cavity seals in reactors. Consequently, the adjusted-case, affected accident sequence frequencies and public risks are effectively zero.

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

This issue does not involve damage to fuel in the reactor core. Thus, there is no reduction in core-melt frequencies. However, the reduction in

accident sequence frequencies for both PWRs and BWRs is estimated to be $2.4\text{E-}06/\text{py}$.

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

$$\Delta W \text{ (PWR)} = 4.6\text{E-}7 \text{ man-rem/py}$$

$$\Delta W \text{ (BWR)} = 2.8\text{E-}7 \text{ man-rem/py}$$

17. Total Public Risk Reduction, (ΔW)_{total}:

Best Estimate (man-rem)	Error Bounds Upper	(man-rem) Lower
<u>5.7E-4</u>	<u>1.7E-2</u>	<u>0</u>

ATTACHMENT 1

The parameters Y and Z are defined here as the probability of inflatable refueling cavity seal failure resulting in serious transfer canal drainage and spent fuel pool drainage, respectively. The first step in calculating frequencies for both Y and Z was to estimate the failure probability for Y which results in serious transfer canal drainage, we begin with the frequency of refueling cavity seal failure. The historical seal failure rate is estimated at approximately $1.0\text{E-}2/\text{py}$ based on information developed by Sailor, et al. (1987).

This estimate was confirmed by calculating the historical failure rate. This was done by dividing the number of historical seal failures by the total years of reactor operation. There have been seven incidents of inflatable seal failures to date. This number of events was divided by the total accumulated reactor-yrs of operation (estimated here at approximately 1170 reactor-years). The result, $6\text{E-}3/\text{py}$, is reasonably close to the estimate of $1\text{E-}2/\text{py}$ seal failure rate derived by Sailor, et al. (1987). In addition, Sailor, et al., estimate the seal failure rate has decreased by a factor of ten, from $1\text{E-}2/\text{py}$ to $1\text{E-}3/\text{py}$, due to improvements in design and increased awareness of the problem following the Haddam Neck incident.

To estimate the frequency of transfer canal drainage accidents (Y), the probability of refueling cavity seal failure was multiplied by the probability that spent fuel is being transferred at the time of seal failure and by the probability that reactor operators do not recover from the seal failure incident in time to prevent drainage of the refueling cavity. The probability that fuel is being transferred is estimated based on the assumption that an average plant spends two weeks per year refueling:

$$2 \text{ wks refueling}/52 \text{ wks per year} = 3.8\text{E-}2 \text{ refueling/py}$$

The probability of nonrecovery is taken from Sailor, et al. (1987) in which this probability is estimated to be $5\text{E-}2/\text{demand}$. Thus, the probability of refueling cavity seal failure resulting in serious transfer canal drainage (Y) is estimated at:

$$Y = (1.0\text{E-}3/\text{py}) * (3.8\text{E-}2) * (5\text{E-}2) = 1.9\text{E-}6/\text{py}$$

The parameter Z, the probability of refueling cavity seal failure resulting in serious spent fuel pool drainage, is estimated to be the product of the probability of spent fuel pool drainage and the probability of no recovery. The probability of spent fuel pool drainage resulting from inflatable seal failure has been estimated at $1\text{E-}5/\text{py}$ (Sailor, et al. 1987). This value incorporates the estimated seal failure rate of $1\text{E-}3/\text{py}$. The probability of nonrecovery from this incident is assumed to be same value as that used above to estimate the value for parameter Y ($5\text{E-}2/\text{demand}$). The resulting frequency for parameter Z is:

$$Z = (1.0\text{E-}5/\text{py}) * (5.0\text{E-}2/\text{demand}) = 5.0\text{E-}7/\text{py}$$

The next step is to multiply the accident frequencies by their respective containment failure probabilities. The applicable containment failure mode in this case is assumed to be represented by containment penetration leakage (B).

This is because occurrence of accidents Y and Z will not produce a core-melt accident. This, in turn, precludes steam explosions, hydrogen burning, and base metal melt through containment failure modes. The containment failure probability used in this issue was taken from NUREG/CR-2800 (Andrews, et al., 1980), Appendix A, which is based on the Oconee-3 PWR. The containment failure probability associated with penetration leakage is $7.3\text{E-}3$. The release frequencies for parameters Y and Z are then calculated as follows:

Transfer canal:

$$F_y = Y * 7.3\text{E-}3 = (1.9\text{E-}6/\text{py}) * (7.3\text{E-}3) = 1.4\text{E-}8/\text{py}$$

Spent Fuel Pool:

$$F_z = Z * 7.3\text{E-}3 = (5.0\text{E-}7/\text{py}) * (7.3\text{E-}3) = 3.7\text{E-}9/\text{py}$$

$$\bar{F} = F_y + F_z = 1.77\text{E-}8/\text{py for both PWRs and BWRs.}$$

Radiation exposures to the public as a result of the seal failure accidents were calculated using release categories PWR-9 and BWR-5 which represent public doses from non-core-melt accidents. NUREG/CR-2800 (Andrews, et al., 1980) provided the following dose estimates for these release categories:

<u>Category</u>	<u>Dose Consequence</u>
PWR-9	120 man-rem
BWR-5	20 man-rem

These dose factors are assumed to be applicable to the spent fuel pool drainage scenario but must be scaled down to estimate the doses that would result from drainage of the transfer canal. The dose factors represent the uncovering of the spent fuel in the reactor core, which contains approximately 100 MTU of fuel. The refueling cavity may contain only several fuel assemblies at a time or an estimated 1 MTU of fuel. Therefore, it is estimated that the public doses from the transfer canal accident would be about 1% of the public doses that were estimated to result from the spent fuel pool uncovering accident. The resulting public dose estimate for the refueling canal accident scenario are 1.2 man-rem for PWR plants and 0.2 man-rem for BWR plants.

The base-case affected public risk is obtained by multiplying release probabilities (F_y and F_z) times their corresponding public dose consequences. The per-plant base-case affected public risk estimates are:

Spent Fuel Pool Drainage:

PWR	$3.7\text{E-}9/\text{py}$	*	120 man-rem	=	$4.4\text{E-}7$ man-rem/py
BWR	$3.7\text{E-}9/\text{py}$	*	20 man-rem	=	$2.8\text{E-}7$ man-rem/py

Transfer Canal Drainage:

PWR	$1.4\text{E-}8/\text{py}$	*	1.2 man-rem	=	$1.7\text{E-}8$ man-rem/py
BWR	$1.4\text{E-}8/\text{py}$	*	0.2 man-rem	=	$2.8\text{E-}9$ man-rem/py

$$W (\text{PWR}) = (4.4\text{E-}7 + 1.7\text{E-}8) = 4.6\text{E-}7 \text{ man-rem/py}$$

$$W (\text{BWR}) = (2.8\text{E-}7 + 2.8\text{E-}9) = 2.8\text{E-}7 \text{ man-rem/py}$$

Issue resolution is assumed to effectively eliminate the problem of refueling cavity seal failure. Thus the adjusted-case values for parameters Y and Z are effectively zero, i.e., $Y_a \ll Y$ and $Z_a \ll Z$. The adjusted-case affected

accident sequence frequencies and adjusted-case public risks are effectively zero as a result of SIR.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Refueling Cavity Seal Failure (137)

2. Affected Plants (N):

		<u>N</u>
PWR	(planned)	13
	(operating)	28
BWR	(planned)	4
Total		45

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

		<u>$\bar{T}$</u>
PWR	(planned)	30.0 yr
	(operating)	27.7 yr
BWR	(planned)	30.0 yr

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

Spent fuel pool drainage:
 $(5.0E-7/\text{py}) (1880 \text{ man-rem}) = 9.4E-4 \text{ man-rem/py}$

Transfer canal drainage:
 $(1.9E-6/\text{py}) (1880 \text{ man-rem}) = 3.6E-3 \text{ man-rem/py}$

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

<u>Best Estimate</u>	<u>Error Bounds (man-rem)</u>	
<u>(man-rem)</u>	<u>Upper</u>	<u>Lower</u>
5.8	35	0

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

Additional radiation exposure will be accumulated by personnel during the implementation of the SIR. Installation of a permanent seal requires a significant amount of labor inside the radiation zone. The permanent seal also requires periodic inspection with associated occupational exposures. SIR is not expected to affect occupational exposures associated with inspection and installation of improved-design seals prior to refueling. This is because the inflatable seals are removed after the previous refueling outage so seal inspections are conducted outside radiation zones and therefore do not affect occupational exposures. It is assumed that SIR will not change installation procedures for inflatable seals; the improved seals are assumed to require the same amount of time to install as the previous designs. Therefore the corresponding occupational exposure will not change.

It is estimated that installation of a permanent seal requires 1 man-month (160 man-hrs) inside the radiation zone. This includes labor for

installation of the seal, radiation monitoring, and quality assurance inspection. Once installed, the permanent seal requires periodic inspections which must occur within the radiation zone. A total of 8 man-hours per plant-yr is estimated for this inspection activity.

Installation of a coffer dam in the spent fuel storage pool is estimated to require 6 man-wks of utility labor in a radiation zone. This includes initial installation, radiation monitoring, and quality assurance inspection of the completed structure. No periodic inspection of the coffer dam after the initial installation is assumed to be required.

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

Installation of permanent seals:
 $(160 \text{ man-hrs}) (25 \text{ mR/hr}) = 4 \text{ man-rem/plant}$

Installation of coffer dams:
 $(240 \text{ man-hrs}) (25 \text{ mR/hr}) = 6 \text{ man-rem/plant}$

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

Plants requiring installation of a permanent seal (total of 9 affected plants):
 $(4 \text{ man-rem/plant}) (9 \text{ plants}) = 36 \text{ man-rem}$

Plants requiring installation of coffer dams (total of 5 affected plants):
 $(6 \text{ man-rem/plant}) (5 \text{ plants}) = 30 \text{ man-rem}$

ND (Total): 66 man-rem

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

As discussed in Step 6, utility labor requirements for inspecting permanent seals, which must be performed in a radiation zone, is estimated at 8 man-hrs/py. No additional operation and maintenance labor requirements are assumed to be needed for improved-design inflatable seals and coffer dams.

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

$(D_o) = (8 \text{ man-hrs/py}) (25 \text{ mR/hr}) = 0.2 \text{ man-rem/py}$

11. Total Occupational Dose Increase for SIR Operation and Maintenance (NTD_o):

A total of 9 PWRs and 0 BWRs are assumed to require installation of permanent seals. All 9 affected PWRs are currently in operation)

$NTD_o = (9 \text{ plants}) (27.7 \text{ yrs}) (0.2 \text{ man-rem/py}) = 50 \text{ man-rem}$

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
1.2E+02	3.6E+02	4.0E+01

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Refueling Cavity Seal Failure (137)

2. Affected Plants (N):

		<u>N</u>
PWR	(planned)	13
	(operating)	28
BWR	(planned)	4
Total		45

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

		<u>$\bar{T}$</u>
PWR	(planned)	30.0 yr
	(operating)	27.7 yr
BWR	(planned)	30.0 yr

Industry Costs (steps 4 through 12)

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

$$\Delta(\bar{F}A) = (2.4E-6/\text{py}) (\$7.2E+8) = \$1,730/\text{py}$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$2.2E+6	\$1.3E+7	0

6-7. Per-Plant Industry Resources and Cost for SIR Implementation (I):

The following were obtained from conversations with utility staff and are believed representative of implementation costs. Labor hours are included below for implementation of each of the 4 parts of the SIR.

Installation procedures for improved-design inflatable seals are assumed to not change as a result of SIR. It is assumed the SIR will not change the installation time required for inflatable seals.

A. Plants with single inflatable refueling cavity seals.

o Improved-design inflatable seal	=	\$ 13,000
o Labor for implementation		
(2 man-wk)(\$2270/man-wk)	=	\$ 4,500
I (improved-design seals)	=	\$ 17,500/plant

B. Plants with double inflatable refueling cavity seals
(installation occurs during extended outage)

o Cost of permanent seal	= \$ 50,000
o Labor for installation of permanent seal (3 man-wk) (\$2270/man-wk)	= 6,800
o Labor for Radiation monitor, QA, etc. (2 man-wk) (\$2270/man-wk)	= <u>2,270</u>
I (permanent seal)	= \$ 59,100/plant

C. Installation of coffer dam (installation occurs during extended outage)

o Materials	= \$ 10,000
o Labor for installation (4 man-wks) (\$2270/man-wk)	= 9,100
o Labor for radiation monitor, QA, etc. (2 man-wk) (\$2270/man-wk)	= <u>4,500</u>
I (coffer dam)	= \$17,500/plant

D. Development and implementation of emergency procedures

(4 man-wk) (\$2270/man-wk)	= \$ 9,100/plant
----------------------------	------------------

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

A. (36 plants) (\$17,500/plant)	= \$ 6.3E+5
B. (9 plants) (\$59,100/plant)	= \$ 5.3E+5
C. (5 plants) (\$23,600/plant)	= \$ 1.2E+5
D. (134 plants) (\$9,100/plant)	= <u>\$ 1.2E+6</u>
NI (total)	= \$ 2.5E+6

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

No additional labor is assumed to be required for SIR operation and maintenance of improved-design seals and the installed coffer dams. Permanent seals are assumed to require 1 man-wk/py for operation and maintenance activities. Additional utility labor (estimated to be 0.4 man-wk/py) is also assumed to be required for annual review and revision (if any) of the emergency procedures developed for refueling cavity seal failures.

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

Annual inspection of permanent seals: (1.0 man-wk/py) (\$2270/man-wk)	= \$ 2,270/py
Annual review of emergency guidelines: (0.4 man-wk/py) (\$2270/man-wk)	= \$ 900/py

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

A total of 9 plants, all operating PWRs, are assumed to require replacement of their double inflatable seals with permanent steel seals. All 134 plants, both operating and planned, are assumed to require annual reviews of their emergency guidelines related to this issue.

$$\begin{aligned} \text{Permanent seal :} & \quad (9 \text{ plants})(27.7 \text{ yrs})(\$2270/\text{py}) = \$5.7\text{E}+5 \\ \text{Emergency procedure:} & \quad (134 \text{ plants})(28.3 \text{ yrs})(\$900/\text{py}) = \underline{3.4\text{E}+6} \\ & \quad \bar{N}T\bar{I}_0 \text{ (total):} = \underline{\$4.0\text{E}+6} \end{aligned}$$

12. Total Industry Cost (S_T):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$6.5E+6	\$8.9E+6	\$4.1E+6

NRC Costs (steps 13 through 21)

13. NRC Resources for SIR Development:

The NRC has previously issued IE Bulletin 84-03, reviewed the responses, and developed a proposed SIR (see Minners 1984). In addition, NRC sponsored a research program to evaluate the effects of beyond-design-basis accidents in spent fuel pools (Generic Issue 82; see Sailor, et al., 1987). No additional NRC costs are assumed to be required for SIR development. Additional NRC implementation costs are needed for establishing design criteria for refueling cavity seals and related technical specifications for their use. These activities are assumed to require 2 man-wks/plant of NRC labor. This requirement applies to all 45 affected plants.

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

$$C_D = (2 \text{ man-wk/plant})(45 \text{ plants})(\$2270/\text{man-wk}) = \$2.04\text{E}+5$$

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

NRC labor to support SIR implementation depends upon the types of activities required at each plant. For plants requiring installation of permanent seals and/or coffer dams, NRC resources are estimated at 2 man-wk/plant to review the designs and monitor their installation. For plants that require installation of improved-design inflatable seals, NRC resources are estimated at 0.4 man-wks to review the designs and monitor installation activities. It is assumed that 0.4 man-wks of NRC labor is also required to review revised emergency guidelines and ensure they are implemented.

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

$$\begin{aligned} \text{Improved Seals and} & \quad C = (2 \text{ man-wk/plant})(\$2270/\text{man-wk}) \\ \text{Emergency Guidelines} & \quad = \$4.5\text{E}+3/\text{plant} \end{aligned}$$

$$\begin{aligned}
 \text{Permanent seals and} & \quad C = (0.4 \text{ man-wk/plant})(\$2270/\text{man-wk}) \\
 \text{Coffer dams} & \\
 & = \$9.0\text{E}+2/\text{plant}
 \end{aligned}$$

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

$$\begin{aligned}
 \text{Permanent seals} & = (\$4.5\text{E}+3/\text{plant})(9 \text{ plants}) = \$4.1\text{E}+4 \\
 \text{Coffer dams} & = (\$4.5\text{E}+3/\text{plant})(5 \text{ plants}) = 2.3\text{E}+4 \\
 \text{Improved seals} & = (\$9.0\text{E}+2/\text{plant})(36 \text{ plants}) = 3.2\text{E}+4 \\
 \text{Emergency guidelines} & = (\$9.0\text{E}+2/\text{plant})(134 \text{ plants}) = \underline{1.2\text{E}+5} \\
 & \text{NC (total):} = \underline{\underline{\$2.2\text{E}+5}}
 \end{aligned}$$

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

It is estimated that NRC costs to monitor plant modifications and oversee the establishment of plant procedures will be 1 man-wk/py for affected plants requiring hardware modifications and 0.2 man-wks/py for plants requiring only modifications to emergency guidelines.

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

$$\begin{aligned}
 C_o \text{ (hardware)} & = (1 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$2270/\text{py} \\
 C_o \text{ (guidelines)} & = (0.2 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$ 450/\text{py}
 \end{aligned}$$

20. Total NRC Cost for Review of SIR Operation and Maintenance (NTC_o):

As discussed in Section 1.0, a total of 45 plants require either installation of permanent seals or improved-design inflatable seals. Five of these plants also require installation of a coffer dam. Of the 45 plants that require hardware modifications, 28 are operating PWRs, 13 are PWRs under construction, 4 are operating BWRs. The remaining 89 plants are assumed to require only modifications to emergency guidelines and procedures. Of these 89 plants, 9 are operating PWRs, 40 are planned PWRs, 20 are operating BWRs, and 20 are planned BWRs (based on NUREG/CR-2800, Appendix C). The total NRC cost for SIR operation and maintenance is:

o Plants requiring hardware modifications:

$$\begin{aligned}
 \text{PWRs: } (28 \text{ plants})(27.7 \text{ yrs})(\$2270/\text{py}) & = \$1.8\text{E}+6 \\
 (13 \text{ plants})(30.0 \text{ yrs})(\$2270/\text{py}) & = 8.9\text{E}+5 \\
 \text{BWRs: } (4 \text{ plants})(25.2 \text{ yrs})(\$2270/\text{py}) & = \underline{2.3\text{E}+5} \\
 \text{Subtotal} & = \underline{\underline{2.9\text{E}+6}}
 \end{aligned}$$

o Plants requiring emergency guideline revisions, only:

$$\begin{aligned}
 \text{PWRs: } (19 \text{ plants})(27.7 \text{ yrs})(\$450/\text{py}) & = \$2.4\text{E}+5 \\
 (30 \text{ plants})(30.0 \text{ yrs})(\$450/\text{py}) & = 4.1\text{E}+5 \\
 \text{BWRs: } (20 \text{ plants})(25.2 \text{ yrs})(\$450/\text{py}) & = 2.3\text{E}+5 \\
 (20 \text{ plants})(30.0 \text{ yrs})(\$450/\text{py}) & = \underline{2.7\text{E}+5} \\
 \text{Subtotal} & = \underline{\underline{\$1.2\text{E}+6}}
 \end{aligned}$$

$$\text{NTC}_o = \$ (2.9\text{E}+6 + 1.2\text{E}+6) = \$4.1\text{E}+6$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$4.5E+6	\$6.6E+6	\$2.4E+6

REFERENCES (for Issue 137)

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 138, Deinerting Upon Discovery of Reactor Coolant System Leakage

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: Current NRC guidance permits plants with inerted containment atmospheres (i.e., BWRs with Mark I and II containments) to deinert prior to shutdown, for short periods while operating, and for short periods while an ECCS loop is inoperable. During the time when a plant is deinerted, if a LOCA occurs, it is vulnerable to containment failure as a result of a hydrogen burn or explosion. This containment failure mode has not been evaluated in detail BWR Mark I and II plants primarily because of the inert atmosphere provided in their designs. Consequently, the risk associated with operating these plants while the containment is deinerted has not been evaluated in the past. The resolution to this issue involves revising NRC Technical Specification guidance to preclude operation of these plants while the containment is deinerted.

AFFECTED PLANTS: BWR: Operating = 29; Planned = 4

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 8.9E+03

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	27

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	1.4
SIR Operation/Maintenance =	260
Total of Above =	260
Accident Avoidance =	2.3

NRC COSTS:

SIR Development =	0.86
SIR Implementation =	(included above)
SIR Operation/Maintenance Review =	1.9
Total of Above =	2.8

DEINERTING UPON DISCOVERY OF REACTOR COOLANT SYSTEM LEAKAGE

ISSUE 138

1.0 SAFETY ISSUE DESCRIPTION

Technical Specifications currently permit plants with inert containments to deinert prior to shutdown and for short periods while operating. A relatively common practice is to deinert when a leak is discovered in the reactor coolant system (RCS). Deinerting, which is considered necessary for the protection of workers during containment entries, is done to allow workers to locate and possibly stop the leaks before plant shutdown is required by Technical Specifications (i.e., exit a limiting condition for operation). Typically, plants reduce power but do not shut down for this process.

Inerting in BWR Mark I and II plants is intended to mitigate against the consequences of hydrogen evolution during a loss-of-coolant accident (LOCA). Inerting maintains the oxygen concentration in the drywell and suppression chamber below the concentration that would allow a hydrogen explosion to occur. Deinerting for short periods of time while operating is currently allowed based on the assumption that there is a probabilistically small LOCA risk associated with the short time that a plant would be in a deinerted condition while at power. However, according to the leak before break concept, there is a high probability that a break resulting in a LOCA would be preceded by a RCS leak. As a result, the detection of an RCS leak may indicate the possibility of a subsequent LOCA and thus the need for an inert containment atmosphere. A deinerted containment would be susceptible to a hydrogen explosion following a LOCA.

Technical Specifications also allow plants to operate at power with a deinerted containment and inoperable Emergency Core Cooling Systems (ECCS). The increased risk associated with inoperable ECCS and simultaneous deinerted containment atmosphere is also addressed in this issue.

ISSUE RESOLUTION

The proposed resolution of this issue is for NRC to establish revised guidance regarding Technical Specifications that allow deinerting of BWR Mark I and II plants while operating. The revised guidance would prevent plants from operating while their containment atmospheres are deinerted and would prevent plants from operating in a deinerted condition with an inoperable ECCS.

AFFECTED PLANTS

BWRs with Mark I and II containments. A list of these plants is presented below:

MARK I

Pilgrim - 1*
Dresden - 2* and 3*
Quad Cities - 1* and 2*
Fermi - 2
Hatch - 1* and 2*
Arnold*
Oyster Creek*
Cooper*
Nine Mile Point - 1*
Millstone - 1*
Monticello*
Peach Bottom - 2* and 3*
Fitzpatrick*
Hope Creek - 1*
Browns Ferry - 1*, 2* and 3*
Vermont Yankee*
WNP-2*

MARK II

Brunswick - 1* and 2*
LaSalle County - 1* and 2*
Shoreham
Nine Mile Point - 2
Susquehanna - 1* and 2*
Limerick - 1* and 2

* Operating Plants

The total number of affected plants is 33. Of these 33 plants, 29 are currently operating and 4 are under construction.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this issue are estimated in this section and summarized in Tables 1 and 2. Attachment 1 describes the basis for the risk and dose changes that are estimated in the Tables.

TABLE 1 Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issues:

Deinerting Upon Discovery of Reactor Coolant System Leakage (138)

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

The resolution affects only BWR Mark I and II plants with inerted containments. Mark I and II containments are associated with a total of 33 plants, as shown previously. Of these 33 plants, 29 are currently operating and 4 are under construction.

	<u>N</u>	<u>$\bar{T}$ (yr)</u>
BWR: Operating	29	25.2
Planned	4	30.0
Total	33	26.6

3. Plants Selected for Analysis:

Millstone - 1; representative BWR.

4. Parameters Affected by SIR:

For the analysis of precluding deinerting during plant startup and shutdown, an additional containment failure mode, denoted as "HB" will be used to represent overpressure due to hydrogen burning. For precluding deinerting while a train of the ECCS is inoperable, a new accident sequence is developed and analyzed. Further details on both parts of this issue are presented in Attachment 1.

5. Base-Case Values for Affected Parameters:

See Attachment 1.

6. Affected Accident Sequences and Base-Case Frequencies:

Precluding Deinerting During Plant Startup and Shutdown

<u>Sequence</u>		<u>Frequency (1/py)</u>
T_4 JCFEG	α (BWR-1)	4E-7
	γ' (BWR-3)	4E-6
	γ (BWR-4)	4E-5
	HB (BWR-3)	2E-7

Precluding Deinerting During Plant Startup and Shutdown (Cont.)

<u>Sequence</u>		<u>Frequency (1/py)</u>	
T ₄ KCEFG	α	(BWR-1)	3E-7
	γ'	(BWR-3)	3E-6
	γ	(BWR-4)	3E-5
	HB	(BWR-3)	1E-7
T ₄ LCEFG	α	(BWR-1)	1E-7
	γ'	(BWR-3)	1E-6
	γ	(BWR-4)	9E-6
	HB	(BWR-3)	4E-8
(SB)B	α	(BWR-1)	3E-10
	γ'	(BWR-2)	3E-7
	γ	(BWR-3)	3E-6
	HB	(BWR-3)	1E-8
T ₄ JCEMG	α	(BWR-1)	2E-8
	γ'	(BWR-2)	2E-7
	γ	(BWR-3)	2E-6
	HB	(BWR-3)	8E-9
T ₄ LCEMG	α	(BWR-1)	2E-8
	γ'	(BWR-2)	2E-7
	γ	(BWR-3)	2E-6
	HB	(BWR-3)	8E-9
T ₄ KCEF	α	(BWR-1)	1E-8
	γ'	(BWR-3)	1E-7
	γ	(BWR-4)	9E-7
	HB	(BWR-3)	4E-9

Preclude Deinerting While ECCS is Inoperable

HB-ECCS HB (BWR-2) 1.2E-6 to 1.2E-7

7. Affected Release Categories and Base-Case Frequencies

Precluding Deinerting During Plant Startup and Shutdown

<u>Category</u>	<u>Frequency (1/py)</u>
BWR-1	8.5E-7
BWR-2	7.0E-7
BWR-3	1.6E-5
BWR-4	8.0E-5

Preclude Deinerting While ECCS is Inoperable

BWR-2 1.2E-6 to 1.2E-7

8. Base-Case, Affected Core-Melt Frequency (F):

$$F = 9.71\text{E-}5/\text{py} + \begin{cases} 1.2\text{E-}7/\text{py} \\ 1.2\text{E-}6/\text{py} \end{cases} = 9.72\text{E-}5 \text{ to } 9.83\text{E-}5/\text{py}$$

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

$$W = 137.4 + \begin{cases} 0.85 \text{ man-rem/py} \\ 8.5 \text{ man-rem/py} \end{cases} = 138.3 \text{ to } 145.9 \text{ man-rem/py}$$

10. Adjusted-Case Values for Affected Parameters

The probability of containment failure due to hydrogen burning (HB) during plant startup and shutdown is assumed to be zero if the resolution is implemented. This is assumed to apply to both LOCA and transient-initiated core melt sequences. The new sequence involving deinerting while a train of ECCS is inoperable is also effectively eliminated by the SIR.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Precluding Deinerting During Plant Startup and Shutdown

<u>Sequence</u>			<u>Frequency (1/py)</u>
T ₄ JCFEG	α	(BWR-1)	4E-7
	γ'	(BWR-3)	4E-6
	γ	(BWR-4)	4E-5
T ₄ KCEFG	α	(BWR-1)	3E-7
	γ'	(BWR-3)	3E-6
	γ	(BWR-4)	3E-5
T ₄ LCEFG	α	(BWR-1)	1E-7
	γ'	(BWR-3)	1E-6
	γ	(BWR-4)	9E-6
(SB)B	α	(BWR-1)	3E-10
	γ'	(BWR-2)	3E-7
	γ	(BWR-3)	3E-6
T ₄ JCEMG	α	(BWR-1)	2E-8
	γ'	(BWR-2)	2E-7
	γ	(BWR-3)	2E-6
T ₄ LCEMG	α	(BWR-1)	2E-8
	γ'	(BWR-2)	2E-7
	γ	(BWR-3)	2E-6
T ₄ KCEF	α	(BWR-1)	1E-8
	γ'	(BWR-3)	1E-7
	γ	(BWR-4)	9E-7

Preclude Deinerting While ECCS is Inoperable

HB-ECCS HB (BWR-2) 0

12. Affected Release Categories and Adjusted-Case Frequencies:

Precluding Deinerting During Plant Startup and Shutdown

Category Frequency (1/py)

BWR-1	8.5E-7
BWR-2	7.0E-7
BWR-3	1.5E-5
BWR-4	8.0E-5

Precluding Deinerting While ECCS is Inoperable

BWR-2 0

13. Adjusted-Case, Affected Cored Melt Frequency (F*):

$$F^* = 9.67E-5/\text{py}$$

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

$$W^* = 135.4 \text{ man-rem/py}$$

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

$$\Delta\bar{F} = 5.2E-7/\text{py to } 1.6E-6/\text{py}$$

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

$$\Delta W = 2.9 \text{ man-rem/py to } 10.5 \text{ man-rem/py}$$

17. Total Public Risk Reduction, (ΔW)_{Total}:

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds</u> (man-rem) <u>Upper Lower</u>
2.3E+03 to 8.9E+03	3.7E+06 0

ATTACHMENT 1 (To Table 1)

Risk assessments on Mark I and Mark II BWRs have developed information that indicates that hydrogen explosions are not credible containment failure modes for plants with inerted containments (see NRC 1975). This failure mode is not included in the analysis of these plants. As a result, this issue will be analyzed by assuming that hydrogen explosions at these plants are credible only during the time when the containment atmosphere is deinerted. In essence, this assumption creates an additional containment failure mode for BWR Mark I and II plants.

Resolution of this issue involves two parts. The first part is to revise plant technical specifications to preclude deinerting at all times during plant startup and shutdown. The second part of the resolution is to preclude plant operation with a deinerted containment during the time that a train of the ECCS is inoperable. The public risk reduction associated with these changes to reactor operations is calculated below.

Preclude Deinerting During Plant Startup and Shutdown

The probability of the hydrogen burn containment failure mode was approximated based on the fraction of a plant-year (py) that a plant could be in a deinerted condition at power. According to standard General Electric (GE) BWR/4 plant Technical Specifications, the suppression chamber and drywell atmosphere must be inerted during the time period:

- (a) within 24 hrs after thermal power is greater than 15% of rated power after startup
- (b) within 24 hrs prior to reducing thermal power to less than 15% of rated power, preliminary to a scheduled reactor shutdown.

Thus, a deinerted condition at power could occur for up to 24 hours in accordance with this Technical Specification. Assuming that an individual plant operates at power for, on average, 70% of a year, the plant operates for about 6,100 hours/year. Thus, the fraction of a year that a deinerted condition can exist is approximately $4E-3$. This assumes conservatively that one deinerting activity occurs per plant-yr. Alternatively, this assumption can be stated that a deinerted condition exists during startup operations at a rate of once every two years and during shutdown once every two years for a combined total of one deinerting activity per year. The fraction $4E-3$ will be used as the containment failure probability for this failure mode. According to Hatch, et al. (1981), releases of this type correspond to BWR release category 3. The model plant to be used in this analysis will be Millstone-1 (SAI 1983) because Grand Gulf, the BWR used most frequently in prioritization analyses, is a plant that does not use an inert atmosphere.

Other plant-specific PRAs that modeled Mark I and II containment performance during severe core damage scenarios were reviewed to place the $4E-3$ containment failure probability in perspective. Of all the Mark I and II PRAs reviewed, only the Limerick PRA (Philadelphia Electric Co. 1981) assessed the probability of hydrogen burn or detonation. It was estimated that the probability of the plant not being inerted while operating at power was 0.01. It was further

estimated that if a core melt accident occurs during this time, then the probability of a burn or detonation that is sufficient to cause direct overpressure release is no larger than 0.1. This leads to a containment failure probability of approximately $(0.01) \times (0.1) = 0.001$. Thus, the containment failure probability calculated above (0.004) compares relatively well with the data in the Limerick PRA.

The core melt sequences that are affected by safety issue resolution include all LOCA sequences plus all transient-initiated sequences that include the unavailability of the ECCS. Resolution of this issue would eliminate hydrogen explosions as a potential containment failure mode for the affected plants. Since operating for a short period during deinerted conditions is currently allowed, the base-case core-melt frequency and public risk will be estimated as if the hydrogen explosion failure mode were added to the Millstone-1 PRA (SAI 1983). Thus, the base-case accident sequences and frequencies include the contributions due to this containment failure mode. The adjusted-case will consist of the results of the Millstone-1 PRA without including the hydrogen explosion failure mode. The affected core melt sequences, their frequencies, containment failure modes, and BWR release categories are shown in Table 1.

Preclude Deinerting While ECCS Train is Inoperable

The second part of this issue involves precluding the condition in which the reactor is deinerted while one train of the Low Pressure Coolant Injection (LPCI) system is inoperable. A new accident sequence is postulated involving the following events: (1) discovery that one train of LPCI is inoperable; (2) the inoperable LPCI train is not restored to an operable condition and the plant is required to shutdown; (3) operator deinerts to get a head start on the unscheduled outage; and (4) a small-break LOCA. It is assumed that if these four events occur, a core-melt accident will occur followed by a hydrogen burn or detonation in the containment sufficient to cause a direct overpressure release.

The first event in the above sequence is discovery that a loop of the LPCI system is inoperable. No data could be found on the frequency of LPCI failures so the system was modeled assuming that LPCI downtime is dominated by the unavailabilities of 4 motor-driven pumps and 16 MOVs. These components are tested at 30 day (720 hr) intervals or approximately 12 tests/yr. The probability of failure to start for motor-driven pumps normally in a standby condition (0.01/demand) and failure to open for normally-closed MOVs (0.02/demand) were taken from the NUCLARR data base (NRC 1988, Volume 5, Part 3). Therefore, the probability that a component of the LPCI system is found to be inoperable during monthly testing was estimated at:

$$p(\text{LPCI pump}) = (4 \text{ pumps})(0.01/\text{Demand})(12/\text{yr}) = 0.48/\text{yr}$$

$$p(\text{LPCI MOV}) = (16 \text{ MOVs})(0.02/\text{Demand})(12/\text{yr}) = 3.8/\text{yr}$$

The Millstone-1 PRA (SAI 1983) states that reactor operation is permitted for up to 7 days when 1 component in the LPCI and Essential Service Water (ESW) systems is inoperable. Therefore, plant operators have approximately 6 days (144 hrs) to either restore the component to an operable status or make the decision to begin to shut down the plant. The probability that a pump is not repaired within 6 days was estimated at 0.1 and for MOVs was estimated at 0.02 (NRC 1975).

The frequency of discovering that a LPCI loop is down and not restoring the loop to operable status in sufficient time to avoid a plant shutdown is therefore:

$$p(\text{shutdown} \backslash \text{LPCI failure}) = (0.48/\text{yr})(0.1) + (3.8/\text{yr})(0.02) \\ = 0.124$$

The probability that an operator decides to deinert the containment to get a head start on the unscheduled outage was assumed to be in the range of 0.1 to 0.01. No data exists to confirm these values. Using these values, the annual frequency of the condition in which a plant is deinerted and has an inoperable LPCI system is therefore 0.012 to 0.0012/ry.

The final event in this sequence is a small-break LOCA in which the LPCI system is required to maintain core inventory. Given that the prior events have effectively disabled the LPCI system, the occurrence of a LOCA is anticipated to result in a core melt. Therefore, the frequency of a LOCA (about $1\text{E}-3/\text{ry}$) was multiplied by the frequency of deinerting given LPCI failure to obtain the total core-melt frequency for this accident sequence. The core-melt frequency was estimated as follows:

$$F(\text{HB-ECCS}) = (0.012)(1\text{E}-3) \text{ to } (0.0012)(1\text{E}-3) \\ = 1.2\text{E}-5 \text{ to } 1.2\text{E}-6/\text{ry}$$

The containment failure probability was taken from the Limerick PRA (Philadelphia Electric Co. 1981). It was stated that, if a core melt occurs during the time when the containment is deinerted, the probability of a burn or detonation that is sufficient to result in a direct overpressure release is 0.1. This corresponds to a BWR release category 2 release. As given in NUREG/CR-2800 (Andrews et al. 1983), the consequences of a BWR-2 release are estimated at $7.1\text{E}+6$ person-rem. Thus, the public risk associated with this accident sequence is estimated at:

$$W(\text{upper}) = (1.2\text{E}-5/\text{ry})(0.1)(7.1\text{E}+6 \text{ person-rem}) = 8.5 \text{ person-rem/ry}$$

$$W(\text{lower}) = (1.2\text{E}-6/\text{ry})(0.1)(7.1\text{E}+6 \text{ person-rem}) = 0.85 \text{ person-rem/ry}$$

The proposed resolution would effectively eliminate operation of the plant in a deinerted condition while the LPCI system is inoperable. Thus, the reductions in core-melt frequency and public risk associated with this part of the issue resolution are:

$$\Delta F = 1.2\text{E}-6 \text{ to } 1.2\text{E}-7/\text{ry}$$

$$\Delta W = 0.9 \text{ to } 9 \text{ person-rem/ry}$$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Deinerting upon Discovery of Reactor Coolant System Leakage (138)

2. Affected Plants (N):

The resolution of this issue BWRs with Mark I and Mark II containments. A total of 33 plants are affected; of these, 29 are currently operating and 4 are under construction.

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

	<u>$\bar{T}$ (yr)</u>
Operating BWRs	25.2
Planned	30.0
All	26.6

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

$$\begin{aligned}\Delta\bar{F}D_R &= (2.0E+04 \text{ man rem}) \times \begin{matrix} 1.2E-7/\text{py} \\ 1.2E-6/\text{py} \end{matrix} \\ &= 1.0E-2 \text{ to } 3.2E-2 \text{ man-rem/py}\end{aligned}$$

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

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
8.5 to 27	1.0E+4	0

6-12. Steps Related to Occupation Doses for SIR Implementation and Operation/Maintenance:

No change in occupational doses will result from issue resolution. Plant workers will need to inspect and repair (if needed) the systems within containment, regardless of the guidance on deinerting the containment atmosphere while operating. Issue resolution primarily affects the timing and plant status when the inspection/repair activities can be performed. No additional work in radiation zones would be required for SIR implementation, operation, or maintenance.

3.0 SAFETY ISSUE COSTS

The resolution of this issue involves development of revised NRC guidance on deinerting the containment atmospheres of BWR Mark I and II plants for short periods while operating. The industry and NRC costs for implementation of the two revised technical specifications (e.g., NRC development and industry review of model technical specifications, preparing license amendment, and revising operating procedures) associated with SIR are taken from NUREG/CR-4627 (NRC 1986). Training in the revised technical specifications will also be required for plant operators. The issue also affects plant availability because the plants would be required by the revised guidance to begin to shut down after discovering RCS leakage and would not be permitted to deinert after discovering that a train of ECCS is inoperable. This would require the utilities to purchase replacement power during the time when the plant is shut down. Industry and NRC costs are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue

Deinerting Upon Discovery of Reactor Coolant System Leakage (138)

2. Affected Plants (N):

The resolution of this issue affects BWRs with Mark I and II containments. A total of 33 plants are affected; of these, 29 are currently operating and 4 are under construction.

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

	<u>$\bar{T}$ (yr)</u>
Operating BWRs	25.2
Planned	30.0
All	26.6

Industry Costs (Steps 4 through 12)

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

$$\Delta(\bar{F}A) = (\$1.65E+09) \times \frac{5.2E-7/\text{py}}{1.6E-6/\text{py}} = \$860 \text{ to } 2640/\text{py}$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$7.3E+05 to 2.3E+06	8.3E+08	0

6. Per-Plant Industry Resources for SIR Implementation:

SIR involves two changes to plant technical specifications that are assumed to be uncomplicated and uncontroversial. NRC (1986) estimates the costs for such changes at \$16,000 per change; for two changes, the costs are \$32,000. It is estimated that 1 man-month/plant of utility labor is required to train operators in this relatively uncomplicated change. No costs for equipment or replacement power are required for implementation of the resolution of this issue.

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

For all plants:

$$\begin{aligned} I &= \$32,000 + (4 \text{ man-wks/plant}) (\$2,270/\text{man-wk}) \\ &= \$41,000/\text{plant} \end{aligned}$$

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

$$\begin{aligned} NI &= (33 \text{ plants}) (\$41,000/\text{plant}) \\ &= \$1.4\text{E}+06 \end{aligned}$$

9-10. Steps Related to Per-Plant Industry Costs for SIR Operation and Maintenance

Industry costs for SIR operation and maintenance consist of replacement power costs. Replacement power will be needed when plants shut down to deinert upon discovery of RCS leakage and while ECCS systems are inoperable, rather than the present practice of deinerting for short periods while at power. Replacement power costs were calculated based on the assumption that a deinerted condition during plant startup or shutdown at BWR Mark I and II plants occurs once per plant-year and while ECCS systems are inoperable at a rate of 0.012/py (see attachment 1 to Table 1). Both deinerting conditions are assumed to be of 24 hr duration. Thus, the per-plant industry cost for SIR operation and maintenance (I_o) is:

$$\begin{aligned} I_o &= [(1.0/\text{py} \times 24 \text{ hr}) + (0.012/\text{py} \times 24 \text{ hr})] (\$300,000/\text{day}) (24 \text{ hr}/\text{day}) \\ &= \$3.04\text{E}+05/\text{py} \end{aligned}$$

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

$$\begin{aligned} NTI_o &= (29 \text{ plants}) (25.2 \text{ yrs}) (\$3.04\text{E}+05/\text{py}) \\ &\quad + (4 \text{ plants}) (30.0 \text{ yrs}) (\$3.04\text{E}+05/\text{py}) \\ &= \$2.6\text{E}+08 \end{aligned}$$

12. Total Industry Cost (S_t):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
2.6E+08	3.9E+08	1.3E+08

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development and Support of SIR Implementation:

NRC costs for SIR development and support of SIR implementation include costs for development of model technical specifications, issuance of a generic letter, review and approval of proposed license amendments, and preparation of final license amendment. The costs for these NRC activities are taken directly from NUREG/CR-4627 (NRC 1988) and amount to \$13,000/change per plant. Two technical specification changes are required for SIR so the total NRC development costs are \$26,000/plant.

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

$$\begin{aligned} C_D + NC &= (\$26,000/\text{plant})(33 \text{ plants}) \\ &= \$8.6E+05 \end{aligned}$$

15-17. Per-Plant NRC Labor and Costs for Support of SIR Implementation:

Estimated in Steps 13 and 14.

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

It is estimated that total NRC labor for SIR operation and maintenance is 1 man-wk/py for a periodic review of the implementation of the technical specification changes. No additional NRC costs are identified for Support of SIR implementation or review of SIR operation and maintenance.

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

$$C_o = (1 \text{ man-wk/py}) (\$2,270/\text{man-wk}) = \$2.3E+03/\text{py}$$

20. Total NRC Cost for Review of SIR Operation and Maintenance ($\overline{N\overline{TC}_o}$):

$$\begin{aligned} \overline{N\overline{TC}_o} &= \$2.3E+3/\text{py} \times [(29 \text{ plants})(25.2\text{yr}) + (3 \text{ plants})(30 \text{ yrs})] \\ &= \$1.9E+06 \end{aligned}$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$2.8E+06	\$3.8E+06	\$1.8E+06

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 142, Leakage Through Electrical Isolators in Instrumentation Circuits

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: Electronic isolators are used to maintain electrical separation between safety and non-safety-related electrical systems in nuclear power plants. These isolators are designed and tested to prevent the maximum credible voltage or current transient on the non-Class 1E (non-safety) side of the isolator from degrading the performance of the Class 1E safety-related circuit below a certain acceptable level. The safety issue arises from the recent observation that for electrical transients below the maximum credible level, some electrical energy may leak through the isolator and potentially degrade the performance of a safety function. The proposed resolution involves replacement of certain types of isolators with electrical components that are not susceptible to this type of fault (e.g., surge arresters, filter chokes, capacitors, and power conditioners).

AFFECTED PLANTS: All Plants (PWRs - 90; BWRs - 44)

RISK/DOSE RESULTS (man-rem):

	<u>Upper Bound</u>	<u>Best Estimate</u>	<u>Lower Bound</u>
PUBLIC RISK REDUCTION =	3.6E+6	2.1E+5	5.0E+4
OCCUPATIONAL DOSES:			
SIR Implementation =		1.5E+3	
SIR Operation/Maintenance =		1.5E+4	
Total of Above =		1.7E+4	
Accident Avoidance =	2.5E+4	1.5E+3	3.3E+2

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =		25	
SIR Operation/Maintenance =		94	
Total of Above =		120	
Accident Avoidance	2.1E+3	130	27

NRC COSTS:

SIR Development =		0.4	
SIR Implementation =		1.2	
SIR Operation/Maintenance			
Review =		4.3	
Total of Above =		5.9	

LEAKAGE THROUGH ELECTRICAL ISOLATORS IN INSTRUMENTATION CIRCUITS

ISSUE 142

1.0 SAFETY ISSUE DESCRIPTION

Electronic isolators are used to maintain electrical separation between safety and non-safety-related electrical systems in nuclear power plants, preventing malfunctions in the non-safety systems from degrading performance of safety-related circuits. Isolators are primarily used where signals from Class-1E safety-related systems are transmitted to non-Class 1E control or display equipment. The potential for electrical energy leakage through electronic isolation devices has been identified as a possible safety issue (Sheron 1987).

There are a number of devices which may qualify as electrical isolators in a nuclear power plant, including fiber optic and photo-electric couplers, transformer modulated isolators, current transformers, amplifiers, circuit breakers, and relays. These isolators are designed and tested to prevent the maximum credible voltage or current transient on the non-Class 1E side of the isolator from degrading the performance of the safety-related circuit below a certain acceptable level.

The potential safety issue arises from the recent observation (Nielsen 1986) that for electrical transients below the maximum credible level, a relatively high level of noise may pass through certain types of isolation devices and be transmitted to safety-related circuitry. In some cases, the amount of energy that can pass through the isolator may be sufficient to damage or seriously degrade the performance of Class 1E components, while in other cases, electrical noise on the circuit may cause the isolation device to give a false output.

Modulated transformer-type analog isolators are used extensively in the domestic nuclear power industry. Unfortunately, in the presence of power surges and radio-frequency noise (RF), this class of isolation device offers relatively poor protection. These devices have been found to allow electrical noise in excess of 200 W in some cases to pass between supposedly isolated terminals (Nielsen 1986). Electrical transients of this magnitude can be sufficient to damage or cause malfunction of safety system components unless additional protection has been provided on the Class 1E circuit.

The digital optically-coupled isolator has been demonstrated to be susceptible to false actuation in the presence of radio-frequency noise (Nielsen 1986). It is conceivable that an isolation device of this type, operating in the presence of significant electromagnetic interference, may cause an instrument to give false readings or cause a control system to take incorrect action. When this type of isolator is used to connect instruments from the engineered safety function (ESF) system to the safety-related circuit, a spurious actuation may result in a component of the system being either unavailable when needed or being activated inappropriately. In one recorded incident, a voltage transient induced by a power line fault caused a false indication that the turbo-generator output breaker had tripped, which resulted in a reactor scram. While in this instance, other safety system components operated as designed to prevent a more serious

incident, a similar false isolator output elsewhere in the system might have resulted in more critical consequences.

Circuit breakers are the most commonly employed type of isolation device separating Class 1E from non-Class 1E systems. Circuit breakers used as isolation devices provide separation between circuits in the presence of a sustained fault exceeding the breaker's current rating. Subsequent to an isolation trip signal, the breaker trips to provide an open circuit between the safety and non-safety related circuits until the breaker is either automatically or manually reclosed. However, for faults below the maximum credible level, for high frequency noise, and for high voltage surges, the breaker may not trip before significant transient energy has reached the safety-related network. In some cases, breaker and relay contacts may be welded in unprotected and possibly compromised. Circuit breakers should, therefore, not be used as the sole means of separation between sensitive Class 1E and non-safety related circuits unless additional protection in the form of surge arresters and noise filters are provided. Such precautions are advisable unless it can be conclusively demonstrated that all affected Class 1E electrical components are otherwise protected or are sufficiently immune to any credible fluctuation in power.

Fuses employed as isolation devices function very much as the circuit breakers discussed above. A fuse will open in response to a sustained overcurrent level in the time required to explode the fuse filament and then to extinguish the resulting current arc. While a fuse will interrupt a sustained fault above a specified level, it will be ineffective against reduced current faults, RF, and high voltage impulses. As with circuit breakers, fuses should, therefore, not be used as the sole means of separation between sensitive Class 1E and non-safety related circuits unless additional protection is provided. As in the case of circuit breakers, fuses as isolation devices should not be employed alone unless it can be conclusively demonstrated that affected Class 1E electrical components are otherwise protected or are sufficiently immune to any potential fluctuation in power.

A number of topics must be considered to address the questions that arise from this issue. First, safety system components and their interactions must be studied to determine their vulnerability to electrical transients and compared to those discussed by Nielsen (1986). In the course of this preliminary investigation, through site visits at the WNP-2 nuclear power plant, it was found that varying levels of protection are in place on the safety-related electrical system to manage the transients that might arise in non-Class 1E electrical circuits. Any energy passing through isolation devices would tend to have varying degrees of influence on safety system components, depending on: (1) the amount and type of energy leakage; (2) the type of equipment connected to the circuit; (3) whether additional electrical protection was designed into the device; and (4) on the distance of the component from the "leaking" isolator. Class 1E power systems from representative PWR and BWR reactors should be thoroughly evaluated for adequacy of performance under transient conditions. This evaluation has not been done to date.

PROPOSED RESOLUTION

As discussed above, several major questions need to be resolved before meaningful actions can be taken to resolve this issue. The first is to determine the extent to which potentially susceptible isolators are in use in nuclear power

plants and in which systems they are in use. An Inspection/Enforcement Bulletin is proposed to be issued by the NRC to all power reactor licensees to provide input on these questions.

Assuming the NRC determines from the licensee responses to the IE Bulletin that a potential problem exists, a research program would be initiated to develop the final resolution to this issue. The research program would involve two major activities. The first activity would be to develop test procedures and acceptance criteria for isolators that the licensees could use to determine the adequacy of their installed isolators. The second activity would involve development of appropriate hardware fixes that could resolve the issue, including the safety and cost analyses necessary to determine the cost-effectiveness of the proposed hardware fixes.

Electrical hardware currently exists to either greatly reduce the amount of energy that may leak through the electrical barriers provided by various types of isolation device, or to minimize the consequences of any unwanted signals that may leak through the isolator. Some of these devices are described below.

Surge arresters, also called lightning arresters, provide an effective means of eliminating high voltage transients from a circuit. These devices are simply connected from the conductor directly to ground, preferably as close as possible to the device to be protected. The arresters function by simply shunting to ground any voltage spikes above a certain level.

Filter chokes and capacitors can greatly attenuate high frequency electrical noise. These components create an impedance to the passage of electrical energy proportionate to the frequency of the signal, and are especially effective against radio frequency noise. Filter chokes, or reactors, also function as current limiters in AC circuits, and thus offer additional protection from overload currents.

At power frequencies, power conditioners can be employed to eliminate all unwanted signals. Power line conditioners function by rectifying an AC signal into DC, and then reconvert power through an inverter into a clean, noise-free AC signal. These devices prevent notches, spikes, RF, brownouts, and overload power at the input terminals from degrading the quality of power at the protected output.

The next step in resolving this issue, assuming the NRC concludes that the safety issue resolution is cost-effective, is envisioned to consist of NRC issuance of a Generic Letter (GL) to the licensees that contains the following requirements:

- 1) inspection and testing of all electrical isolation devices between Class 1E and non-Class 1E systems,
- 2) repair/replacement of isolators that fail the tests, including description of acceptable hardware fixes to the isolators, and
- 3) implementation of an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems.

AFFECTED PLANTS

Resolution of this safety issue is assumed to affect all nuclear power plants that are in operation as well those under construction.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with this issue are estimated in this section. Because of major uncertainties in the extent of this issue and the potential effects on plant risk and safety, a different approach than that used in most Generic Safety Issue analyses is used to examine this issue. A "sensitivity" analysis is performed to bound the potential reduction in core-melt frequency and public risk. As shown in Tables 1 and 2, three sensitivity cases are evaluated, including lower bound, best estimate, and upper bound cases.

TABLE 1 Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Leakage Through Electrical Isolators in Instrumentation Circuits (142)

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

All PWRs and BWRs are assumed to be affected.

	<u>N</u>	<u>$\bar{T}$ (yr)</u>
PWRs	90	28.8
BWRs	44	27.4
All	134	28.3

3. Plants Selected for Analysis:

Representative PWR: Oconee 3

Representative BWR: Grand Gulf 1

4. Parameters Affected by SIR:

As discussed above, three sensitivity cases are examined in this analysis. The affected parameters are slightly different for each sensitivity case. Case 1, which represents a best estimate, affects all parameters that include; control circuitry failures, operator errors that could arise from misinterpretation of sensor readings or sensor errors, and functional failures of Engineered Safety Feature (ESF) actuation circuits. The lower bound case is assumed to affect only the control circuitry and ESF actuation failures. The upper bound case is assumed to affect all three types of parameters. See Attachment 1 for a list of the parameters affected by each sensitivity case.

5. Base-Case Values for Affected Parameters:

Failure of electronic isolators was not explicitly addressed in the Oconee and Grand Gulf PRAs. Consequently, the base-case values of the affected parameters were modified to model the effects of this issue. The modifications to the base-case values were performed by increasing the unavailabilities of the affected parameters by an amount equal to the

unavailability due to electronic isolator failure. Different amounts were added to each affected parameter to model the three sensitivity cases, as discussed in Attachment 1. The base-case values are listed in Attachment 1.

6. Affected Accident Sequences and Base-Case Frequencies:

The Oconee-3 and Grand Gulf mini-computer codes were used to estimate the change in core-melt frequency and public risk reduction. As such, the affected accident sequences and base-case frequencies were not calculated directly. These calculations are performed within the computer codes. However, it should be noted that all of the Oconee accident sequences were affected except for V, T₁(B₃)MLU, and T₂KMU. All Grand Gulf accident sequences were affected except for T₂₃C.

7. Affected Release Categories and Base-Case Frequencies:

As with the affected accident sequences, calculations involving the affected release categories and base-case frequencies are performed by the Oconee and Grand Gulf mini-computer codes.

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

Sensitivity Case	$\bar{F}^*_{PWR}$	$\bar{F}^*_{BWR}$
Best Estimate	6.57E-5/RY	3.70E-5/RY
Lower Bound	4.52E-5/RY	3.11E-5/RY
Upper Bound	4.65E-4/RY	1.46E-4/RY

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

Sensitivity Case	Man-rem	
	W^*_{PWR}	W^*_{BWR}
Best Estimate	145.9	246.3
Lower Bound	101.9	206.7
Upper Bound	1104.9	982.0

10. Adjusted-Case Values for Affected Parameters:

The adjusted-case (i.e., after implementation of issue resolution) is represented by the original parameter values from the Oconee and Grand Gulf PRAs, as discussed in Attachment 1.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

See Step 6.

12. Affected Release Categories and Adjusted-Case Frequencies:

See Step 7.

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

$$\bar{F}_{PWR} = 3.98E-5/RY$$

$$\bar{F}_{BWR} = 2.90E-5/RY$$

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

$$W_{PWR} = 88.9 \text{ man-rem}$$

$$W_{BWR} = 192.9 \text{ man-rem}$$

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

Sensitivity Case	$\Delta\bar{F}_{PWR}^*$	$\Delta\bar{F}_{BWR}^*$
Best Estimate	2.59E-5/RY	7.98E-6/RY
Lower Bound	5.37E-6/RY	2.07E-6/RY
Upper Bound	4.35E-4/RY	1.17E-4/RY

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

Sensitivity Case	Man-rem/RY	
	ΔW_{PWR}	ΔW_{BWR}
Best Estimate	57	53
Lower Bound	13	14
Upper Bound	1016	789

17. Total Public Risk Reduction, (ΔW)_{Total}:

<u>Sensitivity Case</u>	<u>Total Man-rem</u>
Best Estimate	2.12E+5
Lower Bound	5.03E+4
Upper Bound	3.58E+6

ATTACHMENT 1 (To Table 1)

There is a large uncertainty associated with this issue. The major sources of uncertainty include the extent to which potentially susceptible isolators are in use at nuclear power plants, the amount of electrical energy leakage through isolation devices that could compromise the function of Class 1E system components, and which components would such compromises be critical. While a recent study conducted by Nielsen (1986) strongly indicated that a safety problem may exist due to energy leakage through electronic devices, no definitive research has been conducted, to date, to indicate the character and magnitude of the associated safety issues. As a result, it is not possible to characterize the potential risk reduction using standard safety issue analysis format (see Andrews, et al., 1983).

A "sensitivity analysis" format that was intended to bound the potential public risk reduction associated with this issue was used to develop this preliminary analysis. Estimates of the upper and lower bounds were developed as well as a third case that represents the "best estimate" based on the available information.

The parameters affected by this issue are those involving control circuitry failures and functional failure of Engineered Safety Feature (ESF) actuation systems. These components may be directly affected by energy leakage through isolation devices that are intended to protect them from signals originating in connected non-Class 1E systems. It is also possible that sensors in the Class 1E safety system may be affected by the electrical energy leakage from the non-Class 1E system. These sensors may include valve position, temperature, and pressure sensors that alert plant operators to take a particular action. In this case, plant operators may be misled into taking an incorrect action or to not taking action at a time when an action is required. For this reason, operator errors are also included as potentially affected parameters.

The affected parameters were identified and modified to model the three sensitivity cases. The parameter modifications were performed as described below:

Best Estimate: All of the affected control circuitry failure, ESF actuation functional failure, and operator error terms were multiplied by a factor of two (assumed) to account for the potential additional failures associated with electrical isolators.

Upper Bound: All of the affected control circuitry failure, ESF actuation functional failure, and operator error terms were multiplied by a factor of ten (assumed) to account for the potential additional failures associated with electrical isolators.

Lower Bound: The control circuitry and ESF actuation functional failures were multiplied by a factor of 1.4. This factor is based on a 100% increase in the probability of fuse failures, which are included in the control circuitry unavailability values. There is assumed to be no effect on the operator error terms in this case.

After the failure terms were modified, they were combined with the remaining unaffected portions of the parameter unavailabilities to calculate the

revised unavailabilities. The affected cut set elements and their base-case and adjusted-case unavailabilities are shown on the next page. Note that the adjusted-case represents the original Oconee and Grand Gulf parameter values. This is because isolator failures were not considered as potential causes of failure in the original Oconee and Grand Gulf PRAs. Since the base-case is intended to represent the situation in which isolator failures are considered as possible causes of safety system failures and the adjusted-case represents the situation after the resolution is implemented, the modified parameter values are used in the base-case. For this reason, the base-case affected core-melt frequencies and public risk values shown in Steps 8 and 9 of Table 1, respectively, may be larger than the original values from the Oconee and Grand Gulf PRAs.

The base-case and adjusted-case values of the affected parameters were input to the Oconee and Grand Gulf mini-computer codes. The output from the codes includes the change in core-melt frequency and public risk reduction (See Table 1).

Base-Case and Adjusted-Case Values of Affected Parameters

<u>Parameter</u>	<u>Adjusted-^(a) Case</u>	<u>Base-^(b) Case 1</u>	<u>Base-^(c) Case 2</u>	<u>Base-^(d) Case 3</u>
<u>Grand Gulf</u>				
H	0.0212	0.0225	0.0217	0.0329
HACT, RACT	0.00123	0.00223	0.00163	0.0102
R	0.0512	0.0530	0.0518	0.067
L	0.0213	0.0226	0.0218	0.033
LRACT, BRACT	0.00123	0.00223	0.00163	0.0102
LA2, LB2	0.0140	0.0151	0.0144	0.0240
LB1	0.0134	0.0138	0.0135	0.017
LC	0.0215	0.0230	0.0220	0.035
VGA1, VGB1	0.0148	0.0156	0.0150	0.022
VGA2, VGB2	0.0236	0.0273	0.0238	0.0553
SA, SB	0.0144	0.0150	0.0146	0.0198
SAACC, SBACC	0.00123	0.00223	0.00163	0.0102
SSA, SSB	0.0205	0.0223	0.0209	0.0361
SSC	0.0140	0.0151	0.0144	0.0239
SAC, SBC, SCC	0.00123	0.00223	0.00163	0.0102
V1, V2	0.00803	0.0091	0.00813	0.0173
V3	0.0033	0.0064	0.0033	0.0296
SCVA, SCVB	0.0315	0.0333	0.0321	0.0477
<u>Oconee</u>				
B, C	0.0033	0.0043	0.0037	0.0121
D, E	0.0231	0.0354	0.0249	0.1334
CONST1	2.05E-4	4.83E-4	2.98E-4	7.00E-4
CONST2	6.30E-4	1.25E-3	8.27E-4	1.23E-2
A1, C1	0.0098	0.0163	0.0124	0.0683
B1	0.0349	0.0502	0.0410	0.1718
G1	0.0136	0.0172	0.0150	0.046
RCSRBCM	3.20E-5	6.40E-5	3.20E-5	3.20E-4
WXCM	0.003	0.006	0.003	0.03
D·E	4.94E-4	0.00121	5.80E-4	0.0178
W·X	8.78E-5	2.52E-4	1.44E-4	4.51E-3
B·W, C·X	2.73E-5	6.44E-5	4.07E-5	8.09E-4
D·X, E·W	2.09E-4	5.53E-4	2.91E-4	8.95E-3
B·D, E·C	6.27E-5	1.39E-4	7.89E-5	1.60E-3

-
- (a) Represents original Oconee and Grand Gulf PRA values.
 (b) Best estimate. See text for case description.
 (c) Lower bound case. See text for case description.
 (d) Upper bound case. See text for case description.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Leakage Through Electrical Isolators in Instrumentation Circuits (142)

2. Affected Plants (N):

All PWRs and BWRs are assumed to be affected.

	<u>N</u>
PWRs	90
BWRs	44
All	134

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

	<u>$\bar{T}$ (yr)</u>
PWRs	28.8
BWRs	27.4
All	28.3

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

PWR

Best Estimate = (19,860 person-rem)(2.59E-5/RY) = 0.51 person-rem/RY
 Lower Bound = (19,860 person-rem)(5.37E-6/RY) = 0.11 person-rem/RY
 Upper Bound = (19,860 person-rem)(4.35E-4/RY) = 8.64 person-rem/RY

BWR

Best Estimate = (19,860 person-rem)(7.89E-6/RY) = 0.16 person-rem/RY
 Lower Bound = (19,860 person-rem)(2.07E-6/RY) = 0.041 person-rem/RY
 Upper Bound = (19,860 person-rem)(1.17E-4/RY) = 2.32 person-rem/RY

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

<u>Sensitivity Case</u>	<u>Total Man-rem</u>
Best Estimate	1.5E+3
Lower Bound	3.3E+2
Upper Bound	2.5E+4

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

Implementation of SIR is assumed to include an initial response to an IE Bulletin, response to a Generic Letter, and repair, replacement, and testing of potentially susceptible isolators. Total utility labor in radiation zones for these activities is estimated at 34 man-days/plant for PWRs and 56 man-days/plant for BWRs. See Section 3.0 for further details.

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

Radiation fields of 25 mrem/hr are assumed to exist inside containment where most of the isolators are located. A 75% efficiency factor is used for labor in radiation zones.

$$D_{PWR} = (25 \text{ mrem/hr})(8 \text{ hrs/d})(34 \text{ d})/0.75 = 9.1 \text{ man-rem/plant}$$

$$D_{BWR} = (25 \text{ mrem/hr})(8 \text{ hrs/d})(56 \text{ d})/0.75 = 14.9 \text{ man-rem/plant}$$

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

$$\begin{aligned} ND &= (90 \text{ PWRs})(9.1 \text{ man-rem/plant}) + (44 \text{ BWRs})(14.9 \text{ man-rem/plant}) \\ &= 1.5E+3 \text{ man-rem} \end{aligned}$$

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

Annual test program:	PWRs - 10 man-days/Ry BWRs - 16 man-days/Ry
Replacement of remaining isolators	PWRs - 2.4 man-days/Ry BWRs - 4.2 man-days/Ry
Total:	PWRs - 12.4 man-days/Ry BWRs - 20.2 man-days/Ry

(See Section 3.0 for the basis for these estimates)

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

$$D_o \text{ (PWR)} = (25 \text{ mrem/hr})(8 \text{ hr/d})(12.4 \text{ d/Ry})/0.75$$

$$= 3.3 \text{ man-rem/Ry}$$

$$D_o \text{ (BWR)} = (25 \text{ mrem/hr})(8 \text{ hr/d})(20.2 \text{ d/Ry})/0.75$$

$$= 5.4 \text{ man-rem/Ry}$$

11. Total Occupational Dose Increase for SIR Operation/Maintenance, (NTD_o):

$$\begin{aligned} NTD_o &= (90 \text{ PWRs})(28.8 \text{ y})(3.3 \text{ man-rem/Ry}) \\ &\quad + (44 \text{ BWRs})(27.4 \text{ y})(5.4 \text{ man-rem/Ry}) \\ &= 1.5E+4 \text{ man-rem} \end{aligned}$$

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
16,500	49,000	5500

3.0 SAFETY ISSUE COSTS

The costs for SIR are difficult to quantify because of the uncertainty associated with this issue. This SIR involves several proposed activities, as outlined previously, that are designed to reduce the uncertainties over a period of time. The labor requirements and costs for each activity are discussed in this section.

The first activity involves NRC issuance of an IE Bulletin that is intended to help determine the extent to which potentially susceptible isolators are in use in nuclear power plants and in which systems they are in use. It is estimated that 2 man-wk of NRC labor is required to prepare the Bulletin and obtain management and legal concurrence. It is estimated that approximately 2 man-wk/plant of utility labor are required to develop responses to the Bulletin. The utility responses to the Bulletin must then be reviewed and analyzed by the NRC. A 6 man-month subcontract is estimated to be needed to perform this review and analysis.

The second activity, assuming the NRC concludes the issue warrants further attention, involves a research program would be initiated to develop the final resolution to this issue. The research program would involve two major activities. The first activity would be to develop test procedures and acceptance criteria for isolators that the licensees could use to determine the adequacy of their installed isolators. It is estimated that a \$50,000 subcontract, plus \$10,000 for an NRC Technical Monitor, will be needed to accomplish this activity. The second activity would involve development of appropriate hardware fixes that could resolve the issue, including the safety and cost analyses necessary to determine the cost-effectiveness of the proposed hardware fixes. A \$150,000 subcontract plus \$20,000 for an NRC Technical Monitor are estimated for this activity.

The next step is to prepare and issue a Generic Letter to all power reactor licensees. Approximately 4 man-wks of NRC labor are estimated to be required to prepare and obtain management and legal concurrence with the GL. It is assumed that each utility is required to respond formally to the GL. Therefore, 6 man-months of NRC labor are estimated to be required to review and evaluate each utilities' response to the GL.

The Generic Letter (GL) is assumed to contain the following requirements that are applicable to all utilities:

- (1) inspection and testing of all electrical isolation devices between Class 1E and non-Class 1E systems,
- (2) replacement of failed or unacceptable isolators, including descriptions of acceptable hardware fixes to the isolators, and
- (3) implementation of an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems.

The estimated industry labor requirements and costs for these activities are as follows.

The initial testing and inspection program is estimated to require approximately 4 man-wks/plant of industry labor for planning and 8 man-wks/plant for review and evaluation of the data, preparation of the formal response to the

GL, and preparation of a safety analysis. Industry labor to conduct the initial test program is highly uncertain because there are unknown numbers of affected systems and susceptible isolators at each plant. For this analysis, the number of potentially affected isolators was approximated using the number of safety system components in the Oconee and Grand Gulf PRAs with functional and/or control circuitry failure terms. The numbers of isolators was estimated at 46 for BWRs and 78 for PWRs. Assuming a two-man test team can test 10 isolators per day, labor requirements for the initial test/inspection required by the GL were estimated at 10 man-days/plant for PWRs and 16 man-days/plant for BWRs.

Isolators that fail the initial tests must be replaced or repaired. It is assumed that 25% of the tested isolators will fail the tests, which results in 12 failures of isolators at PWRs and 20 failures of isolators at BWRs. The estimated costs to purchase, install, test, and perform adequate Quality Control of acceptable replacement isolators is estimated at \$10,000/isolator. This estimate includes approximately 2 man-days of utility labor per isolator to be replaced. The total isolator replacement costs are estimated to be \$120,000/PWR and \$200,000/BWR.

The GL is proposed to include a requirement for annual testing and inspection of all electronic isolators. The industry labor requirements for this activity are estimated to be 1 man-wk/Ry for test planning (this is significantly lower than the 4 man-wk for planning the initial test program), plus 10 man-days/Ry to conduct the tests at PWRs and 16 man-days/Ry to conduct the tests at BWRs. An additional 1 man-wk/Ry at all plants to review the test results and prepare a report for the NRC is also included.

The annual testing program is likely to determine that there are additional failed or suspect isolators that require replacement. For this analysis, it is assumed that all the remaining isolators (i.e., other than those that were replaced as a result of the initial test program) will eventually be replaced with acceptable components. The number of remaining isolators to be replaced at PWRs is estimated at 38 (i.e., 46 - 12) over a 28.8 yr period or 1.2/Ry. At BWRs, the annual replacement rate is equal to 58 (i.e., 78 - 20) over a 27.4 yr period or 2.1/Ry. The annual replacement costs are thus estimated at \$12,000/Ry for PWRs and \$21,000/Ry for BWRs.

The NRC is assumed to review the implementation of the requirements in the GL. The NRC is also anticipated to review the test procedures, review plant-specific implementation plans, and prepare a safety evaluation. The NRC labor for this review is estimated to be 4 man-wk/plant. An additional 0.5 man-wks/Ry is estimated to be required for an annual review of the operation and maintenance of the SIR.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Leakage Through Electrical Isolators in Instrumentation Circuits (142)

2. Affected Plants (N):

All PWRs and BWRs are assumed to be affected.

	<u>N</u>
PWRs	90
BWRs	44
All	134

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

	<u>$\bar{T}$ (yr)</u>
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, ($\Delta\bar{F}$):

PWR

Best Estimate = $(\$1.65\text{E}+9)(2.59\text{E}-5/\text{RY}) = \$ 4.27\text{E}+4/\text{RY}$
 Lower Bound = $(\$1.65\text{E}+9)(5.37\text{E}-6/\text{RY}) = \$ 8.86\text{E}+3/\text{RY}$
 Upper Bound = $(\$1.65\text{E}+9)(4.35\text{E}-4/\text{RY}) = \$ 7.18\text{E}+5/\text{RY}$

BWR

Best Estimate = $(\$1.65\text{E}+9)(7.89\text{E}-6/\text{RY}) = \$ 1.32\text{E}+4/\text{RY}$
 Lower Bound = $(\$1.65\text{E}+9)(2.07\text{E}-6/\text{RY}) = \$ 3.42\text{E}+3/\text{RY}$
 Upper Bound = $(\$1.65\text{E}+9)(1.17\text{E}-4/\text{RY}) = \$ 1.93\text{E}+5/\text{RY}$

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

<u>Sensitivity Case</u>	<u>Total \$</u>
Best Estimate	1.3E+8
Lower Bound	2.7E+7
Upper Bound	2.1E+9

6. Per-Plant Industry Resources for SIR Implementation:

The industry labor requirements for responding to the IE Bulletin, planning and conducting the initial isolator test program are summarized below. See Section 3.0 for additional details.

Labor Requirements

Respond to IE Bulletin	.	.	.	.	2 man-wks
Plan initial test program	.	.	.	.	4 man-wks
Conduct tests (at 10 tests/day) - PWRs	.	.	.	.	10 man-days
- BWRs	.	.	.	.	16 man-days
Review and analyze test data, prepare report and safety analysis	.	.	.	.	8 man-wks

Hardware (25% failures assumed; \$10,000/replacement)

PWRs (12 replacements).	.	.	.	.	\$ 120,000
BWRs (20 replacements).	.	.	.	.	\$ 200,000

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

PWRs: (16 man-wks)(\$2270/man-wk) + \$ 120,000 = \$ 156,000/plant
 BWRs: (17.2 man-wks)(\$2270/man-wk) + \$ 200,000 = \$ 239,000/plant

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

NI = (90 PWRs)(\$156,000/plant) + (44 BWRs)(\$239,000/plant)
 = \$2.45E+7

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

Labor for Testing Program

Plan	.	.	.	.	1 man-wk/Ry
Conduct tests (see Step 6) - PWRs	.	.	.	.	10 man-days/Ry
- BWRs	.	.	.	.	16 man-days/Ry
Review/analyze test data	.	.	.	.	1 man-wk/Ry

Hardware Costs (at \$10,000/replacement)

PWRs (1.2 replacements/yr)	.	.	.	.	\$ 12,000/Ry
BWRs (2.1 replacements/yr)	.	.	.	.	\$ 21,000/Ry

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

I_o (PWR) = (4 man-wk/Ry)(\$2270/man-wk) + \$12,000 = \$ 21,100/Ry
 I_o (BWR) = (5.2 man-wk/Ry)(\$2270/man-wk) + \$21,000 = \$ 32,800/Ry

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

N $\bar{I}$ _o = (90 PWRs)(28.8 yr)(\$21,100/Ry) + (44 BWRs)(27.4 yr)(\$32,800/Ry)
 = \$9.42E+7

12. Total Industry Cost (S_T):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.2E+8	\$1.7E+8	\$4.9E+7

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

IE Bulletin

Prepare/issue bulletin	.	.	.	.	\$ 4,000
Review/analyze responses	.	.	.	.	50,000
TOTAL:					54,000

Develop test procedures/acceptance criteria

Subcontract	.	.	.	.	\$ 50,000
Technical monitor	.	.	.	.	10,000
TOTAL:					60,000

Develop resolution, safety/cost evaluation						
Subcontract	.	.	.	.	.	\$ 150,000
Technical monitor	.	.	.	.	.	<u>20,000</u>
TOTAL:						170,000

Generic Letter						
Prepare/issue GL	.	.	.	.	.	\$ 10,000
Analyze/document responses to GL						
Subcontract	.	.	.	.	.	55,000
Technical monitor	.	.	.	.	.	<u>10,000</u>
TOTAL:						75,000

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

$$C_D = \$ 50,000 + 60,000 + 170,000 + 75,000 \\ = \$ 355,000$$

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

It is estimated that 4 man-wks/plant of NRC labor is required to review plant-specific test procedures, review responses to the Generic Letter, review plant-specific implementation plans, and prepare safety evaluations (See Section 3.0).

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

$$C = (4 \text{ man-wks/plant})(\$2270/\text{man-wk}) = \$ 9080/\text{plant}$$

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

$$NC = (134 \text{ plants})(\$ 9080/\text{plant}) = \$ 1.22\text{E}+6$$

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

It is estimated that 0.5 man-wk/Ry of NRC labor are required for review of the annual isolator test/inspection results.

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

$$C_o = (0.5 \text{ man-wk/Ry})(\$2270/\text{man-wk}) = \$ 1135/\text{Ry}$$

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

$$\overline{N}TC_o = (\$1135/\text{Ry}) [(90 \text{ PWRs})(28.8 \text{ yr}) + (44 \text{ BWRs})(27.4 \text{ yr})] \\ = \$4.31\text{E}+6$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$5.9E+6	\$8.1E+6	\$3.6E+6

REFERENCES (For Issue 142)

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.

Nielsen, J. R. 1986. *Electronic Isolators Used in Safety Systems of U.S. Nuclear Power Plants*. NUREG/CR-3453. Prepared by EG&G Idaho, Inc. for the U. S. Nuclear Regulatory Commission, Washington D.C.

Sheron, B. W. "Proposed Generic Issue: Leakage Through Electrical Isolators in Instrumentation Circuits," Memorandum to B. Morris dated June 23, 1987. U.S. Nuclear Regulatory Commission, Washington D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 143, Availability of Chilled Water Systems and Room Cooling

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: In recent years, several nuclear power plants have experienced problems resulting from a partial or total loss of heating, ventilation, and air conditioning (HVAC) systems. Operability of some safety-related components is dependent upon operation of HVAC and chilled water systems to remove heat from the rooms containing the components. If chilled water and HVAC systems are unavailable to remove heat, the ability of the safety equipment within the rooms to operate as intended cannot be assured. The proposed resolution of this issue is for the NRC to issue guidance to licensees to prepare an engineering study and safety evaluation to identify these dependencies and propose procedural changes and hardware modifications to eliminate dependencies that are critical to the plant core damage frequency. The proposed procedural changes and hardware modifications would then be implemented by the licensees where appropriate.

AFFECTED PLANTS: All plants (BWRs - 44; PWRs - 90)

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION =	1.9E+5
OCCUPATIONAL DOSES:	
SIR Implementation =	2.2E+4
SIR Operation/Maintenance =	0
Total of Above =	2.2E+4
Accident Avoidance =	1.1E+3

COST RESULTS (\$10⁶):

INDUSTRY COSTS:	
SIR Implementation =	1.0E+8
SIR Operation/Maintenance =	0
Total of Above =	1.0E+8
Accident Avoidance =	9.3E+7
NRC COSTS:	
SIR Development =	6.3E+4
SIR Implementation =	3.0E+5
SIR Operation/Maintenance Review =	0
Total of Above =	3.7E+5

AVAILABILITY OF CHILLED WATER SYSTEMS AND ROOM COOLING

ISSUE 143

1.0 SAFETY ISSUE DESCRIPTION

In recent years, several nuclear power plants have experienced problems resulting from a partial or total loss of heating, ventilation, and air conditioning (HVAC) systems. Many of these problems exist for two reasons: (1) the desire to provide increased fire protection; and (2) advances in equipment control circuit designs. Since the Browns Ferry fire, considerable effort has been expended to improve the fire protection for equipment required for safe shutdown. Generally, this was accomplished by enclosing the equipment in small, isolated rooms. This has resulted in a significant increase in the importance of room cooling.

A second reason for the increase in the importance of room cooling is the advancement in control circuit design. With the introduction of electronic integrated circuits, plant control and safety have improved. These circuits, however, are more susceptible to damage as a result of severe changes in temperature. With these two facts in mind, the potential problems associated with HVAC systems were identified as a potential generic safety issue (Kerr 1987).

PROPOSED RESOLUTION

The proposed Safety Issue Resolution (SIR) consists of a re-evaluation of each plant's room heat capacity in order to identify areas in which a reduction in the dependence of equipment operability on HVAC and room cooling may be implemented. While the total elimination of this dependence may not be possible at all plants, this analysis will identify areas in which this dependence is critical. The determination of the critical dependencies and the ability to reduce it could be accomplished through the use of a plant-specific probabilistic risk assessment such as the Individual Plant Examinations (IPEs). After the critical dependencies are identified, each plant would implement procedural changes to eliminate or reduce the dependencies where possible. Hardware modifications may be needed for situations in which a procedure change cannot be implemented to reduce a critical dependency.

AFFECTED PLANTS

All plants are assumed to be affected by the proposed SIR.

2.0 SAFETY ISSUE RISK AND DOSE

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

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Availability of Chilled Water Systems and Room Cooling (143)

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

All PWRs and BWRs are assumed to be affected.

	<u>N</u>	<u>$\bar{T}$ (yr)</u>
PWRs	90	28.8
BWRs	44	27.4
All	134	28.3

3. Plants Selected for Analysis:

Representative PWR: Oconee 3
Representative BWR: Grand Gulf 1

4. Parameters Affected by SIR:

See Attachment 1.

5. Base-Case Values for Affected Parameters:

See Attachment 1.

6. Affected Accident Sequences and Base-Case Frequencies:

<u>Grand Gulf</u>	<u>Sequence</u>	<u>Frequency, RY^{-1}</u>
	TB	2.0E-5
	TBVW	1.1E-6
	TBUI	2.4E-6
	TBU1VW	9.5E-8
	TBU	9.4E-7
	TBVW	<u>2.1E-8</u>
	TOTAL:	<u>2.46E-5</u>

See Attachment 1 for further information.

7. Affected Release Categories and Base-Case Frequencies:

All base-case affected accident sequences were assigned to release category BWR-2 (see Andrews, et al., 1983).

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

$$\bar{F}_{PWR} = 5.47E-5/R\bar{Y}$$

$$\bar{F}_{BWR} = 2.46E-5/R\bar{Y}$$

The analysis was performed for BWRs and scaled for PWRs using the scaling relationships given by Andrews, et al. (1983). See Attachment 1.

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

$$W_{PWR} = 145.6 \text{ person-rem/R}\bar{Y}$$

$$W_{BWR} = 174.7 \text{ person-rem/R}\bar{Y}$$

10. Adjusted-Case Values for Affected Parameters:

See Attachment 1.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

<u>Grand Gulf</u>	<u>Sequence</u>	<u>Frequency, RY⁻¹</u>
	TB	1.35E-5
	TBVW	7.66E-7
	TBUI	1.36E-6
	TBU1VW	5.46E-8
	TBU	6.20E-7
	TBVW	<u>1.74E-8</u>
	TOTAL:	<u>1.66E-5</u>

See Attachment 1 for further information.

12. Affected Release Categories and Adjusted-Case Frequencies:

All adjusted-case affected accident sequences were assigned to release category BWR-2 (see Andrews, et al., 1983).

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

$$\bar{F}_{PWR}^* = 3.69E-5/R\bar{Y}$$

$$\bar{F}_{BWR}^* = 1.66E-5/R\bar{Y}$$

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

$$W_{PWR}^* = 98.3 \text{ person-rem/R}\bar{Y}$$

$$W_{BWR}^* = 117.9 \text{ person-rem/R}\bar{Y}$$

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

$$\Delta\bar{F}_{PWR} = 1.8E-5/R\bar{Y}$$

$$\Delta\bar{F}_{BWR} = 8.0E-6/R\bar{Y}$$

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

$$\Delta W_{PWR} = 47 \text{ person-rem/R}\bar{Y}$$

$$\Delta W_{BWR} = 57 \text{ person-rem/R}\bar{Y}$$

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

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
1.9E+5	1.8E+7	0

ATTACHMENT 1 (To Table 1)

GRAND GULF

The Grand Gulf Unit 1 probabilistic risk assessment (PRA) (Drouin, et al., 1987) was chosen to represent a "typical" boiling water reactor (BWR) PRA for this analysis. Several systems at Grand Gulf are affected by HVAC or room cooling failures. These include: (1) emergency switchgear and battery rooms; (2) standby diesel generator rooms; and (3) pump rooms for RHR, RCIC, HPCS, and LPCS. A loss of cooling is assumed to fail operating diesel generators in 15 minutes; battery, emergency switchgear, and low pressure ECCS (RHR and LPCS) pumps in 4 hours; and HPCS and RCIC in 12 hours. A complete description of the Grand Gulf emergency ventilating system is provided by Drouin, et al. (1987). An examination of the system fault trees (Appendix A of Drouin, et al. 1987) provides a list of basic events that result in failure of systems due to HVAC failure. Table A.1 of this attachment lists the key basic events used in the analysis. A complete list of basic events is provided in Section IV.8 of Drouin, et al. (1987).

Next, the core melt consequences for the dominant core damage categories were examined. When a sequence was found that contained one of the basic events listed in Table A.1, it was reviewed to determine what effect eliminating the dependence on HVAC/room cooling would have on it. Three effects were possible. First, the sequence could no longer result in core damage. These scenarios were eliminated, and their frequency contributed directly to a reduction in core damage frequency. Second, the HVAC/room cooling basic event was replaced with another basic event, e.g., replacing diesel generator room cooling failure with a diesel generator hardware failure. The new sequence was compared with the other sequences in the dominant sequence model, and if an identical sequence was found, the new sequence was absorbed and the frequency of the original sequence contributed directly to a reduction in core damage frequency. The third effect is similar to the second except that no other sequence was found to be identical. In this case, the new sequence was substituted for the original, and a new sequence frequency was calculated. The contribution to changing the core melt frequency was the difference between the original and the new frequency, and could result in a reduction or an increase in the core damage frequency. Table A.2 lists the sequences that were affected and indicates the resultant change in core damage frequency.

The evaluation was performed assuming that all dependence on HVAC and room cooling could be eliminated. The elimination of this dependence is expected to be accomplished by a reevaluation of room heat capacity and heatup, by the implementation of procedures that provide alternate cooling, or by implementing hardware modifications that would eliminate the dependence. The result of the evaluation was a reduction in the core damage frequency at Grand Gulf Unit 1 of $8.0E-6$ events per year. It should be noted that this reduction is a maximum using the above assumptions and may not be achievable in all plants. However, every nuclear plant should realize some reduction in risk of core damage.

The reduction in core melt frequency was translated into a reduction in public risk using the factors in Appendix D of Andrews, et al. (1983). For conservatism, the affected release category for this issue was BWR-2 in Andrews, et al. (1983). That is, all affected accident sequences were assigned to this release category which represents the highest public dose consequences. The base-

case and adjusted-case core-melt frequencies were therefore multiplied by $7.1\text{E}+6$ person-rem/event to calculate the base-case and adjusted-case public risk estimates.

OCONEE

The HVAC systems at Oconee were examined by Sugnet, et al. (1984). They consist of various flow-through systems as well as individual room coolers. Two flow-through systems, one for the control room and one for the auxiliary building, and the room coolers for the low pressure injection (LPI) and reactor building spray system (RBSS) were examined in detail. The PRA determined that only the room coolers for LPI and RBSS had a potential contribution to core damage. This contribution is examined below. The unavailability values used in the evaluation were obtained from Appendix A of Sugnet, et al. (1984).

Room Cooler Failure	= Fan 1 * (Operator + Fan 2)
	= $1.9\text{E}-04 * (1.0\text{E}-03 + 1.9\text{E}-04)$
	= $2.4\text{E}-07$
LPI Pump Fails (start and run)	= $1.3\text{E}-03$
Room Cooler Contribution	= 0.02%
RBSS Pump Fails (start and run)	= $8.3\text{E}-03$
Room Cooler Contribution	= 0.003%

As a final part of the evaluation, an examination of the leading core damage sequences from internal events was done. Failures of the LPI and RBSS pumps were not contributors to any of these sequences.

A second potential contribution from HVAC or room cooling was through the loss of service water, which acts as the heat sink. Service water, however, also provides direct cooling of the high pressure injection (HPI) pumps. An examination of the event trees for Oconee shows that core damage mitigation using the low pressure systems can only be accomplished if HPI is successful first. Therefore, a failure of service water results in a loss of HPI and core damage, and no contribution from loss of room cooling would be found.

It was concluded based on the preceding discussion that no further analysis using Sugnet, et al. (1984), would be performed because of a lack of dependency on HVAC and room cooling. Therefore, to model the effects of SIR on a generic PWR, the scaling technique described on Page 6.12 of Andrews, et al. (1983), was used. The scaling equations are as follows:

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

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

Substituting the quantities derived previously for $(\bar{F})_{\text{BWR}}$ and $(W)_{\text{BWR}}$ yields the following results for the generic PWR:

$$(\bar{F})_{\text{PWR}} = (1/0.45)(\bar{F})_{\text{BWR}} = 5.47\text{E}-5/\text{RY}$$

$$(W)_{\text{PWR}} = (1/1.2)(W)_{\text{BWR}} = 145.6 \text{ person-rem/Ry}$$

A similar procedure was followed to calculate the adjusted-case affected core-melt frequency ($\bar{F}^*$) and adjusted-case affected public risk (W^*) for the generic PWR.

TABLE A.1: HVAC Key Basic Events

System Affected	Basic Event Designator	Basic Event Description
DC Power	DCP-PHN-LP-BAT	125V DC Bus Failure due to Loss of HVAC
Diesel Generator 11	EHV-PSF-HW-77CIA	Diesel Generator 11 Room Cooler Hardware Failure
	EHV-PSF-MA-77CIA	Diesel Generator 11 Room Cooler Out for Maintenance
	ESW-HTX-MA-HX4A	Diesel Generator 11 Cooling Heat Exchanger Train SS6 Out for Maintenance
	ESW-PSF-HW-SS6	Diesel Generator 11 Cooling Heat Exchanger Train Failure
Diesel Generator 12	EHV-PSF-HW-77CIB	Diesel Generator 12 Room Cooler Hardware Failure
	EHV-PSF-MA-77CIB	Diesel Generator 12 Room Cooler Out for Maintenance
	ESW-HTX-MA-HX4B	Diesel Generator 12 Cooling Heat Exchanger Train SS20 Out for Maintenance
	ESW-PSF-HW-SS20	Diesel Generator 12 Cooling
Diesel Generator 13	EHV-PSF-HW-77C02	Diesel Generator 13 Room Cooler Hardware Failure
	EHV-PSF-MA-77C02	Diesel Generator 13 Room Cooler Out for Maintenance
	ESW-MDP-MA-2C	SSW Pump 2C Out for Maintenance
	ESW-MOV-CC-11	SSW Valve MV11 Fails to Open
	ESW-MOV-MA-11	SSW Valve MV11 Out for Maintenance
	ESW-PSF-HW-SS11	SSW Pump Train C Failure
	ESW-XVM-PG-13	SSW Train C Common Injection Valve XV13 Plugs

TABLE A.1 Continued

System Affected	Basic Event Designator	Basic Event Description
Diesel Generator 11, LPCS, RCIC, and RHR Train A	ESW-MDP-MA-1A	SSW Pump 1A Out for Maintenance
	ESW-MOV-CC-1A	SSW Train A Valve MV1A Fails to Open
	ESW-MOV-MA-1A	SSW Train A Valve MV1A Out for Maintenance
	ESW-MOV-CC-5A	SSW Train A Valve MV5A Fails to Open
	ESW-MOV-MA-5A	SSW Train A Valve MV5A Out for Maintenance
	ESW-PSF-HW-SS1	SSW Pump Train A Failure
	ESW-PSF-PG-SS2	SSW Pump 1A Oil Cooler Train Failure
Diesel Generator 12, RHR Train B and C	ESW-MDP-MA-1B	SSW Pump 1B Out for Maintenance
	ESW-MOV-CC-1B	SSW Train B Valve MV1B Fails to Open
	ESW-MOV-MA-1B	SSW Train B Valve MV1B Out for Maintenance
	ESW-MOV-CC-5B	SSW Train B Valve MV5B Fails to Open
	ESW-MOV-MA-5B	SSW Train B Valve MV5B Out for Maintenance
	ESW-PSF-HW-SS16	SSW Pump Train 1B Failure
	ESW-PSF-PG-SS17	SSW Pump 1B Oil Cooler Failure

Table A.2: Summary of Core Damage Evaluation of Grand Gulf Unit 1

Plant Damage State	Description of Modification	Original Frequency	Adjusted-Case Frequency, yr ⁻¹
TB	Modified sequences absorbed in the following retained sequences: 1, 2, 3, 4, 13, 14, 19, 20, 21, 30, 31, 32, 52, 53, 54, 66, 72, and 165. Sequences 30 and 52 were changed to a frequency of 1.35E-8 by removing RA-DCHW-J. Sequences 66 and 72 increased to 7.53E-7 to include HPCS hardware failure.	2.0E-05	1.35E-05
TBVW	Sequences 17 to 49 were modified and absorbed in Sequences 1 to 16	1.1E-06	7.66E-07
TBUI	Modified sequences were absorbed by the following retained sequences: 15, 16, 17, 24, 25, 26, 36, 40, 41, 42, 43, 87, 91, 92, 109, 110, 114, 121, 258-294, 353-365, 394-405, 597-623, 627, 628, 629, 630, 638, 639, 643, 644, 661, 662, and 670. Sequence 24 frequency was increased to 5.89E-7 to account for HPCS failure.	2.4E-06	1.36E-06
TBU1VW	Modified Sequences 1 to 35 and 55 to 72, which were absorbed in the remaining sequences	9.5E-08	5.46E-08
TBU	Modified sequences absorbed by the following retained sequences: 1-101, 132-155, 197-214, 224-243, 277-284, 291-302, 383-390, 400, 401, 412-418	9.4E-07	6.2E-07
TBUVW	Modified sequences were absorbed by the following retained sequences: 1-50, 64, 65, 69, 70, 74, 75, 79, 80, 84, 85, 89, and 90	2.1E-08	1.74E-08
TBUX	No Modifications	3.8E-06	3.8E-06
ATWS	No Modifications	1.8E-07	1.8E-07
TQUV	No Modifications	3.4E-05	3.4E-07
Total (Modified Sequences Only)		2.46E-05	1.66E-05

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Availability of Chilled Water Systems and Room Cooling (143)

2. Affected Plants (N):

All PWRs and BWRs are assumed to be affected.

	<u>All plants</u>	<u>Backfit</u>	<u>Forward-fit</u>
PWRs	90	47	43
Bwrs	44	24	20

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

	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	27.7	30	28.8
BWRs	25.2	30	27.4

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

$$\begin{aligned} \text{PWR} &= (19,860 \text{ man-rem})(1.8\text{E-}5/\text{RY}) = 0.36 \text{ man-rem/RY} \\ \text{BWR} &= (19,860 \text{ man-rem})(8.3\text{E-}6/\text{RY}) = 0.16 \text{ man-rem/RY} \end{aligned}$$

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

<u>Best Estimate</u>	<u>Error Bounds (person-rem)</u>	
<u>(person-rem)</u>	<u>Upper</u>	<u>Lower</u>
1.1E+3	5.0E+6	0

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

Labor requirements for removal and replacement of 8 heat exchangers within containment was estimated by SEA (1986) to be approximately 4000 man-hrs. A labor productivity factor of 3.1 (SEA 1986) was used to adjust the labor estimate to account for inefficiencies caused by work in radiation zones, in congested areas, and in difficult access areas. The total labor in radiation zones is therefore 12,400 man-hrs/plant. See Attachment 2 for additional information.

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

Radiation fields of 25 mrem/hr (0.025 rem/hr) are assumed to exist inside containment where the room coolers are located.

$$D = (0.025 \text{ rem/hr})(12,400 \text{ man-hrs/plant}) = 310 \text{ person-rem/plant}$$

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

Occupational doses will be incurred only at operating plants.

$$\begin{aligned} \text{ND} &= (47 \text{ PWRs} + 24 \text{ BWRs})(310 \text{ person-rem/plant}) \\ &= 2.2\text{E}+4 \text{ person-rem} \end{aligned}$$

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

No incremental labor is foreseen for operation and maintenance of the SIR.

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

$$D_o = 0$$

11. Total Occupational Dose Increase for SIR Operation/Maintenance, ($\overline{\text{NTD}}_o$):

$$\overline{\text{NTD}}_o = 0$$

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
2.2E+4	6.6E+4	7.3E+3

3.0 SAFETY ISSUE COSTS

The industry and NRC costs associated with resolution of this safety issue 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:

Availability of Chilled Water Systems and Room Cooling (143)

2. Affected Plants (N):

All PWRs and BWRs are assumed to be affected.

	<u>All plants</u>	<u>Backfit</u>	<u>Forward-fit</u>
PWRs	90	47	43
BWRs	44	24	20

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

	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	27.7	30	28.8
BWRs	25.2	30	27.4

Industry Costs (Steps 4 through 12):

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

$$\Delta\bar{F}A \text{ (PWR)} = (\$1.65\text{E}+9)(1.8\text{E}-5/\text{RY}) = \$ 3.0\text{E}+4/\text{RY}$$

$$\Delta\bar{F}A \text{ (BWR)} = (\$1.65\text{E}+9)(8.0\text{E}-6/\text{RY}) = \$ 1.3\text{E}+4/\text{RY}$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$9.3E+7	\$1.7E+9	\$0

6. Per-Plant Industry Resources for SIR Implementation:

Engineering study 52 man-wks
Hardware modification labor 12,400 man-hr

See Attachment 2.

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

Engineering study \$ 100,000/plant
Hardware modification \$ 1,324,000/plant
Procedural changes \$ 7,200/plant
I (TOTAL) \$ 1,431,000/plant

See Attachment 2.

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

These costs are applicable to only backfit plants.

$$NI = (47 \text{ PWRs} + 24 \text{ BWRs})(1,431,000/\text{plant}) = \$1.0\text{E}+8$$

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

No incremental industry resources are foreseen for SIR operation and maintenance.

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

$$I_o = 0$$

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

$$NTI_o = 0$$

12. Total Industry Cost (S_i):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.0E+8	\$1.5E+8	\$5.0E+7

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC development costs are estimated directly in Step 14.

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

Research program (contractor) . . .	\$ 50,000
Issue Generic Letter	<u>\$ 13,000</u>
C_D (TOTAL)	<u>\$ 63,000</u>

See Attachment 2.

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

In Attachment 2, it was estimated that 1 man-wk of NRC labor is required to review the engineering analyses submitted for each plant in response to the Generic Letter.

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

$$C = (1 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$2270/\text{plant}$$

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

$$NC = (90 \text{ PWRs} + 44 \text{ BWRs})(\$2270/\text{plant}) = \$3.0\text{E}+5$$

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

No incremental NRC resources are foreseen for SIR operation and maintenance.

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

$$C_o = 0$$

20. Total NRC Cost for Review of SIR Operation and/Maintenance ($\overline{NTC}_o$):

$$\overline{NTC}_o = 0$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$3.7E+5	\$5.2E+5	\$2.2E+5

ATTACHMENT 2 (To Table 3)

This attachment presents the basis for the labor and resource requirements that are necessary to implement the proposed SIR. The first step in SIR involves issuance of a Generic Letter by the NRC to all power reactor licensees to evaluate the dependencies on adequate room cooling of the operation of HVAC and chilled water systems. Issuance of the Generic Letter is assumed to cost the NRC approximately the same as a simple technical specification change, as described in SEA (1986), including the necessary reviews and approvals. This cost estimate is \$13,000. This Generic Letter would include a description of the proposed resolution and guidance on acceptable procedures for licensees to use to evaluate the potential dependencies in the designs of these systems and alternative solutions to improving the independence of systems that are critical to plant risk. It was assumed that a research project would form the basis for a more fully developed SIR and for the guidance in the Generic Letter. This research project, which is assumed to be performed by an NRC contractor, was estimated to require a 6 man-month effort.

The next item in the proposed SIR is for licensees to evaluate the dependencies of HVAC and chilled water systems on room cooling systems and the subsequent effects on plant risk. This engineering analysis is assumed to require approximately 1 man-yr of effort (\$100,000/plant) and would include the identification and evaluation of dependencies, identification of appropriate procedure changes and hardware modifications to minimize the effects of the dependencies on plant risk, and related safety analyses. The licensee would submit this evaluation to the NRC for review. NRC approval of the proposed modifications at each plant is assumed to result from this review. This review was estimated to require approximately 1 man-wk of NRC labor per plant.

The next step in this SIR would be for the licensees to implement the proposed procedural changes and hardware modifications at the plants. Procedural changes would be proposed where possible because of the expense and occupational doses associated with implementing hardware modifications in radiation areas. However, it is believed that procedural changes will not resolve the dependencies in some cases, particularly for cases in which failure of room cooling is a critical failure mode for an important safety-related component. In these cases, it may be necessary to replace a particular piece of equipment with one that can be operated independently of HVAC and chilled water systems. Alternatively, a component of the HVAC or chilled water system, such as one or more of the room coolers themselves, may be replaced with one of a different design that can be operated independently of the HVAC or chilled water systems.

The cost estimates for the procedural changes and hardware modifications were taken from SEA (1986). At Grand Gulf, the emergency ventilation system removes heat from 10 rooms: emergency switchgear and battery rooms; diesel generator rooms (3); and safety-related pump rooms (6), including three residual heat removal (RHR) pump rooms, one high pressure core spray pump room, one reactor core isolation cooling (RCIC) pump room, and one low pressure core spray (LPCS) pump room. It was assumed that each plant would implement 2 procedural changes that would eliminate dependencies on room cooling. The remaining 8 rooms would require replacement of their heat exchangers to eliminate this dependency.

The procedure changes are anticipated to be relatively complex. Therefore, according to SEA (1986), the costs for implementing complex procedural changes are approximately \$3600/change or \$7200/plant.

The hardware upgrade is assumed to be represented by removal and replacement of containment spray heat exchangers, as described in SEA (1986). The proposed SIR is somewhat similar to the removal and replacement of the containment spray heat exchangers because work would be performed in a radiation environment and in a relatively congested area. It is also stated in SEA (1986) that these cost estimates should be applied only to plants that are operating or at the 70% or higher completion stage of construction. The removal and replacement cost, estimated by SEA (1986) and summarized below, was therefore applied only to operating plants. The proposed resolution was assumed to result in no incremental costs for plants that are currently under construction.

(1) Purchase cost - 8 room coolers	\$ 968,000
(2) Labor costs (500 man-hrs/cooler @ \$28.71/hr times labor productivity adjustment factors, removal factors, engineering, and QA factors)	<u>\$ 356,000</u>
TOTAL:	<u>\$1,324,000</u>

Therefore, the industry costs for implementation were estimated to be \$100,000/plant for the engineering study plus \$1,324,000/plant for the hardware upgrade plus \$7200/plant for procedure changes for a total implementation cost of \$1,431,000/plant.

These upgrades and changes are anticipated to be one-time activities. No additional utility resources are expected for operation and maintenance of the upgraded equipment. In addition, no incremental NRC resources are foreseen for review of the operation and maintenance of the upgraded equipment.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 144, SCRAM Without a Turbine/Generator Trip

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: This issue addresses the potential for failure of the turbines to trip after a reactor trip and the associated concomitant effects of this event which may include reactor vessel rupture (i.e., pressurized thermal shock effects), the potential generation of turbine missiles that may result from a turbine overspeed condition, reactor coolant pump seal LOCA, small-break LOCA, and steam generator tube rupture. All of these latter events may result directly from or be exacerbated by the reactor trip without turbine trip event. The proposed safety issue resolution (SIR) involves four activities designed to improve the probability that a turbine trip occurs following a reactor trip (Kniel 1988). These include installation of redundant and diverse emergency turbine trip systems; improved periodic inspection, testing, and maintenance of turbine/generator systems; clarification of procedures and training for operating personnel; and revisiting design basis calculations.

AFFECTED PLANTS: This safety issue affects only operating plants. Both operating PWRs and BWRs are affected.

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 1.2E+4

OCCUPATIONAL DOSES:

SIR Implementation =	9.0E+0
SIR Operation/Maintenance =	0
Total of Above =	9.0E+0
Accident Avoidance =	9.1E+1

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	1.9E+1
SIR Operation/Maintenance =	7.8E+0
Total of Above =	2.7E+1
Accident Avoidance =	7.5E+0

NRC COSTS:

SIR Development =	5.5E-2
SIR Implementation =	7.8E-1
SIR Operation/Maintenance Review =	2.6E+0
Total of Above =	3.4E+0

SCRAM WITHOUT A TURBINE/GENERATOR TRIP

ISSUE 144

1.0 SAFETY ISSUE DESCRIPTION

This issue addresses the potential for failure of the turbines to trip after a reactor trip and the associated concomitant effects of this event. Failure of the turbine to trip may lead to a primary system overcooling event. The degree of overcooling, and subsequent events, are dependent upon the length of time that steam is allowed to continue to pass into the turbine. This event is evaluated in terms of its effects on the probability of reactor vessel rupture (i.e., pressurized thermal shock effects), the potential generation of turbine missiles that may result from a turbine overspeed condition, reactor coolant pump seal LOCA, small-break LOCA, and steam generator tube rupture. All of these latter events may result directly from or be exacerbated by the reactor trip without turbine trip event.

PROPOSED RESOLUTION

The proposed safety issue resolution (SIR) involves four activities designed to improve the probability that a turbine trip occurs following a reactor trip (Kniel 1988). The first item is to provide for redundant and diverse actuation trains (e.g., solenoid actuated emergency turbine trip valves) for tripping the turbine/generator downstream of the point of arrival of the trip signals. The second activity is to improve periodic inspection, testing, and maintenance of turbine/generator systems. The third activity is to clarify procedures and training for operating personnel to respond to reactor trip without turbine trip events. For example, operators should be aware of the potential catastrophic effects of turbine over-speed conditions that could be produced if the generator is tripped. A fourth step is to perform design basis calculations to assure that transients involving turbine/generator trip failure do not produce conditions worse than those previously analyzed and to support the procedures and training activities described above.

AFFECTED PLANTS

This safety issue is assumed to affect only operating plants, including PWRs and BWRs. It is believed that implementation of SIR can be achieved at future plants with minimal incremental costs and, since the proposed SIR would be designed into the plant, would not reduce the public risks associated with future plant operation.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this issue are estimated in this section and summarized in Tables 1 and 2. Attachment 1 describes the basis for the risk and dose changes that are estimated in the Tables.

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

SCRAM Without a Turbine/Generator Trip (144)

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

All PWRs and BWRs are assumed to be affected.

	<u>N</u>	<u>$\bar{T}$ (yr)</u>
PWRs	47	27.7
BWRs	24	25.2

3. Plants Selected for Analysis:

Representative PWR: Oconee-3

Representative BWR: Grand Gulf-1

4. Parameters Affected by SIR:

This safety issue was not amenable to standard safety issue prioritization analysis methodology as described by Andrews, et al. (1983). Therefore, new parameters and accident sequences were developed to determine the base-case core-melt frequency and public risks. The parameter affected by this safety issue was designated "TT," failure of the turbine to trip after a reactor trip (see Attachment 1). In addition, SIR would affect the probabilities of human errors associated with failure to take manual actions in the control room as well as locally at the turbine control panels to trip the turbine.

5. Base-Case Values for Affected Parameters:

The base-case value of TT was $1.7E-3/\text{demand}$. Base-case values for the other parameters affected by SIR are discussed in Attachment 1.

6. Affected Accident Sequences and Base-Case Frequencies:

<u>PWRs</u>	<u>(RY⁻¹)</u>
Pressurized Thermal Shock	2.3E-9
Turbine Missile	3.4E-7
Steam Generator Tube Rupture	3.5E-6
PORV LOCA	1.0E-6
Reactor Coolant Pump Seal LOCA	<u>1.4E-7</u>
TOTAL:	<u>5.0E-6</u>

<u>BWRs</u>	<u>(RY⁻¹)</u>
Pressurized Thermal Shock	2.0E-9
Turbine Missile	2.9E-7
PORV LOCA	5.9E-7
Reactor Coolant Pump Seal LOCA	<u>7.9E-8</u>
TOTAL:	<u>9.6E-7</u>

7. Affected Release Categories and Base-Case Frequencies:

	<u>Containment Failure Mode</u>	<u>Release Category</u>	<u>Frequency, RY⁻¹</u>
PWRs	γ	3	2.5E-6
	β	5	3.7E-8
	ϵ	7	<u>2.5E-6</u>
			<u>5.0E-6</u>
BWRs	γ	3	4.8E-7
	δ	4	<u>4.8E-7</u>
			<u>9.6E-7</u>

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

PWRs: $\bar{F} = 5.0E-6/\text{RY}$

BWRs: $\bar{F} = 9.6E-7/\text{RY}$

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

$W_{\text{PWR}} = 1.3E+1 \text{ person-rem/RY}$

$W_{\text{BWR}} = 2.9E+0 \text{ person-rem/RY}$

10. Adjusted-Case Values for Affected Parameters:

The adjusted-case values for mechanical and human error related turbine trip parameters are discussed in Attachment 1.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

<u>PWRs</u>	<u>(RY⁻¹)</u>
Pressurized Thermal Shock	6.7E-10
Turbine Missile	9.9E-8
Steam Generator Tube Rupture	1.4E-6
PORV LOCA	3.0E-7
Reactor Coolant Pump Seal LOCA	4.0E-8
TOTAL:	1.8E-6

<u>BWRs</u>	<u>(RY⁻¹)</u>
Pressurized Thermal Shock	5.8E-10
Turbine Missile	8.6E-8
PORV LOCA	1.8E-7
Reactor Coolant Pump Seal LOCA	2.4E-8
TOTAL:	2.9E-7

12. Affected Release Categories and Adjusted-Case Frequencies:

	<u>Containment Failure Mode</u>	<u>Release Category</u>	<u>Frequency, RY⁻¹</u>
PWRs	γ	3	9.0E-7
	β	5	1.3E-8
	ϵ	7	9.0E-7
			1.8E-6
BWRs	γ	3	1.5E-7
	δ	4	1.5E-7
			2.9E-7

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

$$\text{PWRs: } \bar{F}^* = 1.8\text{E-6/RY} \quad \text{BWRs: } \bar{F}^* = 2.9\text{E-7/RY}$$

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

$$W^*_{\text{PWR}} = 5.0\text{E+0 person-rem/RY} \quad W^*_{\text{BWR}} = 8.3\text{E-1 person-rem/RY}$$

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

$$\Delta\bar{F}_{\text{PWR}} = 3.2\text{E-6/RY} \quad \Delta\bar{F}_{\text{BWR}} = 6.7\text{E-7/RY}$$

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

$$\Delta W_{\text{PWR}} = 8.0\text{E+0 person-rem/RY} \quad \Delta W_{\text{BWR}} = 2.1\text{E+0 person-rem/RY}$$

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

Best Estimate (person-rem)	Error Bounds (person-rem)	
	<u>Upper</u>	<u>Lower</u>
1.2E+4	5.6E+5	0

ATTACHMENT 1

This issue addresses the potential for failure of the turbines to trip after a reactor trip and the associated concomitant effects of this event. Failure of the turbine to trip may lead to a primary system overcooling event. The degree of overcooling, and subsequent events, are dependent upon the length of time that steam is allowed to continue to pass into the turbine. Accident sequences involving failure of the turbine to trip following a reactor SCRAM are not necessarily quantified in existing Probabilistic Risk Assessments (PRAs). As a result, new accident sequence models were developed for this analysis.

PLANT MODEL AND BASIC EVENT QUANTIFICATION

Plant response to a reactor trip and failure of the turbine to trip are as follows. The first response is for the Integrated Control System (ICS) to rapidly close the turbine governor valves. Operators may also manually trip the turbine from the control room. The next potential response is for operators to trip the generator which, if successful, will subsequently trip the turbine. Generator trip actuators are located in the control room on the same panel as turbine trip actuators. If these actions are not successful, the plant would be susceptible to turbine missiles resulting from turbine overspeed conditions as well as pressurized thermal shock (PTS) of the reactor vessel that would result from overcooling of the primary coolant system. The plant would also be susceptible to other failures, as discussed later in this analysis.

The previous discussion led to the development of an event tree model for failure of the turbine to trip after a reactor trip. This model is shown in Figure 1. The figure shows the plant conditions and frequencies for each possible outcome. The frequencies on the event tree models were used as the starting point for examining subsequent events that could lead to core damage. The subsequent events represent plant conditions related to successful operation of main and emergency feedwater systems and successful initiation of emergency core cooling and inventory makeup functions leading to establishment of long-term decay heat removal. These systems would be needed to recover from turbine trip failure and to respond to simultaneous failures in other systems, such as steam generator tube ruptures, that may be exacerbated by turbine trip failure. Separate event trees were developed for feedwater-related events and for core cooling/inventory makeup events. The feedwater event tree is presented in Figure 2. Core cooling/inventory makeup and long-term cooling are substantially different for PWRs and BWRs. Therefore, separate event trees, shown in Figures 3a and 3b, were developed for PWRs and BWRs, respectively. Brief descriptions of the events on the event trees and their probabilities are provided in the following pages.

FIGURE 1: Turbine Trip Event Tree Model

Plant State

N = Normal Feedwater Flow
 UF = Underfeed
 OF = Overfeed

FIGURE 2: Feedwater Event Tree Model

NA = Not applicable
 CM = Core melt
 OK = Adequate core cooling and inventory established

FIGURE 3A: Event Tree Model for Establishment of Long-term Core Cooling and Inventory at PWRs

	RPT	SRV	LT-DHR	Plant State	Seq. No.	Conditional PWR	Probability BWR
				OK	a'	NA	4.9E-3
				CM	b'	NA	
				OK	c'	NA	
				CM	d'	NA	2.4E-2
				OK	e'	NA	1.5E-4
				CM	f'	NA	
				OK	g'	NA	7.5E-4
				CM	h'	NA	

NA = Not applicable
 CM = Core melt
 OK = Adequate core cooling and inventory established

FIGURE 3B: Event Tree Model for Establishment of Long-term Core Cooling and Inventory at BWRs

Initiating Event - Reactor Transient: The frequencies of reactor transients were taken from Mackowiak, Gentillon, and Smith (1985). The total frequencies were 8.5/RY for PWRs and 7.4/RY for BWRs.

Turbine Trip (TT): The probability of turbine trip failure given reactor trip, including operator error and equipment failure contributions, was taken from Charlton (1988) and amounts to $1.7E-3/\text{demand}$.

Operator Fails to Trip Generator (GEN-OP): Operator error to manually initiate a generator trip from the control room, which would subsequently lead to a turbine trip, was assigned a value of 0.1/demand. This estimate is considered to be conservative, given that the generator trip actuators are located in the control room on the same panel as the turbine trip actuators and positive indication is given for turbine trip success or failure.

Generator Trip Failure - Equipment Faults (GEN-EF): Generator trip failures that are caused by equipment faults were assigned a value of 0.5/demand. No data was available to more accurately predict the value for this parameters so a conservative value of 0.5 was used.

Automatic Turbine Runback - Control Valves (TRB-A): As with generator trip equipment faults, this parameter was assigned a failure probability of 0.5/demand.

Manual Turbine Runback (TRB-M): This parameter was assigned a value of 0.2/demand. The operators could provide this function from the control room and would be alerted to turbine trip failure by positive indications as discussed above for parameter GEN-OP. However, to model the increased likelihood that operators would fail to perform a second action, given failure to perform the first manual action (i.e., the first potential operator action was to manually initiate generator trip), the value for GEN-OP was multiplied by a factor of 2.

Manual Turbine Runback - Equipment Faults (TRB-EF): Equipment faults in the turbine runback system were assigned a failure probability of 0.5/demand as was done for other hardware failures (see parameters TRB-A and GEN-EF).

Local Turbine Trip - Operator Action (LTT-OP): Turbine speed indications are provided at the turbine control console near the turbines. A trip is also provided locally at the turbine control console. This would be the third potential operator action in this event tree. Because of a lack of applicable data, the failure probability for this parameter was assigned a value of 2 times the previous operator error probability or 0.4/demand.

Local Turbine Trip - Equipment Faults (LTT-EF): As was done for the failure probabilities for other equipment faults, a value of 0.5/demand was assigned for this parameter.

Main Feedwater Unavailable (MFW): The value for this parameter was derived from information presented by Mackowiak, Gentillon, and Smith (1985). It was observed that the frequencies of main feedwater and MSIV malfunctions leading to reactor transients were 2.96/RY for PWRs and 1.62/RY for BWRs.

This represents 35% of the annual transient frequency for PWRs and 22% of the annual transient frequency for BWRs. Therefore, the conditional probability that feedwater is available given a reactor transient was estimated to be 0.35/demand for PWRs and 0.22/demand for BWRs.

Main Feedwater Pump Runback (MFP-R): A value of 0.003/demand was assigned to this parameter. This value is based on the probability of an electric pump to start on demand (Bertucio and Julius 1990).

Main Feedwater Pump Trip (MFP-T): A value of 0.003/demand was assigned to this parameter based on the probability of an electric pump to start on demand (Bertucio and Julius 1990).

Emergency Feedwater Pump Fails to Start (EFW-S): The failure probability for this parameter was assigned a value of 0.011/demand. This value is based on the failure probability of a turbine-driven EFW pump to start on demand (Bertucio and Julius 1990).

Emergency Feedwater Level Control (EFW-LC): A value of 0.003/demand was assigned to this parameter. This was based on the failure of a motor-operated valve in the EFW system to open on demand (Bertucio and Julius 1990).

Reactor Coolant Pump Trip (RCP): The value used for this parameter was assumed to be the sum of operator failure to trip the RCP and equipment faults in the RCP trip system. The values used were taken from Sugnet, et al., (1983) and amounted to 0.01/demand for operator errors and 0.02/demand for equipment faults. The total failure probability was estimated to be 0.03/demand.

Steam Generator Isolation (SG-ISO): The failure probability for this event was assigned a value of 0.25/demand based on operator failure to close MSIVs (Sugnet, et al., 1983).

Operator Initiates Feed/Bleed Cooling (F/B): Two values were used in this analysis. The first represents the situation in which successful steam generator isolation occurs and the second represents the situation in which steam generator isolation is not successful. A review of the probabilities given by Sugnet, et al., (1983) indicated that a range of values from 0.015/demand to 0.071/demand have been used for this parameter in previous PRAs. For this analysis, the lower range will be used to represent the successful steam generator isolation case and the upper range will be used for unsuccessful steam generator isolation.

Pilot-Operated Relief Valve Cycles Open/Closed Successfully (PORV): This event was inserted in the event tree (Figure 3a) to model the potential LOCA sequence that could result from a stuck-open PORV. The failure probability for this event was assigned a value of 0.5/demand.

Establish Long-Term Decay Heat Removal (LT-DHR): This event can be accomplished in several ways at both types of plants depending upon the prior events. In general, this could be accomplished through recovery of feedwater or condensate systems or through lowering of the primary system pressure to the level at which normal decay heat removal systems can be

initiated. Lowering primary system pressure may be accomplished at BWRs by automatic or manual actuation of the Automatic Depressurization System (ADS) or through lowering steam pressure using the turbine bypass system. Primary coolant inventory may be maintained through the high pressure core spray or reactor isolation cooling (RCIC) systems or, if depressurization is successful, low pressure core coolant injection systems may be used. At PWRs, lowering primary system pressure may be accomplished by restarting the RCPs and manual operation of pressurizer sprays or through lowering steam pressure using the turbine bypass system. Manual operation of the RCPs is important in controlling heat transfer from the primary to secondary systems.

Two values were used for this event. The first value represents the case in which the previous event, PORV for PWRs and SRV for BWRs, is successful. Failure to initiate long-term cooling was assigned a value of 0.01/demand for this case. A value of 0.05/demand was assigned for the second case in which PORV or SRV operation is unsuccessful. These values are conservative estimates based on data given by Sugnet, et al., (1983) and Bertucio and Julius (1990).

Recirculation Pump Trip (RPT): The value used for this parameter on Figure 3b (BWRs) was assumed to be the sum of operator failure to trip the recirculation pump and equipment faults in the Recirculation pump trip system. The values used were assumed to be identical to RCP trip failure probabilities at PWRs that were taken from Sugnet, et al., (1983) and amounted to 0.01/demand for operator errors and 0.02/demand for equipment faults. The total failure probability was estimated to be 0.03/demand.

Safety Relief Valve Successfully Opens/Reseats (SRV): This event was inserted in the event tree (Figure 3b) to model the potential LOCA sequence that could result from a stuck-open SRV at BWRs. The failure probability for this event was assigned a value of 0.5/demand (see also event PORV).

CORE-MELT SEQUENCE FREQUENCIES

The effects of turbine trip failures following a reactor SCRAM could potentially include pressurized thermal shock (PTS), turbine missiles, and loss of core-cooling and inventory-makeup. These effects and their associated core-melt frequencies are described in this section.

Pressurized Thermal Shock

Significant overcooling of the primary coolant system could result in a pressurized thermal shock (PTS) condition which could lead to failure of the reactor vessel. To determine the conditional probabilities of reactor vessel failure given the occurrence of an overcooling transient, the PTS studies of Calvert Cliffs (Selby, et al., 1985a) and H. B. Robinson (Selby, et al., 1985b) were reviewed. These documents used computerized thermal-hydraulics and fracture mechanics models to estimate the probabilities of reactor vessel failure for a series of potential overcooling conditions. Included in the analyses were several overcooling sequences that were initiated by large steam-line breaks occurring downstream of the MSIVs. A large steam-line break downstream of the MSIVs was stated to be functionally similar to the failure of the main turbine to trip

following a reactor trip (Selby, et al., 1985b). Therefore, the conditional reactor vessel failure probabilities for large steam-line break sequences were examined to estimate the probability of reactor vessel failure given occurrence of a turbine trip failure following a reactor trip.

The possible steam-line break sequences evaluated by Selby, et al., (1985a and 1985b) were reviewed in light of the event trees shown in Figures 1 and 2. Selby, et al., (1985a and 1985b) included, among others, sequences involving failure to isolate the steam generators. One sequence included blowdown of one SG, failure to isolate auxiliary feedwater (AFW) to the low pressure SG, and failure to throttle AFW. This sequence, which is functionally similar to the events leading to Plant State TT-F on Figure 1 followed by Plant State OF on Figure 2, resulted in a conditional reactor vessel failure probability of $3.7\text{E-}4$. If the low pressure SG does become isolated, the overcooling condition exists for a much shorter time period, and the reactor vessel failure probability is significantly reduced. No credit was taken for this possibility. It was assumed that these sequences, if they result in reactor vessel failure, would lead to a core damage accident. Therefore, the frequency of these events is the product of the following three terms:

- sum of the turbine trip failure sequence (plant state TT-F) frequencies from Figure 1. These frequencies are $2.1\text{E-}3/\text{RY}$ for PWRs and $1.8\text{E-}3/\text{RY}$ for BWRs.
- sum of the conditional probabilities of main feedwater overfeed conditions from Figure 2 (i.e., sum of probabilities of sequences 2, 5, 7, and 9 from Figure 2). The probability of this plant state is $2.9\text{E-}3/\text{demand}$ for PWRs and $3.0\text{E-}3/\text{demand}$ for BWRs.
- the conditional probability of reactor vessel failure ($3.7\text{E-}4$).

The resulting core-melt frequencies were estimated to be $2.3\text{E-}9/\text{RY}$ for PWRs and $2.0\text{E-}9/\text{RY}$ for BWRs.

Turbine Missiles

A second potential effect of the failure of the main turbine to trip following a reactor trip is to cause a turbine overspeed condition which could result in turbine failure and ejection of missiles from the turbine. Two references (Sugnet, et al., 1984 and Scott, et al., 1987) were reviewed to develop estimates of the probability that a turbine missile could be generated and subsequently result in a safety compromise.

The probability of unacceptable damage resulting from turbine missiles is generally expressed as:

$$P_4 = P_1 * P_2 * P_3$$

where: P_4 = the overall probability of reactor damage due to turbine missiles.
 P_1 = the probability of turbine missile generation and penetration of the turbine casing,

- P_2 = the probability that a turbine missile strikes a critical reactor target, given missile generation and penetration.
- P_3 = the probability that a critical safety-related target is damaged, given a missile strike.

It was assumed that successful closure of the TGV or isolation of the SG (MSIV closure) would terminate the event from a turbine missile generation standpoint because steam flow to the turbine would be halted. Therefore, plants would be vulnerable to turbine missiles if they are in plant state TT-F in Figure 1. For this analysis, the value of P_1 is taken as the probability of turbine trip failure following a reactor trip that is not isolated in any way (i.e., the sum of the frequencies of TT-F plant states from Figure 1) times the probability that a missile is generated given turbine failure. The probability that a missile is generated given turbine failure was estimated to be 0.76 by Scott, et al., (1987). Therefore, the values of P_1 are:

$$\begin{aligned} P_1 \text{ (PWR)} &= (2.1\text{E-}3/\text{RY})(0.76) = 1.6\text{E-}3/\text{RY} \\ P_1 \text{ (BWR)} &= (1.8\text{E-}3/\text{RY})(0.76) = 1.4\text{E-}3/\text{RY} \end{aligned}$$

The values of P_2 and P_3 were taken from the Oconee PRA (Sugnet, et al., 1984). The results of the turbine missile analysis in the Oconee PRA was used because it interjected some plant-specific parameters into the analysis of these probabilities that are believed to more accurately represent the actual situation at most power plants. For example, the turbines are located such that missiles ejected at certain angles, such as missiles ejected directly up or down, would not cause any damage to critical components.

The values of P_2 and P_3 that were used in this analysis were derived as follows. It was observed in the Oconee PRA that the turbine missile frequency was dominated by a one specific vulnerability that consisted of a "window" region of essentially no shielding between the spent fuel storage area and the containment. The frequency of safety compromise for this specific missile was estimated at $1.6\text{E-}9/\text{RY}$. It was also stated that only one turbine blade could potentially be ejected into this region. The failure rate for turbine blades that was used in the Oconee PRA was a $7.81\text{E-}6/\text{blade/yr}$ (effectively the value of P_1). Therefore, for this particular situation, the value of $P_2 * P_3$ was estimated to be $1.6\text{E-}9/7.81\text{E-}6 = 2.05\text{E-}4$. This value represents the conditional probability of safety-significant damage given generation and penetration of a turbine missile.

The resulting core-melt frequencies from turbine missiles, which are the product of parameters P_1 , P_2 , and P_3 , were estimated to be:

- PWR: $(1.6\text{E-}3/\text{RY})(2.1\text{E-}4) = 3.4\text{E-}7/\text{RY}$
- BWR: $(1.4\text{E-}3/\text{RY})(2.1\text{E-}4) = 2.9\text{E-}7/\text{RY}$

Loss of Core Cooling and Inventory Makeup Functions

The events depicted in Figure 1 may also be compounded by a simultaneous occurrence of other initiating events such as primary coolant pipe breaks, steam generator tube ruptures, loss of main feedwater, loss of offsite power, etc. These events may occur independently of the turbine trip failure or may be

exacerbated by the turbine trip. In either case, the frequencies and consequences of these additional failures need to be examined.

Primary coolant system loss of coolant accidents (LOCAs) are categorized according to the size of the break. For large- and medium- LOCAs, the break is large enough to effectively depressurize the primary system. For small LOCAs, the break is not large enough to result in depressurization. The core-damage frequencies from all three types of LOCAs are dominated by one ESF failure; failure to implement recirculation cooling before the Refueling Water Storage Tank (RWST) is depleted (Sattison and Hall 1990). Since the initiating event frequencies for large and medium LOCAs are approximately two orders of magnitude lower than small LOCAs, the large and medium LOCAs will be neglected from further analysis. Steam generator tube ruptures (SGTRs) and reactor coolant pump (RCP) seal LOCAs at PWRs and Recirculation Pump (RP) seal LOCAs at BWRs will be examined further because of their close functional relationships to small pipe-break LOCAs. In addition, failure to restart the RCPs to control heat transfer from the primary to secondary systems may result in heatup and repressurization of the primary system. In this situation, the Pilot-Operated Relief Valves (PORVs) at PWRs and Safety Relief Valves (SRVs) at BWRs may lift to relieve primary system pressure. If the PORV(s) or SRV(s) do not successfully close after the primary system pressure has been relieved, steam/fluid may continue to escape from the primary system resulting in conditions similar to a small LOCA. These sequences are examined further in this analysis.

Rapid steam generator blowdown that would result from turbine trip failure will result in overcooling of the primary system. This overcooling will cause the primary coolant volume to increase which produces a net decrease in the moderator (boron) concentration in the primary coolant. At relatively low boron concentrations, the potential exists for a recriticality event that may produce a pressure pulse that could fail the reactor vessel. However, this analysis addresses events in which the reactor successfully trips so successful insertion of control rods and control blades into the core has occurred prior to this overcooling transient. No mechanism exists for relocation of the control rods as a result of turbine trip failure. In addition, emergency coolant injection in PWRs is normally accomplished using borated water supplies (Scott, et al., 1990) which would prevent recriticality in these plants. In BWRs, the Standby Liquid Control System (SLCS) is the primary source of emergency boration. Therefore, BWRs are also provided with a system that is capable of ensuring subcritical conditions in the core, even without control rods. Turbine trip failure has no apparent effect on the successful actuation of the SLCS. Finally, Scott et al. (1990) concluded that even if a recriticality event occurs, the event is not likely to generate a pressure pulse significant enough to fail the reactor vessel. The frequency and consequences of recriticality events will therefore not be analyzed further.

Loss of main feedwater (MFW) could also occur simultaneously with turbine trip failure. However, in order for a core damage sequence to result from this initiating event, a number of safety systems have to also fail. The dominant safety system failure after a loss of MFW for the Zion plant included loss of service water or component cooling water (CCW) which induces a RCP seal LOCA and failure to recover service water or CCW. The conditional probabilities of these safety system failures are significantly smaller than the responses to LOCAs and are therefore not analyzed further.

Losses of offsite power were also not analyzed further because the failure probabilities for the responses to this initiating event are substantially smaller than those for the LOCAs described above. Other potential initiating events, including ATWS and support system failures (e.g., losses of service water, CCW, and interfacing LOCAs) were also not evaluated further because their frequencies are significantly smaller than RCP seal LOCAs and SGTRs and their unavailabilities are not dependent upon the turbine trip failure event.

Based on the preceding discussion, it is necessary to develop estimates of the probabilities of small break LOCA conditions induced by a stuck-open PORV or SRV, steam generator tube rupture (SGTR), and RCP or RP seal LOCA. These events are treated as being independent of the turbine trip failure. However, the plant responses to these events are dependent upon the prior actions taken to respond to the turbine trip failure.

The probability of a SGTR was derived from information presented in NUREG-0844 (NRC 1988). This document derived estimates for the conditional probabilities of SGTRs after several potential events, including main steam line breaks (MSLB). As discussed above, MSLB accidents result in conditions that are very similar to turbine trip failures following a reactor SCRAM. The conditional probabilities of SGTRs given prior occurrence of a MSLB were given as follows:

- 1 SGTR - conditional probability = 0.025
- 2 to 10 SGTRs - conditional probability = 0.015
- greater than 10 SGTRs - conditional probability = 0.01.

The core-melt frequencies for SGTR events must also include the probabilities of failure of the emergency core cooling and makeup systems. The probabilities of these events are the sum of the conditional probabilities for plant state "CM" in Figure 3a, or 0.044/demand. A factor of 0.5 was applied to sequences involving more than 2 tube ruptures. This is because operators would not be required to take any action to depressurize the primary system following a multiple tube rupture incident. Therefore, the core-melt frequencies associated with SGTRs are as follows:

- 1 SGTR = $(2.1\text{E-}3/\text{RY})(0.025)(0.044)$ = $2.3\text{E-}6/\text{RY}$
- 2-10 SGTR = $(2.1\text{E-}3/\text{RY})(0.015)(0.5)(0.044)$ = $6.9\text{E-}7/\text{RY}$
- > 10 SGTR = $(2.1\text{E-}3/\text{RY})(0.01)(0.5)(0.044)$ = $4.6\text{E-}7/\text{RY}$

The total core melt frequency for SGTRs is the sum of these values or $3.5\text{E-}6/\text{RY}$.

The conditional probabilities of reactor coolant pump (RCP) seal LOCAs at PWRs and Recirculation Pump (RP) seal LOCAs at BWRs were taken from the Zion PRA. According to Sattison and Hall (1990), the failure probability for an RCP seal with normal leakage (i.e., less than 240 gallons per minute) is 0.049. This value is taken to be the conditional probability of RCP or RP seal LOCA given a reactor trip without turbine trip. As with the SGTR event, simultaneous occurrence of the RCP seal LOCA and reactor trip without turbine trip events does not necessarily lead to core-melt. According to the Zion PRA, the core-melt frequency for the RCP seal LOCA sequences are dominated by failure to depressurize the primary system or failure to switch to recirculation cooling. This sequence also requires failure to recover the RCP seal cooling system (i.e., restore service water or

component cooling water) as well as failure to establish long-term decay heat removal. The probability of non-recovery of the RCP seal cooling system was given as 0.03/demand. The probabilities of failing to establish long-term decay heat removal were taken from Figure 3a for PWRs and Figure 3b for BWRs (0.044/demand and 0.03/demand, respectively). Therefore, the core melt frequencies associated with RCP or RP seal LOCAs, given reactor trip without turbine trip, are:

- PWR: $(2.1\text{E-}3/\text{RY})(0.049)(0.03)(0.044) = 1.4\text{E-}7/\text{RY}$
- BWR: $(1.8\text{E-}3/\text{RY})(0.049)(0.03)(0.03) = 7.9\text{E-}8/\text{RY}$

The core-melt sequences that may lead to a PORV or SRV LOCA include the failure to trip the turbine following a reactor trip, feedwater underfeed, and failure to maintain core cooling and inventory makeup functions. The conditional probability of feedwater underfeed events is the sum of the probabilities of the sequences leading to plant state "UF" in Figure 2. These include sequences 3, 6, and 10 on Figure 2. Therefore, the total conditional probability of feedwater underfeed was estimated to be 0.011/demand. The core-melt frequency associated with PORV or SRV LOCAs is therefore:

- PWR: $(2.1\text{E-}3/\text{RY})(0.011)(0.044) = 1.0\text{E-}6/\text{RY}$
- BWR: $(1.8\text{E-}3/\text{RY})(0.011)(0.03) = 5.9\text{E-}7/\text{RY}$

BASE-CASE PUBLIC RISK ESTIMATES

The next step in this analysis was to combine the accident sequence frequencies above with corresponding containment failure probabilities. Containment failure probabilities were taken from Andrews, et al., (1983) for similar accident sequences where possible. It was observed that the small-break LOCA, SGTR, and RCP seal LOCA sequences are similar to sequence S₃FH in the Oconee PRA and sequence SI in the Grand Gulf PRA (see Appendices A and B, respectively, in Andrews, et al., 1983). It was more difficult to match sequences involving PTS and turbine missiles with those in Andrews, et al., (1983). Therefore, a conservative approach was taken that incorporated the most damaging (and highest offsite doses) containment failure modes. The containment failure modes selected were the those with the largest value of the failure mode probability times the consequences. The most damaging containment failure mode series for Oconee is the same as that for sequences S₃H. For Grand Gulf, the most damaging containment failure mode series is that for accident sequence T₂₃PQE.

The base-case affected public risk was determined by combining the accident sequence frequencies, containment failure probabilities, and dose consequences for the release categories corresponding to the appropriate containment failure modes. The dose consequence estimates given by Andrews, et al., (1983) for each release category were used in this analysis. The calculated base-case affected core-melt frequency and public risks are shown in the following tables.

EFFECTS OF PROPOSED SIR

The effects of safety issue resolution (SIR) would improve the probability that the turbine will trip on demand after a reactor trip. Therefore, the probability of event "TT" on Figure 1 will be reduced as a result of SIR. The SIR is also expected to result in reductions in the probabilities of other events on

Figure 1 because, primarily, of enhanced training and awareness of operators to respond to a reactor trip without turbine trip.

The adjusted-case value of parameter "TT" was taken from the analysis provided by Charlton (1988). The possible probabilities for TT that were provided by Charlton (1988) ranged from $1.0\text{E-}4$ to $3.6\text{E-}3$ with a best estimate value of $1.7\text{E-}3$. It is assumed that SIR will reduce the probability of TT to one-half the base case value or $8.5\text{E-}4/\text{demand}$. In addition, human error probabilities were assumed to be reduced by 50% as a result of enhanced training and operator awareness. These values were substituted into Figure 1 to calculate revised initiating event (i.e., reactor trip without turbine trip) frequencies. The adjusted-case event tree model and resulting turbine trip failure sequence frequencies are presented in Figure 4. The adjusted-case frequency for plant state TT-F is $6.2\text{E-}4/\text{RY}$ for PWRs and $5.4\text{E-}4/\text{RY}$ for BWRs. No changes were made to the feedwater and core cooling/makeup event trees shown in Figures 2 and 3. The adjusted-case initiating event frequencies were substituted into the calculations described previously to calculate the adjusted-case core melt frequencies and public risks. These calculations are shown on the pages following Figure 4.

Base-Case Core-Melt Frequency and Public Risk Calculations

<u>Sequence</u>	<u>Frequen- cy, RY⁻¹</u>	<u>Cont. Failure Mode</u>	<u>Proba- bility</u>	<u>Rel. Cate- gory</u>	<u>Conse- quence, Person-rem</u>	<u>Risk, Person rem/RY</u>
<u>PWRs</u>						
PTS	2.3E-9	<i>γ</i>	0.5	3	5.4E+6	6.2E-3
		<i>β</i>	0.0073	5	1.0E+6	1.7E-5
		<i>ε</i>	0.5	7	2.3E+3	<u>2.6E-6</u>
						6.2E-3
Turbine Missile	3.4E-7	<i>γ</i>	0.5	3	5.4E+6	9.2E-1
		<i>β</i>	0.0073	5	1.0E+6	2.3E-3
		<i>ε</i>	0.5	7	2.3E+3	<u>3.9E-4</u>
						9.2E-1
SGTR	3.5E-6	<i>γ</i>	0.5	3	5.4E+6	9.5E+0
		<i>β</i>	0.0073	5	1.0E+6	2.6E-2
		<i>ε</i>	0.5	7	2.3E+3	<u>4.0E-3</u>
						9.5E+0
PORV LOCA	1.0E-6	<i>γ</i>	0.5	3	5.4E+6	2.7E+0
		<i>β</i>	0.0073	5	1.0E+6	7.3E-3
		<i>ε</i>	0.5	7	2.3E+3	<u>1.2E-3</u>
						2.7E+0
RCP Seal LOCA	1.4E-7	<i>γ</i>	0.5	3	5.4E+6	3.8E-1
		<i>β</i>	0.0073	5	1.0E+6	1.0E-3
		<i>ε</i>	0.5	7	2.3E+3	<u>1.6E-4</u>
						3.8E-1
GRAND TOTAL	5.0E-6					1.3E+1

Base-Case Core-Melt Frequency and Public Risk Calculations (Continued)

<u>Sequence</u>	<u>Frequen- cy, RY⁻¹</u>	<u>Cont. Failure Mode</u>	<u>Proba- bility</u>	<u>Rel. Cate- gory</u>	<u>Conse- quence, Person-rem</u>	<u>Risk, Person rem/RY</u>
<u>BWRs</u>						
PTS	2.0E-9	γ	0.5	3	5.1E+6	5.1E-3
		δ	0.5	4	6.1E+5	<u>6.1E-4</u> 5.7E-3
Turbine Missile	2.9E-7	γ	0.5	3	5.1E+6	7.4E-1
		δ	0.5	4	6.1E+5	<u>8.8E-2</u> 8.3E-1
SRV LOCA	5.9E-7	γ	0.5	3	5.1E+6	1.6E+0
		δ	0.5	4	6.1E+5	<u>1.9E-1</u> 1.8E+0
RP Seal LOCA	7.9E-8	γ	0.5	3	5.1E+6	2.0E-1
		δ	0.5	4	6.1E+5	<u>2.4E-2</u> 2.2E-1
GRAND TOTAL	9.6E-7					2.9E+0

FIGURE 4: Turbine Trip Event Tree Model (Adjusted Case)

Adjusted-Case Core-Melt Frequency and Public Risk Calculations

<u>Sequence</u>	<u>Frequen- cy, RY⁻¹</u>	<u>Cont. Failure Mode</u>	<u>Proba- bility</u>	<u>Rel. Cate- gory</u>	<u>Conse- quence, Person-rem</u>	<u>Risk, Person rem/RY</u>
<u>PWRs</u>						
PTS	6.7E-10	γ	0.5	3	5.4E+6	1.8E-3
		β	0.0073	5	1.0E+6	4.9E-6
		ϵ	0.5	7	2.3E+3	<u>7.7E-7</u> 1.8E-3
Turbine Missile	9.9E-8	γ	0.5	3	5.4E+6	2.7E-1
		β	0.0073	5	1.0E+6	7.2E-4
		ϵ	0.5	7	2.3E+3	<u>1.1E-4</u> 2.7E-1
SGTR	1.4E-6	γ	0.5	3	5.4E+6	3.8E+0
		β	0.0073	5	1.0E+6	1.0E-2
		ϵ	0.5	7	2.3E+3	<u>1.6E-3</u> 3.8E+0
PORV LOCA	3.0E-7	γ	0.5	3	5.4E+6	8.1E-1
		β	0.0073	5	1.0E+6	2.2E-3
		ϵ	0.5	7	2.3E+3	<u>3.5E-4</u> 8.1E-1
RCP Seal LOCA	4.0E-8	γ	0.5	3	5.4E+6	1.1E-1
		β	0.0073	5	1.0E+6	2.9E-4
		ϵ	0.5	7	2.3E+3	<u>4.6E-5</u> 1.1E-1
GRAND TOTAL	1.8E-6					5.0E+0

Adjusted-Case Core-Melt Frequency and Public Risk Calculations (Continued)

<u>Sequence</u>	<u>Frequen- cy, RY⁻¹</u>	<u>Cont. Failure Mode</u>	<u>Proba- bility</u>	<u>Rel. Cate- gory</u>	<u>Conse- quence, Person-rem</u>	<u>Risk, Person rem/RY</u>
<u>BWRs</u>						
PTS	5.8E-10	γ	0.5	3	5.1E+6	1.5E-3
		δ	0.5	4	6.1E+5	<u>1.8E-4</u>
						1.7E-3
Turbine Missile	8.6E-8	γ	0.5	3	5.1E+6	2.2E-1
		δ	0.5	4	6.1E+5	<u>2.6E-2</u>
						2.5E-1
SRV LOCA	1.8E-7	γ	0.5	3	5.4E+6	4.6E-1
		δ	0.5	4	6.1E+5	<u>5.5E-2</u>
						5.1E-1
RP Seal LOCA	2.4E-8	γ	0.5	3	5.1E+6	6.1E-2
		δ	0.5	4	6.1E+5	<u>7.3E-3</u>
						6.8E-2
GRAND TOTAL	2.9E-7					8.3E-1

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

SCRAM Without a Turbine/Generator Trip (144)

2. Affected Plants (N):

	<u>N</u>
PWRs	47
BWRs	24
All plants	71

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

	<u>$\bar{T}$</u>
PWRs	27.7 yr
BWRs	25.2 yr
All plants	26.9 yr

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

$$\Delta(\bar{F}D_R)_{PWR} = (19860 \text{ person-rem})(3.2E-6/R\bar{Y}) = 6.4E-2 \text{ person-rem/R}\bar{Y}$$

$$\Delta(\bar{F}D_R)_{BWR} = (19860 \text{ person-rem})(6.7E-7/R\bar{Y}) = 1.3E-2 \text{ person-rem/R}\bar{Y}$$

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

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
91	850	0

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

As explained in Attachment 2 to Table 3, a total of 1.25 man-wk/plant of utility labor in radiation zones was estimated to be required for SIR implementation. This estimate applies to all plants.

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

$$\begin{aligned} D &= (1.25 \text{ man-wk/plant})(40 \text{ man-hr/man-wk})(2.5E-3 \text{ rem/hr}) \\ &= 0.125 \text{ person-rem/plant} \end{aligned}$$

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

$$ND = (71 \text{ plants})(0.125 \text{ person-rem/plant}) = 8.9 \text{ person-rem}$$

9-11 Steps Related to Per-Plant Utility Labor in Radiation Zones for SIR Operation and Maintenance:

No additional utility labor in radiation zones is anticipated for SIR operation and maintenance. Therefore, D_o and NTD_o are 0.0.

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
9	27	3

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

SCRAM Without a Turbine/Generator Trip (144)

2. Affected Plants (N):

	<u>N</u>
PWRs	47
BWRs	24
All Plants	71

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

	<u>$\bar{T}$</u>
PWRs	27.7 yr
BWRs	25.2 yr
All plants	26.9 yr

Industry Costs (Steps 4 through 12):

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

$$\Delta(\bar{F}A)_{PWR} = (\$1.65E+9)(3.2E-6/R\bar{Y}) = \$ 5.3E+3/R\bar{Y}$$

$$\Delta(\bar{F}A)_{BWR} = (\$1.65E+9)(6.7E-7/R\bar{Y}) = \$ 1.1E+3/R\bar{Y}$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$7.5E+6	\$7.0E+7	0

6. Per-Plant Industry Resources for SIR Implementation:

As described in Attachment 2, industry resources for SIR implementation include hardware costs and labor for installation of improved emergency turbine trip systems, development of revised operating and maintenance procedures, development of and attendance at training courses relative to turbine trip systems, and for revisiting design-basis analyses. The per-plant industry resources for these items are as follows:

Hardware

• 2 36-in valves at \$ 70,000/valve	\$140,000
• Valve control system	<u>\$ 10,000</u>
TOTAL:	<u>\$150,000</u>

Labor Requirements

• Install enhanced turbine trip systems	22.5 man-wk
• Training	4 man-wk
• Design-basis analyses	<u>24 man-wk</u>
TOTAL:	<u>50.5 man-wk</u>

Procedure Changes 2 at \$3900/procedure \$7800

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

$$\begin{aligned} I &= (\$150,000) + (50.5 \text{ man-wk/plant})(\$2270/\text{man-wk}) + (\$7800) \\ &= \$2.7\text{E}+5/\text{plant} \end{aligned}$$

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

$$NI = (\$2.7\text{E}+5/\text{plant})(71 \text{ plants}) = \$1.9\text{E}+7$$

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

As discussed in Attachment 2, annual operation and maintenance labor requirements were estimated to be 4 man-day/Ry for reviewing procedures and an additional 1 man-wk/Ry for periodic refresher training. Therefore, a total of 1.8 man-wk/Ry of industry labor was estimated to be needed for SIR operation and maintenance.

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

$$I_o = (1.8 \text{ man-wk/Ry})(\$2270/\text{man-wk}) = \$ 4086/\text{Ry}$$

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

$$\begin{aligned} NTI_o &= (\$ 4086/\text{Ry})[(47 \text{ PWRs})(27.7 \text{ yr}) + (24 \text{ BWRs})(25.3 \text{ yr})] \\ &= \$ 7.9\text{E}+6 \end{aligned}$$

12. Total Industry Cost (S_I):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$2.7E+7	\$3.7E+7	\$1.7E+7

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC resources for SIR development were estimated in Attachment 2 to be approximately 0.55 man-yr of contractor and NRC technical monitor labor.

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

$$C_D = (0.55 \text{ man-yr})(\$100,000/\text{man-yr}) = \$5.5E+4$$

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

As discussed in Attachment 2, a total of 5 man-wk/plant of NRC labor was estimated to be required for support of SIR implementation.

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

$$C = (5 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$1.1E+4/\text{plant}$$

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

$$NC = (\$1.1E+4/\text{plant})(71 \text{ plants}) = \$7.8E+5$$

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

NRC resource requirements for review of SIR implementation were estimated to be 3 man-days/RY or 0.6 man-wk/RY (See Attachment 2).

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

$$C_o = (0.6 \text{ man-wk/RY})(\$2270/\text{man-wk}) = \$1360/\text{RY}$$

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

$$\begin{aligned} N\bar{T}C_o &= (\$1360/\text{RY})[(71 \text{ PWRs})(27.7 \text{ yr}) + (24 \text{ BWRs})(25.2 \text{ yr})] \\ &= \$2.6E+6 \end{aligned}$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$3.4E+6	\$4.8E+6	\$2.1E+6

ATTACHMENT 2 (To Table 3)

This attachment describes the basis for the industry and NRC cost estimates presented in Table 3. As described previously, SIR involves four activities:

- Design and installation of redundant and diverse turbine trip systems.
- Improved test/maintenance procedures to enhance the reliability of turbine trip systems.
- Enhanced operating and maintenance procedures and associated training. For example, ensure operators are aware of potential catastrophic failures that may result from turbine over-speed conditions caused by tripping the generator.
- Revisit design-basis calculations to assure that transients involving a turbine/generator failure to trip do not produce more severe conditions than those previously analyzed and to support the operating procedures and training enhancements described above.

Industry Costs

The first activity is assumed to require installation of 2 solenoid actuated emergency trip valves downstream of the existing turbine trip signals. These are anticipated to be large valves because of the size of the main steam lines. Sciacca (1989) provides estimated capital costs and labor hours for installation of valves up to 36-in. diameter. The capital costs were estimated at \$70,000/valve and the associated installation labor was estimated at 424 man-hr/valve. In addition, valve controls and associated hardware must be added to the control room and linked to the valves. The costs of the valve control system are estimated at about \$10,000/valve and are estimated to require approximately 1 man-wk to install. Since installation of the valve controls will occur in a radiation zone, a labor productivity factor of 0.8 was applied. The total installation labor for the valve controls is therefore estimated at 1.25 man-wks/plant. All of these modifications are assumed to be implemented during scheduled plant outages and therefore no additional plant downtime is anticipated.

Improved maintenance and testing of the turbine trip systems were assumed to be required on a six-month interval. It was estimated that these activities can be performed in about 2 man-days. Therefore, the total annual test/maintenance requirements for this system was estimated to be 4 man-days/RV.

Operating procedures and training costs were taken from Sciacca (1989). It was assumed that one revised operating procedure and one revised maintenance procedure will be required. The costs for "complex" operating and maintenance procedures were used in this analysis (\$3,900/procedure). To this must be added to costs of initial training of operators in the new procedures and periodic refresher training. It was estimated that initial training would require approximately 4 man-wk/reactor to develop training materials and perform the

actual training. An additional 1 man-wk/Ry was estimated to be needed for periodic refresher training.

The fourth activity called for in the SIR is to revisit the design-basis analyses in each plant's Safety Analysis Report (SAR) to ensure that turbine trip failure events do not result in more severe conditions than those previously analyzed. These are relatively complex calculations that will require the use of large thermal-hydraulic computer codes. It was estimated that approximately 6 man-month of effort would be needed to perform these analyses, including satisfying quality assurance requirements.

NRC Costs

NRC costs to develop the SIR were estimated to include a study to review the various turbine trip systems and develop recommendations on improved systems. This study would include reviewing plant designs, researching ways to improve the turbine trip reliability, and developing recommendations on how each plant could improve the reliability of their turbine trip systems. It was estimated that this study could be completed with approximately 6 man-months (0.5 man-yr) of contractor resources. An additional 10% (0.05 man-yr) was added to account for an NRC technical monitor for the contractor.

NRC costs for support of SIR implementation include resources to review plant-specific designs for the enhanced turbine trip systems, to review proposed revisions to operating and maintenance procedures, and to review the design basis analyses. It was estimated that approximately 2 man-wks/plant of NRC resources are needed to review plant-specific turbine trip system designs, 1 man-wk/plant to review the revised procedures, and 2 man-wk/plant are needed to review the design basis analyses. Therefore, a total of 5 man-wk/plant of NRC labor are required to support SIR implementation.

NRC costs to review SIR operation and maintenance, including reviews of trend analyses on the enhanced turbine trip systems (where appropriate), and reviews of operating and maintenance records, are anticipated to be minimal. A total of 3 man-days/Ry was estimated for these activities.

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ISSUE SUMMARY WORKSHEET

ISSUE NO./TITLE: 147, Fire-Induced Alternate Shutdown/Control Room Panel Interactions

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: This issue relates to the potential for control system vulnerabilities as a result of fire induced alternate shutdown-control room panel interactions. The proposed resolution involves the development of industry guidance for evaluating remote shutdown capability and the utilization of this guidance to investigate likely sources of control system interactions between the control room, remote shutdown panel, and shutdown systems during fire conditions. Affected plants would be required to identify current system vulnerabilities and execute permanent corrective actions through the implementation of equipment design and/or procedural changes.

AFFECTED PLANTS: All plants (BWRs - 44; PWRs - 90)

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 11,000

OCCUPATIONAL DOSES:

SIR Implementation =	20
SIR Operation/Maintenance =	0
Total of Above =	20
Accident Avoidance =	210

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	6.0
SIR Operation/Maintenance =	4.3
Total of Above =	10.3
Accident Avoidance =	18

NRC COSTS:

SIR Development =	0.11
SIR Implementation =	0.40
SIR Operation/Maintenance =	1.7
Total of Above =	2.2

FIRE INDUCED ALTERNATE SHUTDOWN-CONTROL ROOM PANEL INTERACTIONS

ISSUE 147

1.0 SAFETY ISSUE DESCRIPTION

This issue relates to the potential for control system vulnerabilities as a result of fire induced alternate shutdown-control room panel interactions. Concern over these proposed interactions arose as the product of an NRC sponsored Fire Protection Research Program in conjunction with the Risk Methodology Integration and Evaluation Program (RMIEP) which was focused on current fire protection practices for control rooms, remote shutdown areas, control transfer areas and local control areas.

As a result of the fire analysis studies (Lambright, et al., 1989), a number of potential control systems vulnerabilities were identified as areas of concern, including the following:

- the loss of control power for the operated device as the result of blown fuses.
- the spurious actuation of one or more safety-related or safe shutdown related components as the result of fire induced cable faults or component failures.
- total loss of system function as the result of fire induced redundant component failure or electrical distribution system failure.

PROPOSED RESOLUTION

The proposed safety issue resolution involves the development of guidance for evaluating remote shutdown capability. This guidance would be utilized by all individual nuclear utilities in investigating likely sources of control system interactions between the control room, remote shutdown panel, and shutdown systems given a fire at any plant location.

Current remote shutdown implementation practices for system interaction vulnerabilities would be reviewed with consideration given to the following:

- electrical independence
- loss of control power before transfer
- damage to transfer switch as leading to loss of control of associated equipment from both the main control room and remote shutdown panel
- spurious actuation of components leading to damage
- spurious opening of valves leading to LOCA

- spurious opening of high pressure/low pressure interface valves

Upon completion of the review, identified vulnerabilities would be resolved through equipment and/or procedural changes. Specific training would be given to operators to ensure awareness of procedures to respond to a fire, particularly in alternative ways of starting high pressure injection systems or depressurizing in combination with starting low pressure injection systems. Operators would also be trained to start potential alternative systems that could be used to inject coolant (e.g., condensate systems).

AFFECTED PLANTS

Resolution of this safety issue is assumed to affect all nuclear plants that are in operation or under construction.

2.0 SAFETY ISSUE RISK AND DOSE

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

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

Fire Induced Alternate Shutdown-Control Room Panel Interactions (147)

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

All PWRs and BWRs are assumed to be affected.

	<u>N</u>	<u>$\bar{T}$ (yr)</u>
PWRs	90	28.8
BWRs	<u>44</u>	<u>27.4</u>
All	134	28.3

3. Plants Selected for Analysis:

Representative plant: LaSalle 2

Results for PWRs are assumed to be equivalent to the results derived in this analysis for BWRs.

4. Parameters Affected by SIR:

Based on definitions of the basic event descriptions, material presented in NUREG/CR-5088 (Lambright, et al., 1989), and discussions with the authors of NUREG/CR-5088, the parameters listed in Attachment 1 are affected.

5. Base-Case Values for Affected Parameters:

The base-case values of the affected parameters are listed below:

RA-FIRE-3-12-80M = 4.0E-3/RY
OPFAILSCDS-OE = 1.0/RY
RA-FIRE-3-12-1H = 6.7E-2/RY

See Attachment 1.

6. Affected Accident Sequences and Base-Case Frequencies:

The accident sequence applied in this analysis was generated using the LaSalle event sequence 8 described in NUREG/CR-5088. A new accident sequence, designated "FIRE," will form the basis for this analysis. This scenario involved a control room fire, consuming the Emergency Core Cooling System and Electrical Distribution Panels with their controls

going to the most undesired state. This event included the failure of all high pressure systems, successful reactor depressurization, and failure of all low pressure systems. The base-case frequencies of the affected accident sequence are:

FIRE	γ (BWR-3)	7.7E-7/RY
	δ (BWR-5)	6.9E-6/RY
		<u>7.7E-6/RY</u>

See Attachment 1 for additional details.

7. Affected Release Categories and Base-Case Frequencies:

<u>Release Category</u>	<u>Frequency, RY⁻¹</u>
BWR-3	7.7E-7
BWR-5	6.9E-6
	<u>7.7E-6</u>

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

$$\bar{F}_{\text{BWR}} = 7.7\text{E-6 /RY} \quad \bar{F}_{\text{PWR}} = 7.7\text{E-6/RY}^{(a)}$$

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

$$W_{\text{BWR}} = 8.1 \text{ person-rem/RY} \quad W_{\text{PWR}} = 8.1 \text{ person-rem/RY}^{(a)}$$

10. Adjusted-Case Values for Affected Parameters:

The adjusted-case values for the affected parameters are:

$$\begin{aligned} \text{RA-FIRE-3-12-80M} &= 2.8\text{E-3/RY} \\ \text{OPFAILSCDS-OE} &= 2.0\text{E-3/RY} \\ \text{RA-FIRE-3-12-1H} &= 4.7\text{E-2/RY} \end{aligned}$$

See Attachment 1 for details of their derivation.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

FIRE	γ (BWR-3)	4.91E-7/RY
	δ (BWR-5)	4.42E-6/RY
		<u>4.91E-6/RY</u>

12. Affected Release Categories and Adjusted-Case Frequencies:

<u>Release Category</u>	<u>Frequency, RY⁻¹</u>
BWR-3	4.91E-7
BWR-5	4.42E-6
	<u>4.91E-6</u>

(a) The base-case frequencies and public risks for FIRE sequences at PWRs were assumed to be equal to the values derived in this analysis for BWRs.

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

$$\bar{F}_{BWR}^* = 4.91E-6/RY$$

$$\bar{F}_{PWR}^* = 4.91E-6/RY$$

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

$$W_{BWR}^* = 5.2 \text{ person-rem/RY}$$

$$W_{PWR}^* = 5.2 \text{ person-rem/RY}$$

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

$$\Delta\bar{F}_{BWR} = 2.8E-6/RY$$

$$\Delta\bar{F}_{PWR} = 2.8E-6/RY$$

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

$$\Delta W_{BWR} = 2.9 \text{ person-rem/RY}$$

$$\Delta W_{PWR} = 2.9 \text{ person-rem/RY}$$

17. Total Public Risk Reduction, (ΔW)_{Total}:

<u>Best Estimate</u> <u>person-rem</u>	<u>Error bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
1.1E+4	9.2E+5	0

ATTACHMENT 1 (To Table 1)

The analysis performed for the fire-induced alternate shutdown-control room panel interactions was based primarily on information obtained from communication with Sandia National Laboratory and from NUREG/CR-5088, Fire Risk Scoping Study (Lambright, et al., 1989). The accident sequence was generated using the LaSalle event sequence number 8 and initiating event frequency ($2.2\text{E-}4/\text{yr}$) as described in NUREG/CR-5088. This scenario involved a control room fire, consuming the Emergency Core Cooling System and Electrical Distribution Panels with their controls going to the most undesired state. This event included the failure of all high pressure systems, successful reactor depressurization, and failure of all low pressure systems.

Minimal cut-set data, basic event descriptions and base case frequencies were obtained from Sandia. The accident sequence was comprised of approximately 350 minimal cut sets. All of the cut sets included fire-induced failures of one or more components in addition to an operator recovery term and one or more independent failures. The fire-induced failures were assigned values of one (1.0) in the Sandia study because their control stations were located in the fire zone.

The parameters that would be affected by the proposed resolution and their base-case values are given below:

RA-FIRE-3-12-80M = Recovery action: 80 minutes for the operators to decide and take action and recover the system and/or component of interest ($4\text{E-}3$).

OPFAILSCDS-OE = Operator fails to use the condensate system (1).

RA-FIRE-3-12-1H = Recovery action: 1 hour for the operators to decide and take action and recover the system and/or component of interest ($6.7\text{E-}2$).

Independent failures of the high pressure core spray pump and standby component cooling water system were assumed to be unaffected by issue resolution. Issue resolution also does not affect the frequency of fire initiating events. Resolution of this issue is assumed to affect the likelihood that operators take appropriate actions to recover from a fire, including actuation of systems that are not part of the emergency core cooling capabilities (e.g., condensate system).

This information was used to calculate a base case core-melt frequency for fire initiated events of $(2.2\text{E-}4/\text{RY}) \times (3.5\text{E-}2) = 7.7\text{E-}6/\text{RY}$. The value of $3.5\text{E-}2$ is the probability of failure of high and low pressure systems given that the control room fire has occurred. This value was derived from the minimal cut set data provided by Sandia. This frequency was calculated based on the assumption that any basic events representing systems or components in the fire zone of interest would not function for the duration of the accident.

The evaluation of adjusted-case frequencies focused on affected parameters relating to operator recovery actions. Basic event probabilities involving operator actions and/or decisions were reduced by 30% to obtain adjusted-case values. This reduction would result from improved procedures and training

relative to recovery from fires. Operator failure to use alternative coolant injection systems, such as the condensate system, was reduced from a frequency of 1.0/demand to a frequency of $2.0\text{E-}3/\text{demand}$ based on past practice and engineering judgement. This large decrease is believed to be reasonable because resolution of the issue would include development of procedures and training to alert operators to potential backup sources of cooling water. These procedures are not currently in place as the failure probability was assumed by Sandia to be 1.0 (see base-case values of affected parameters).

The adjusted-case values for all the affected parameters are shown on Table 1. These adjusted-case values were substituted into the minimal cut-sets provided by Sandia to calculate the adjusted-case core-melt frequency for accidents initiated by a fire. The adjusted-case core-melt frequency was estimated to be $4.91\text{E-}6/\text{RY}$.

The next step in the calculation of the change in public risks associated with SIR was to multiply the core melt frequencies given above by the probabilities of containment failure given the accident has occurred. The containment failure probabilities for the fire-initiated core-melt sequences were not in the scope of the analysis performed by Lambright, et al. (1989). Therefore, the fire risk analysis performed in the Oconee PRA (Sugnet, et al., 1984) was used as the basis for these parameters. It is recognized that the FIRE sequences are for a BWR and Oconee 3 is a PWR. However, the Oconee PRA was the only available PRA that addressed fires in sufficient detail to match the accident sequences generated in this analysis with containment failure modes. Therefore, it was necessary to use the Oconee PRA in spite of the problem related to the comparability of PWRs and BWRs.

All the fire-initiated accident sequences investigated by Sugnet et al. were placed in one of two plant damage bins. Of these two bins, the most severe type of containment failure was estimated to occur 10% of the time following a core-melt accident and the least severe occurs 90% of the time. Therefore, the containment failure probabilities used in this analysis are 0.1 and 0.9. The release categories that correspond to these containment failure probabilities are discussed below.

For this analysis, it was judged that the containment failure scenario from fires are similar to those described for BWR-3 and BWR-4 release categories as described by Andrews, et al., (1983). This was because BWR-1 category releases result primarily from core-melt sequences involving failures of safety-relief valves (SRVs) to reseal and failure of residual heat removal (RHR) systems after a loss of coolant accident (LOCA). Neither of these scenarios are consistent with the fire scenario. BWR-2 category releases result from accident sequences involving failure of the power conversion system, failure to render the reactor subcritical, and failure of the SRVs to reseal. Again, these are not consistent with the core melt sequences given by Lambright, et al., (1989). BWR-3 and BWR-4 release categories include core-melt sequences involving transients followed by loss of the emergency core cooling systems, high pressure core spray, reactor core isolation cooling, and low pressure core cooling systems. These core melt scenarios are consistent with those presented by Lambright, et al., (1989). As a result, the containment failure modes and consequences for BWR-3 and BWR-4 release categories will be used for the fire-initiated accident sequences. The containment failure probabilities for these release sequences are 0.1 for the

BWR-3 release category and 0.9 for BWR-4, based on comparisons of system failures as given by Lambright, et al., with those given by Sugnet, et al.

TABLE 2. Occupational Dose Worksheet

1. Title and Identification Number of Safety Issue:

Fire Induced Alternate Shutdown-Control Room Panel Interactions (147)

2. Affected Plants (N)

All PWRs and BWRs are assumed to be affected.

	<u>All plants</u>	<u>Backfit</u>	<u>Forward-fit</u>
PWRs	90	47	43
BWRs	44	24	20

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

	<u>$\bar{T}$</u>		
	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	27.7	30	28.8
BWRs	25.2	30	27.4

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

$$\Delta(\bar{F}D_R) = (19,860 \text{ person-rem})(2.79\text{E-}6/\text{RY}) = 5.5\text{E-}2 \text{ person-rem/RY}$$

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

<u>Best Estimate</u>	<u>Error Bounds (man-rem)</u>	
<u>(man-rem)</u>	<u>Upper</u>	<u>Lower</u>
2.1E+2	3.5E+3	0

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

As discussed in Attachment 2, SIR implementation involves a 2.7 man-wk walk-down inspection in radiation zones at all plants. However, radiation doses will only be incurred at backfit plants. In addition, SIR includes installation of modified hardware in 10% of the backfit plants. It was estimated that an additional 2.7 man-wks of labor in radiation zones is needed to complete the hardware modifications.

All plants : 2.7 man-wks (doses applicable only at backfit plants)
10% of backfit plants: 2.7 man-wks

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

The dose rates in areas in which the walk-down inspection crew and hardware installation crew activities will be performed are assumed to be 2.5 mrem/hr.

$$D = (2.7 \text{ man-wks/plant})(40 \text{ hrs/man-wk})(2.5 \text{ mrem/hr}) \\ = 0.27 \text{ person-rem/plant}$$

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

$$D = (71 \text{ plants})(0.27 \text{ person-rem/plant}) + (7 \text{ plants})(0.27 \text{ person-rem/plant}) \\ = 20.1 \text{ person-rem}$$

9-11. Steps for Estimating Occupational Doses Associated with SIR Operation and Maintenance:

SIR involves no labor in radiation zones for SIR operation and maintenance. Therefore, the occupational doses for operation and maintenance of the proposed SIR are 0.

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds</u> <u>Upper</u>	<u>(man-rem)</u> <u>Lower</u>
20	60	0

3.0 SAFETY ISSUE COSTS

The industry and NRC costs associated with resolution of this safety issue are estimated in this section. The results are summarized in Table 3.

TABLE 3 Safety Issue Cost Worksheet

1. Title and Identification Number of Safety Issue:

Fire Induced Alternate Shutdown-Control Room Panel Interactions (147)

2. Affected Plants (N):

All PWRs and BWRs are assumed to be affected.

	<u>All plants</u>	<u>Backfit</u>	<u>Forward-fit</u>
PWRs	90	47	43
BWRs	44	24	20

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

	<u>$\bar{T}$</u>		
	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	27.7	30	28.8
BWRs	25.2	30	27.4

Industry Costs (Steps 4 through 12):

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

$$\Delta\bar{F}A = (\$1.65E+9)(2.79E-6/R\bar{Y}) = \$4.6E+3/R\bar{Y}$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.8E+7	\$2.9E+8	0

6. Per-Plant Industry Resources for SIR Implementation:

Attachment 2 describes the following cost elements:

- All plants evaluate remote-shutdown capability: 18.7 man-wks
- Backfit plants require procedural changes: \$900/plant
- Hardware modifications at 10% of backfit plants
 - Design, safety analysis, QA: 4 man-wks
 - Hardware procurement/installation: \$30,000

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

- All plants : (18.7 man-wks)(\$2270/man-wk) = \$42,400/plant
- Backfit plants: = \$900/plant
- 10% of backfit plants: (4 man-wks)(\$2270/man-wk) + \$30,000
= \$39,080

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

$$\begin{aligned} NI &= (134 \text{ plants})(\$42,400) + (71 \text{ plants})(\$900) + (7 \text{ plants})(\$39,100) \\ &= \$6.0\text{E}+6 \end{aligned}$$

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

Industry labor for SIR operation and maintenance consists of an estimated 0.5 man-wk/Ry for an annual review of remote shutdown systems in relation to other plant modifications and/or regulatory changes that have occurred in the previous year. See Attachment 2.

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

$$\begin{aligned} I_o &= (0.5 \text{ person-wk/Ry})(\$2270/\text{person-wk}) \\ &= \$1140/\text{Ry} \end{aligned}$$

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

$$\begin{aligned} NTI_o &= [(90 \text{ PWRs})(28.8 \text{ yr}) + (44 \text{ BWRs})(27.4 \text{ yr})](\$1140/\text{Ry}) \\ &= \$4.3\text{E}+6 \end{aligned}$$

12. Total Industry Cost (S_T):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.0E+7	\$1.7E+7	\$2.6E+6

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

As described in Attachment 2, NRC resources for SIR development are estimated to include 1 man-yr of NRC contractor labor and 1 man-mont of NRC labor to monitor the contractor.

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

$$C_o = (\$100,000) + (4 \text{ man-wks})(\$2270/\text{man-wk}) = \$1.1\text{E}+5$$

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

The following resource requirements are described in Attachment 2:

- Review of remote shutdown capability evaluation: 1.0 man-wk
- Review of procedure modifications: 0.4 man-wk
- Review of hardware modifications: 2.0 man-wk

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

- Review of remote shutdown capability evaluation
= (1 man-wk/plant)(\$2270/man-wk) = \$2270/plant
- Review of procedure modifications
= (0.4 man-wk/plant)(\$2270/man-wk) = \$908/plant
- Review of hardware modifications
= (2 man-wk/plant)(\$ 2270/man-wk) = \$4540/plant

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

$$\begin{aligned} \text{NC} &= (134 \text{ plants})(\$2270/\text{plant}) + (71 \text{ plants})(\$908/\text{plant}) \\ &\quad + (7 \text{ plants})(\$4540/\text{plant}) \\ &= \$4.0\text{E}+5 \end{aligned}$$

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

As discussed in Attachment 2, the per-plant NRC resource requirements for this element were estimated to be 0.2 man-wk/Ry.

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

$$C_o = (0.2 \text{ man-wk/Ry})(\$2270/\text{man-wk}) = \$450/\text{Ry}$$

20. Total NRC Cost for Review of SIR Operation and/Maintenance ($\overline{\text{N}}\text{TC}_o$):

$$\overline{\text{N}}\text{TC}_o = (134 \text{ plants})(28.3 \text{ yr})(\$450/\text{Ry}) = \$1.7\text{E}+6$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$2.2E+6	\$4.0E+6	\$4.5E+5

ATTACHMENT 2 (to Table 3)

The proposed safety issue resolution (SIR) involves the development of guidance for evaluating remote shutdown capability. The guidance would consider, at minimum, the following potential vulnerabilities:

- electrical independence
- loss of control power before transfer
- damage to transfer switch as leading to loss of control of associated equipment from both the main control room and remote shutdown panel
- spurious actuation of components leading to damage
- spurious opening of valves leading to LOCA
- spurious opening of high pressure/low pressure interface valves

Upon completion of the review, identified vulnerabilities would be resolved through equipment and/or procedural changes. The following paragraphs address the NRC and industry resources and costs necessary to address this issue.

The first activity related to the proposed SIR is for the NRC to develop the guidance necessary for utilities to evaluate the remote shutdown capabilities of their plants. This is anticipated to be a substantial effort because some consideration needs to be given to differences in plant designs. It is estimated that development of this guidance would require approximately 1 man-yr of effort for an NRC contractor plus about 1 man-month of NRC labor to monitor the contractor.

The next activity would be for the industry to evaluate the remote shutdown capabilities of their plants using the guidance developed above. This is anticipated to be a significant effort for each plant because an extensive review of the control room design, remote shutdown panel design, and associated electrical and instrument connections is expected to be needed. In addition, a significant effort is likely to be needed to evaluate procedures associated with the remote shutdown system. A third major activity, which would involve some occupational doses, would include a comprehensive walkdown inspection of the remote shutdown system, including connections to the plant hardware. The estimated resource requirements for these three activities are:

1)	Evaluate plant designs	8 man-wks/plant
2)	Evaluate remote shutdown procedures	8 man-wks/plant
3)	Walk down inspection	2 man-wks/plant ^(a)

It is anticipated that, as a result of the industry's implementation of the SIR, some plants will require modifications to procedures and/or plant equipment to reduce their vulnerability to this issue. It is not possible at this time to accurately estimate the number of plants that will require procedural and/or hardware modifications. For this analysis, it is assumed that all backfit plants will require one minor procedural modification and that 10% of the backfit plants will require modifications to hardware. According to NUREG/CR-4627 (SEA 1986),

(a) A 75% efficiency factor was applied to labor in radiation zones. Thus, the actual labor requirement for this activity is 2.7 man-wks/plant.

the per-plant cost for a minor procedural change is approximately \$900. For the hardware modifications, it was assumed that procurement and installation costs are a nominal \$30,000 and that 4 man-wks of industry labor are required for associated design studies, safety analyses, and quality assurance activities. Hardware procurement and installation costs include a nominal 2 man-wks in radiation zones.^(a) The final industry cost item is for an annual review of procedure/hardware modifications in relation to potential changes in plant configurations and/or NRC regulations. It was assumed that the incremental costs of procedural changes and hardware modifications to forward-fit plants are zero; i.e., these modifications can be made during construction without additional costs. However, the costs for evaluation of designs, procedures, and the walk-down inspection are incremental costs for forward-fit plants.

Costs will be incurred by the NRC for reviewing the evaluations of each plant, reviewing proposed procedural changes, and reviewing the hardware modifications. It is estimated that about 1 man-wk of NRC labor will be required to review the remote shutdown capability evaluation of each plant. Reviews of minor procedural changes are not necessarily resource-intensive and are estimated to require 2 man-days (0.4 man-wks) of NRC labor. NRC reviews of the plants requiring hardware modifications will require much more substantial resources to review design changes, safety analyses, and QA documentation. It is estimated that approximately 2 man-wks/plant is necessary for these reviews.

(a) A 75% efficiency factor was applied to labor in radiation zones. Thus, the actual labor requirement for this activity is 2.7 man-wks/plant.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 148, Smoke Control and Manual Firefighting Effectiveness

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: A generic safety issue has been identified concerning the effectiveness of smoke control and manual firefighting effectiveness in nuclear power plants. The issue revolves around the difficulties that smoke may cause in terms of hampering manual firefighting efforts, damaging or degrading electronic equipment, interfering with normal plant shutdown, and initiating automatic suppression systems that could damage safety systems. The proposed safety issue resolution (SIR) includes development of guidance in developing fire response plans, searches for plant-specific vulnerabilities, and selection and installation of proper smoke removal systems as well as detection and suppression systems.

AFFECTED PLANTS: This issue affects all operating and planned plants.

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION =	1.1E+2
OCCUPATIONAL DOSES:	
SIR Implementation =	1.2E+2
SIR Operation/Maintenance =	1.9E+1
Total of Above =	1.4E+2
Accident Avoidance =	9.1E+1

COST RESULTS (\$10⁶):

INDUSTRY COSTS:	
SIR Implementation =	1.3E+1
SIR Operation/Maintenance =	1.7E+0
Total of Above =	1.5E+1
Accident Avoidance =	7.5E+0
NRC COSTS:	
SIR Development =	2.1E-1
SIR Implementation =	3.6E+0
SIR Operation/Maintenance Review =	1.7E+0
Total of Above =	5.5E+0

SMOKE CONTROL AND MANUAL FIREFIGHTING EFFECTIVENESS

ISSUE 148

1.0 SAFETY ISSUE DESCRIPTION

This generic safety issue is concerned with the effectiveness of smoke control and manual firefighting measures in nuclear power plants. NRC regulations regarding fire protection are found in 10 CFR 50, Appendix R. Specific fire protection guidance is found in the Standard Review Plan. The objectives of these requirements are to prevent fires from starting, detect and suppress rapidly those fires that do occur, and to protect important plant safety-related features from both the effects of fires and the effects of fire suppression activities. To accomplish these objectives, nuclear power plants have established fire protection programs designed to prevent or limit the effects of fires through appropriate administrative controls, design features, procedures, and training.

The following four concerns are of primary concern to this issue:

- smoke may cause difficulties in terms of reduced visibility which may hamper manual firefighting efforts or cause misdirected suppression efforts
- smoke may damage or degrade electronic equipment
- smoke may interfere with normal plant shutdown by causing evacuation of control centers,
- smoke may cause the initiation of automatic suppression systems that could damage safety systems.

Each of these concerns were addressed in a recent NRC memorandum (McCracken 1991). The key points from this memorandum are summarized below.

The fire protection program at each plant includes elements designed to address each of these concerns. The combination of the following fire protection program elements directly or indirectly are important to alleviating the four concerns listed above: training of the fire brigade; limitations on the flammability and quantities of flammable materials in safety-related areas; rapid detection of fires; and separation of redundant safety-related systems that reduces the likelihood of fire propagation from one compartment to another. Examples of these elements are discussed in the following paragraphs.

Fire brigades are trained to perform fire fighting tasks under low visibility and high heat conditions. They are trained in the effective use of water to fight fires and the proper methods for fighting fires in buildings and in confined spaces. Their training includes many different types of fires, including fires involving cables, hydrogen, combustible and flammable liquids and hazardous chemicals, and record files.

Administrative controls are implemented for such items as limits on quantities of transient fuels available, limits on ignition sources, etc. Guidance is also applicable to the types of materials used in the plants. Examples of this guidance include the following:

- Cable trays and conduits are required to be constructed of metallic materials.
- Cable insulation must meet flame test requirements specified by the Institute of Electrical and Electronic Engineers (IEEE).

Additional requirements have been imposed to ensure rapid detection and suppression of fires in safety-related areas. Examples include:

- Areas with high cable concentrations must be protected by automatic fire suppression systems and are provided with early warning smoke detection capability.
- Areas containing safety-related cabling must be provided with area smoke detection capability. This capability must be alarmed and annunciated in the control room.
- Diesel generator cells are required to be provided with an automatic fire suppression system designed to combat lubricating oil fires.
- Manual fire suppression capability (hose stations and portable fire extinguishers) are required to be provided and accessible to all areas in the diesel generator cell.

Examples of design features that are applicable to the fire protection program include:

- Diesel generator cells are required to be separated from each other and the other areas of the plant by fire barriers having a 3-hr fire rating.
- The bulk diesel fuel storage facility is required to be located outside a minimum of 50 ft. away from the nearest exposed structure.
- Cable and cable-tray penetrations of fire barriers are required to be sealed with fire-resistive material.

Section III.L of 10 CFR 50, Appendix R, requires that the control room be provided with alternate shutdown capability which is physically and electrically independent of the control room. Plants have procedures to implement plant shutdown from outside the control room and contingency plans that address fire brigade actions in response to a remote shutdown.

In consideration of the previous discussion, it appears that an effective fire protection program exists at each nuclear power plant. However, in response to the specific concerns raised by this issue, a proposed resolution and the risk and cost information will be developed.

PROPOSED RESOLUTION

The proposed safety issue resolution (SIR) includes the following key items:

- NRC development of guidance in developing fire response plans,
- searches for plant-specific vulnerabilities to be performed by licensees
- selection and installation of proper smoke removal systems as well as detection and suppression systems in areas determined to be vulnerable to fires.

AFFECTED PLANTS

This safety issue affects all plants, including PWRs and BWRs. Plants operating and under construction are affected.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this issue are estimated in this section and summarized in Tables 1 and 2. Attachment 1 describes the basis for the risk and dose changes that are estimated in the Tables.

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

Smoke Control and Manual Firefighting Effectiveness (148)

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

All PWRs and BWRs are assumed to be affected.

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

3. Plants Selected for Analysis:

Representative PWR: Oconee-3. The analysis is performed for Oconee-3. It was assumed that the quantitative core-melt and public risk reduction estimates are also applicable to BWRs.

4. Parameters Affected by SIR:

The parameter affected by SIR was determined to be τ_s , the mean fire suppression time. See Attachment 1 for a complete description of the parameters and formulae used in this analysis.

5. Base-Case Values for Affected Parameters:

τ_s = 14 minutes. This results in a base-case frequency for large cable shaft fires of $6.0E-5$ /RY.

6. Affected Accident Sequences and Base-Case Frequencies:

FIRE [1] = $1.2E-6$ /RY
FIRE [2] = $5.3E-6$ /RY
FIRE [3] = $3.6E-6$ /RY

7. Affected Release Categories and Base-Case Frequencies:

PWR-1A = $4.2\text{E-}10/\text{RY}$
PWR-1B = $4.4\text{E-}8/\text{RY}$
PWR-2 = $4.2\text{E-}9/\text{RY}$
PWR-3 = $1.1\text{E-}9/\text{RY}$
PWR-4 = $9.9\text{E-}7/\text{RY}$
PWR-5 = $9.1\text{E-}6/\text{RY}$

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

$\bar{F}$ = $1.0\text{E-}5/\text{RY}$

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

W = $2.3\text{E-}1$ person-rem/RY

See Attachment 1 for the worksheets used for these calculations.

10. Adjusted-Case Values for Affected Parameters:

τ_c = 11 minutes. This results in an adjusted-case frequency for large cable shaft fires of $5.2\text{E-}5/\text{RY}$.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

FIRE [1] = $1.04\text{E-}6/\text{RY}$
FIRE [2] = $4.60\text{E-}6/\text{RY}$
FIRE [3] = $3.12\text{E-}6/\text{RY}$

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1A = $3.6\text{E-}10/\text{RY}$
PWR-1B = $3.8\text{E-}8/\text{RY}$
PWR-2 = $3.6\text{E-}9/\text{RY}$
PWR-3 = $9.8\text{E-}10/\text{RY}$
PWR-4 = $8.6\text{E-}7/\text{RY}$
PWR-5 = $7.9\text{E-}6/\text{RY}$

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

$\bar{F}^*$ = $8.6\text{E-}6/\text{RY}$

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

W^* = $2.0\text{E-}1$ person-rem/RY

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

$\Delta\bar{F}$ = $1.2\text{E-}6/\text{RY}$

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

$$\Delta W = 3.2\text{E-}2 \text{ person-rem/RY}$$

17. Total Public Risk Reduction, (ΔW)_{Total}:

Best Estimate (person-rem)	Error Bounds (person-rem)	
	<u>Upper</u>	<u>Lower</u>
1.1E+2	2.6E+4	0

ATTACHMENT 1

The Oconee-3 fire risk analysis developed by Sugnet, et al., (1983) forms the basis for this analysis. As indicated by Sugnet, et al., three accident sequences dominate the fire risk at Oconee-3. These sequences and the event frequencies are:

FIRE [1]	Large cable shaft fire:	6.0E-5/RY
	Safety-relief valve LOCA	1.0E-1
	Failure of operators to initiate HPI	<u>2.0E-1</u>
		1.2E-6/RY
FIRE [2]	Large cable shaft fire:	6.0E-5/RY
	RCP Seal LOCA	3.3E-1
	Failure of operators to initiate HPI	3.0E-1
	Non-occurrence of LOCA sequence	<u>9.0E-1</u>
		5.3E-6/RY
FIRE [3]	Large cable shaft fire:	6.0E-5/RY
	Failure of all Feedwater	1.0
	Failure of operators to initiate HPI	
	or remote shutdown from standby	
	shutdown facility	1.0E-1
	Non-occurrence of other sequences	<u>6.0E-1</u>
		3.6E-6/RY

Resolution of this safety issue affects the frequency of occurrence of a large cable shaft fire. Four parameters were multiplied together by Sugnet, et al., (1983) to obtain the annual frequency of this fire. These four parameters are:

τ_{aux}	= frequency of auxiliary building fire	= 2.3E-2/RY
f_{ER}	= fraction of fires that occur in equipment room	= 1.6E-1
$f_{s,CR}$	= fraction of fires that are large and in cable shaft	= 2.6E-2
$Q_{cs,rg}$	= probability that fire grows and is not suppressed	= 6.2E-1

Since this issue affects the response to a fire situation, particularly the time required to suppress a fire, the parameter affected by this issue is $Q_{cs,rg}$. This parameter was calculated by Sugnet et al. using the following formula:

$$Q(t) = \exp(-t/\tau_s)$$

where $Q(t)$ = conditional probability that a fire is not extinguished by time t

τ_s = mean suppression time

The time required for fire growth, τ_g , was derived by Sugnet, et al., (1983) and the mean value was given as 6.7 min. The conditional frequency of fire growth, given a large fire under the cable trays, is:

$$Q_{cs,rg} = \exp(-\tau_g/\tau_s)$$

Solving this equation for τ , using the known values for the other two parameters results in a value of 14 minutes.

Safety issue resolution is expected to reduce the mean fire suppression time, τ , as a result of improved visibility for firefighters. The issue is not anticipated to have a significant effect on equipment reliability in smoke-filled rooms. This is because the corrosive effects of smoke are long-term degradation mechanisms and are not anticipated to result in near-term control problems (McCracken 1991). Consequently, the main effects of SIR are related to the effectiveness of the fire brigade. The effects of fires on equipment operability (e.g., spurious closure of emergency feedwater valves) were included by Sugnet, et al., (1983).

The adjusted-case value for τ , was taken from NUREG/CR-5088 (Lambright, et al., 1989). The postulated SIR was assumed to reduce the time to detect, initiate suppression, and bring the fire under substantial control to 11 minutes. This value is the upper range of the minimum time estimates provided by Lambright, et al. This represents over a 20% reduction in the mean suppression time. Substituting the adjusted-case value for τ , into the expression above results in an adjusted-case value for parameter $Q_{cs,rs}$ of approximately 0.54. The adjusted-case frequency of a large cable shaft fire was then calculated to be $5.7E-5/RY$. This value was substituted into the accident sequence frequency equations as follows:

FIRE [1]'	Large cable shaft fire:	5.7E-5/RY
	Safety-relief valve LOCA	1.0E-1
	Failure of operators to initiate HPI	<u>2.0E-1</u>
		1.04E-6/RY
FIRE [2]'	Large cable shaft fire:	5.7E-5/RY
	RCP Seal LOCA	3.3E-1
	Failure of operators to initiate HPI	3.0E-1
	Non-occurrence of LOCA sequence	<u>9.0E-1</u>
		4.6E-6/RY
FIRE [3]'	Large cable shaft fire:	5.7E-5/RY
	Failure of all Feedwater	1.0
	Failure of operators to initiate HPI or remote shutdown from SSF	1.0E-1
	Non-occurrence of other sequences	<u>6.0E-1</u>
		3.12E-6/RY

The next step in the analysis was to calculate the base-case and adjusted-case public risk estimates. These parameters were calculated using the containment failure probabilities and public dose consequence estimates developed by Sugnet, et al., (1983). Table 1-A presents the calculations that were performed. A summary of the results is presented below.

	<u>Core Damage Frequency/RY</u>	<u>Public Risk (Person-rem/RY)</u>
Base-Case	1.0E-05	2.3E-01
Adjusted-Case	8.8E-06	2.0E-01
Change	1.3E-06	3.2E-02

Table 1-A Public Risk Calculations

BASE-CASE (see p. 10-283 of Sugnet, et al., 1983)								
CM Bin	CS State	PDB Frequency	Release-category Frequency					
			1A	1B	2	3	4	5
I	C	6.5E-07	1.0E-10	9.1E-09	2.4E-09	---	6.4E-07	
I	E	5.9E-06	1.7E-10	3.0E-08	---	1.3E-11	---	5.9E-06
III	C	3.6E-07	5.6E-11	5.0E-09	1.8E-09	---	3.5E-07	
III	E	3.2E-06	9.2E-11	---	---	1.1E-09	---	3.2E-06
TOTAL		1.0E-05	4.2E-10	4.4E-08	4.2E-09	1.1E-09	9.9E-07	9.1E-06
Consequences, person-rem ^(a)			8.0E+07	2.0E+06	2.0E+07	1.0E+06	1.0E+04	2.0E+03
Risk, person-rem/RY			3.3E-02	8.8E-02	8.4E-02	1.1E-03	9.9E-03	1.8E-02
CALCULATION OF RELEASE CATEGORY PROBABILITIES ^(b)								
CM Bin	CS State	PDB Frequency	Release Category Probability					
			1A	1B	2	3	4	5
I	C	---	1.5E-04	1.4E-02	3.7E-03	0.0E+00	9.8E-01	0.0E+00
I	E	---	2.9E-05	5.1E-03	0.0E+00	2.2E-06	0.0E+00	1.0E+00
III	C	---	1.6E-04	1.4E-02	5.0E-03	0.0E+00	9.7E-01	0.0E+00
III	E	---	2.9E-05	0.0E+00	0.0E+00	3.4E-04	0.0E+00	1.0E+00
ADJUSTED-CASE								
CM Bin	CS State	PDB Frequency	Release Category Probability					
			1A	1B	2	3	4	5
I	C	5.6E-07	8.7E-11	7.9E-09	2.1E-09	---	5.6E-07	0.0E+00
I	E	5.1E-06	1.5E-10	2.6E-08	---	1.1E-11	---	5.1E-06
III	C	3.1E-07	4.9E-11	4.3E-09	1.6E-09	---	3.0E-07	0.0E+00
III	E	2.8E-06	8.1E-11	---	---	9.7E-10	---	2.8E-06
TOTAL		8.8E-06	3.6E-10	3.8E-08	3.6E-09	9.8E-10	8.6E-07	7.9E-06
Consequences, person-rem			8.0E+07	2.0E+06	2.0E+07	1.0E+06	1.0E+04	2.0E+03
Risk, person-rem/RY			2.9E-02	7.6E-02	7.3E-02	9.8E-04	8.6E-03	1.6E-02

- (a) Consequence estimates for each release category were taken from Figure 10-10 of Sugnet, et al., 1983). A 50% exceedence frequency was used.
- (b) Obtained by dividing the release category frequencies by the plant damage bin (PDB) frequencies given in the Table above.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Smoke Control and Manual Firefighting Effectiveness (148)

2. Affected Plants (N):

	<u>N</u>
PWRs	90
BWRs	44
All plants	134

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

	<u>T</u>
PWRs	28.8 yr
BWRs	27.4 yr
All plants	28.3 yr

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

$$\Delta(\bar{F}D_R) = (19860 \text{ person-rem})(1.2E-6/\text{RY}) = 2.4E-2 \text{ person-rem/Ry}$$

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

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
9.1E+1	4.5E+3	0

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

It was estimated that utility labor to implement SIR would amount to approximately 4 man-wks (200 man-hr) to install enhanced detection/suppression equipment and smoke control systems. This labor estimate was increased to account for labor productivity adjustments and equipment removal factors as indicated by Sciacca (1989). The labor productivity adjustment factor was 1.7 and the removal factor was 2.7 (see Abstract 4.1 of Sciacca 1989). This resulted in a total labor requirement of approximately 734 man-hr/plant. This estimate was reduced by 50% (Abstract 4.1 of Sciacca 1989) because it was indicated that 50% of the worker hrs would actually be spent in a radiation field. Therefore, approximately 367 man-hr plant will be spent in a radiation zone.

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

The dose rates given by Andrews, et al., (1983) for areas outside containment are used in these calculations. This dose rate was 2.5 mrem/hr.

$$D = (367 \text{ man-hr})(2.5 \text{ mrem/hr})(1E-3 \text{ rem/mrem}) = 0.9 \text{ person-rem/plant}$$

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

$$ND = (0.9 \text{ person-rem/plant})(134 \text{ plants}) = 1.2E+2 \text{ person-rem}$$

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

Utility labor in radiation zones for SIR operation and maintenance are expected to be minimal relative to the operation and maintenance labor currently spent in the fire protection program. However, a nominal 2 man-hr/RY was included to account for incremental labor in radiation zones for periodic inspections and surveillance of fire protection equipment.

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

$$D_o = (2 \text{ man-hr/RY})(2.5 \text{ mrem/hr})(1E-3 \text{ rem/mrem}) = 5E-3 \text{ person-rem/RY}$$

11. Total Occupational Dose Increase for SIR Operation and Maintenance (NTD_o):

$$\begin{aligned} \overline{NTD}_o &= (5E-3 \text{ person-rem/RY}) \{ (90 \text{ PWRs})(28.8 \text{ yr}) + (44 \text{ BWRs})(27.4 \text{ yr}) \} \\ &= 1.9E+1 \text{ person-rem} \end{aligned}$$

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
1.4E+2	4.2E+2	4.7E+1

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Smoke Control and Manual Firefighting Effectiveness (148)

2. Affected Plants (N):

	<u>N</u>
PWRs	90
BWRs	44
All Plants	134

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

	<u>$\bar{T}$</u>
PWRs	28.8 yr
BWRs	27.4 yr
All plants	28.3 yr

Industry Costs (Steps 4 through 12):

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

$$\Delta(\bar{F}A) = (1.2E-6/R_Y)(\$1.65E+9) = \$1.9E+3/R_Y$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$7.5E+6	\$3.8E+8	0

6. Per-Plant Industry Resources for SIR Implementation:

Industry resources for SIR implementation will be required for two major activities. The first is to search the plant for the potential vulnerabilities identified in this analysis. This search is estimated to require approximately 0.5 man-yr of effort, predominantly in reviewing plant drawings, the existing fire hazards analyses, and a walk-through inspection of potentially susceptible areas.

The second main activity will be install improved smoke removal, fire detection, and suppression systems where indicated. A nominal \$10,000 equipment procurement cost per plant plus an additional 4 man-wks to install the improved equipment was included. This 4 man-wk labor estimate was increased to account for inefficiencies in nuclear power plant labor

productivity resulting from access and handling difficulties, work in radiation zones, congestion and interference (factor of 1.7) and equipment removal factors (factor of 2.7). These labor productivity adjustment factors were taken from Sciacca (1989, Abstract 4.1). The total labor estimate for SIR implementation is 734 man-hr.

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

$$I = (\$50,000/\text{plant}) + (\$10,000/\text{plant}) + (734 \text{ man-hr})(\$56.75/\text{man-hr}) \\ = \$100,000 \text{ (rounded)}$$

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

$$NI = (\$100,000/\text{plant})(134 \text{ plants}) = \$1.3\text{E}+7$$

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

NRC labor for SIR operation and maintenance are expected to be relatively small. There are no major new requirements for periodic inspection or maintenance activities that are not already currently in place. A nominal 1 man-day/RY was added to account for increased operation and maintenance of the improved smoke control, fire detection, and fire suppression systems that are proposed to be installed to replace existing vulnerabilities.

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

$$I_o = (8 \text{ man-hr/RY})(\$56.75/\text{man-hr}) = \$454/\text{RY}$$

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

$$N\bar{I}_o = (\$454/\text{RY}) \{ (90 \text{ PWRs})(28.8 \text{ yr}) + (44 \text{ BWRs})(27.4 \text{ yr}) \} \\ = \$1.7\text{E}+6$$

12. Total Industry Cost (S_i):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.5E+7	\$2.2E+7	\$8.4E+6

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC development costs will be incurred for development of fire protection program guidance in the area of smoke control and for preparation and issuance of a Generic Letter that will transmit the new guidance to all licensees. A nominal 2 man-yrs is estimated for development of the fire protection guidance. The NRC labor needed to prepare and issue a Generic Letter were estimated by Sciacca (1989) to be approximately 4 man-wks.

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

$$\begin{aligned} C_D &= (2 \text{ man-yr})(\$100,000/\text{man-yr}) + (4 \text{ man-wks})(\$2270/\text{man-wk}) \\ &= \$210,000 \text{ (rounded)} \end{aligned}$$

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

Implementation resources are anticipated to include NRC staff labor necessary to review and approve implementation of the enhanced smoke control program and improved equipment and to prepare a safety evaluation. According to Sciacca, NRC labor for these activities are estimated to be 5 man-wk/plant and 7 man-wk/plant, respectively. Therefore, the total NRC labor requirement for SIR implementation is estimated to be 12 man-wk/plant.

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

$$C = (12 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$27,240/\text{plant}$$

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

$$NC = (134 \text{ plants})(\$27,240/\text{plant}) = \$3.6E+6$$

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

NRC labor requirements for SIR operation and maintenance are anticipated to be minimal. A nominal 1 man-day/Ry was included in this analysis for account for periodic reviews of the improved smoke control and manual firefighting programs included in the SIR.

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

$$\begin{aligned} C_o &= [(1 \text{ man-day/Ry})/(5 \text{ man-day/man-wk})] (\$2270/\text{man-wk}) \\ &= \$454/\text{Ry} \end{aligned}$$

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

$$\begin{aligned} \overline{N}C_o &= \$454/\text{Ry} \{ (90 \text{ PWRs})(28.8 \text{ yr}) + (44 \text{ BWRs})(27.4 \text{ yr}) \} \\ &= \$1.7E+6 \end{aligned}$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$5.5E+6	\$7.6E+6	\$3.6E+6

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 149, Adequacy of Fire Barriers

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: A generic safety issue has been identified concerning the fire barriers in nuclear power plants. Two main concerns are addressed by this issue. The first concern is related to the reliability of fire barriers, including barrier penetration seals, doors, and dampers. The second is related to potential deficiencies in the methods used to rate fire barriers. The proposed Safety Issue Resolution (SIR) to deal with potential deficiencies includes requiring licensees to search for potential vulnerabilities, inspect fire barriers for missing or degraded penetration seals, assess the reliability of fire barriers based on site-specific experience, and perform a review and evaluation of the general reliability of U.S.-qualified barriers under actual fire exposure conditions.

AFFECTED PLANTS: This issue affects all operating and planned plants.

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 2.0E+4

OCCUPATIONAL DOSES:

SIR Implementation = 5.4E+1

SIR Operation/Maintenance = 0

Total of Above = 5.4E+1

Accident Avoidance = 1.2E+2

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation = 1.7E+1

SIR Operation/Maintenance = 0

Total of Above = 1.7E+1

Accident Avoidance = 1.0E+1

NRC COSTS:

SIR Development = 9.0E-1

SIR Implementation = 1.4E+0

SIR Operation/Maintenance Review = 0

Total of Above = 2.4E+0

ADEQUACY OF FIRE BARRIERS

ISSUE 149

1.0 SAFETY ISSUE DESCRIPTION

This generic safety issue is concerned with the adequacy of fire barriers in nuclear power plants. NRC regulations regarding fire protection are found in 10 CFR 50, Appendix R. Specific fire protection guidance is found in the *Standard Review Plan*. The objectives of these requirements are to prevent fires from starting, detect and suppress rapidly those fires that do occur, and to protect important plant safety-related features from both the effects of fires and the effects of fire suppression activities. To accomplish these objectives, nuclear power plants have established fire protection programs designed to prevent or limit the effects of fires through appropriate administrative controls, design features, procedures, and training.

The specific areas of concern to this issue are:

- the reliability and failure modes of fire barriers, including barrier penetration seals, doors, and dampers.
- potential deficiencies in the methods used to rate fire barriers.

These two items are discussed in detail in the following subsections.

Fire Barrier Reliability

This item addresses the safety significance of premature failures of fire barriers. Fire barriers include rated fire doors, walls, barrier penetration seals, and dampers which have the objective of containing fires within applicable enclosures and prevent the spread of a fire from one enclosure to another. If fires, hot gases, and smoke can successfully travel from one fire area to another, a single fire could threaten redundant trains of safety-related equipment.

Fire tests on fire barrier components are conducted according to various fire testing standards, including ASTM-E119, NFPA-251, and NFPA-252. A gas-fired furnace is used for the tests. The specimen to be tested is installed into a wall or floor slab which has a known fire-resistive rating. This composite assembly becomes one boundary of the furnace. Generally, two specimens are tested. The first is exposed to the full duration of the test and the second is exposed to 1/2 of the test duration and then subjected to the hose stream test.

The test specimen is installed into the furnace and exposed to direct flame impingement. The test exposes the specimen to temperatures which are controlled over time. In general, the temperatures are controlled as follows:

- 1000°F at 5 minutes
- 1300°F at 10 minutes
- 1550°F at 30 minutes

- 1700°F at one hour
- 1850°F at 2 hours

The structural integrity of the specimen is then tested by the application of a specified standard fire hose water stream. The passage of flame during the fire exposure test and/or water during the hose stream test is considered a failure. In addition, if the temperature on the unexposed side of the test specimen exceeds 325 F for the penetration seals and component enclosures or 650°F for doors and dampers, the test is considered a failure (McCracken 1991).

Potential Deficiencies in Fire Barrier Testing Criteria

Qualification testing of fire barriers in the U. S. does not require a differential pressure across the fire barrier during the furnace exposure tests. It is possible that under actual fire conditions in a nuclear power plant that a positive pressure would develop on the fire side of the barrier from the generation of heat and hot gases and the expansion of the air in the enclosure. Experimental evidence cited in NUREG/CR-5088 (Lambright, et al., 1989) indicates that, should a positive pressure develop on the fire side, the fire barrier may fail prematurely. Such failures may threaten the operability of redundant safety-related components.

International fire barrier tests (e.g., ISO-3008 for fire doors) require a differential pressure of 12 Pa to be maintained on the upper two-thirds of the fire exposure side of the test specimen. Incorporation of this or a similar requirement in U.S. standards would help to assure that a fire barrier would not fail prematurely during an actual fire.

PROPOSED RESOLUTION

The proposed safety issue resolution (SIR) includes the following key items:

- searches for plant-specific vulnerabilities to be performed by licensees
- inspections of fire barriers for missing or degraded penetration seals
- an assessment of the reliability of fire barriers based on site-specific experience
- perform a review and evaluation of the general reliability of U.S.-qualified barriers under actual fire exposure conditions.

AFFECTED PLANTS

This safety issue affects all plants, including PWRs and BWRs. Plants operating and under construction are affected.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this issue are estimated in this section and summarized in Tables 1 and 2. Attachment 1 describes the basis for the risk and dose changes that are estimated in the Tables.

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

Adequacy of Fire Barriers (149)

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

All 134 LWRs are assumed to be affected. Their average remaining life is 28.3 yr.

3. Plants Selected for Analysis:

As discussed in Attachment 1, a generic LWR is assumed.

4-7. Steps Related to Affected Parameters, Accident Sequences, and Release Categories and Their Base-Case Values:

As discussed in Attachment 1, the base-case, affected core-melt frequency is estimated directly for a generic LWR.

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

$\bar{F} = 1.80\text{E-}6/\text{RY}$ (See Attachment 1)

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

$W = 5.94 \text{ person-rem/RY}^1$

10-12. Steps Related to Affected Parameters, Accident Sequences, and Release Categories and Their Base-Case Values:

Based on the discussion by Lambright, et al., (1989), it appears to be reasonable to assume a factor of 10 (i.e., 90%) reduction in the failure probability of a fire barrier is attainable.

¹ An average LWR dose factor of $3.3\text{E}+6$ person-rem is assumed, based on Appendices A to D of Andrews et al. (1983). See, for example, Issue II.B.6 in Supplement 1 of Andrews et al. (1983) for the estimation procedure.

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

$$\bar{F}^* = 1.8\text{E-}7/\text{RY} \text{ (based on 90\% reduction in fire barrier failure probability).}$$

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

$$W^* = 0.59 \text{ person-rem/RY}^2$$

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

$$\Delta\bar{F} = 1.62\text{E-}6/\text{RY}$$

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

$$\Delta W = 5.35 \text{ person-rem/RY}$$

17. Total Public Risk Reduction, (ΔW)_{Total}:

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
2.0E+4	6.8E+5	0

²

An average LWR dose factor of 3.3E+6 person-rem is assumed, based on Appendices A to D of Andrews et al. (1983). See, for example, Issue II.B.6 in Supplement 1 of Andrews et al. (1983) for the estimation procedure.

ATTACHMENT 1

In NUREG/CR-5088, Lambright, et al., (1989) identified six fire risk issues which they felt had not been adequately addressed in the then available PRAs. The following three were estimated to have the highest potential impact on core-melt frequency, each on the order of a factor of 10 relative to the then current PRA estimates as requantified by Lambright et al. (see discussion below):

- Control Systems Interactions
- Manual Fire Fighting Effectiveness
- Adequacy of Fire Analysis Tools.

The following two were estimated to have smaller potential impacts:

- Adequacy of Fire Barriers
- Seismic/Fire Interactions.

The potential impact of the last issue, Total Environment Equipment Survival, remained unknown.

Lambright, et al., requantified the core-melt frequencies due to fire from four PRAs, including the effect of Appendix R modifications, but excluding the six fire risk issues which were identified as inadequately addressed. These reassessed frequencies ranged from $8.8\text{E-}6/\text{RY}$ to $4.6\text{E-}5/\text{RY}$, with a geometric mean of $2.6\text{E-}5/\text{RY}$.

To estimate the core-melt frequency from fire scenarios involving failure of three-hour-rated fire barriers separating adjacent areas containing equipment critical to safe shutdown, Lambright, et al., assumed the following:

1. Approximately 20 pairs of critical adjacent areas exist per plant where three-hour-rated fire barriers are required.
2. For a given area, the typical fire occurrence frequency is $0.0050/\text{RY}$.
3. Only 1/10 of a given area is of interest for fire interaction between the pair.
4. The probability of critical damage occurring on both sides of the barrier if penetrated by fire is 0.1.
5. The probability of failure of a fire barrier is 0.1.

With these assumptions, Lambright, et al., calculated a core-melt frequency of $(0.0050/\text{RY}/\text{critical pair})(0.1)^3 = 5.0\text{E-}6/\text{RY}/\text{critical pair}$, or $1.0\text{E-}4/\text{RY}$ covering all 20 critical pairs.

At first, this estimate may seem inconsistently high relative to the reassessed core-melt frequencies from fire for the four PRAs mentioned above (mean frequency = $2.6\text{E-}5/\text{RY}$, or roughly 1/4 that estimated just from fire barrier inadequacy). However, in light of the Lambright, et al., conclusion that the previously mentioned six fire risk issues were not adequately addressed in any of these PRAs, the inconsistency disappears. If we accept the Lambright, et al., assumptions for the fire barrier scenario and their estimate that the most significant fire risk issues may impact the reassessed core-melt frequencies by a factor of 10, we arrive at a first approximation of $2.6\text{E-}4/\text{RY}$ for the core-melt

frequency due to fire after including the effects of the inadequately addressed fire risk issues.

However, Lambright, et al., estimated that Adequacy of Fire Barriers would have a small impact on the core-melt frequency due to fire when compared to the three most significant fire risk issues. Assuming that a "small impact" is approximately equivalent to no more than a 10% contributor to the core-melt frequency, we estimate either of the following:

- If we take the Lambright, et al., estimate for the core-melt frequency due to fire barrier failure of $1.0\text{E-}4/\text{RY}$ as accurate, we conclude that the overall core-melt frequency due to fire is at least $0.0010/\text{RY}$ (i.e., at least 10 times higher).
- If we take the estimate, based on the Lambright, et al., findings, for the overall core-melt frequency due to fire of $2.6\text{E-}4/\text{RY}$ as accurate, we conclude that the core-melt frequency due to fire barrier failure is no more than $2.6\text{E-}5/\text{RY}$ (i.e., at least 10 times smaller).

This is not a large variation (only a factor of four) in the PRA realm, especially in light of the Lambright, et al., additional calculation of a core-melt frequency due to fire barrier failure of $1.0\text{E-}5/\text{RY}$ under the assumption that the probability of fire barrier failure may be as low as 0.01 (i.e., as much as 10 times smaller than assumed above). It would appear reasonable to infer that Lambright et al. estimate a core-melt frequency due to fire barrier failure in the range from $1\text{E-}5/\text{RY}$ to $1\text{E-}4/\text{RY}$, with a geometric mean of $3\text{E-}5/\text{RY}$.

We can compare this estimate with the following rather crude but reasonable estimate for the core-melt frequency due to fire barrier failure based on the operating history of commercial nuclear power. Based on PNL's most recent draft of the *Regulatory Analysis Technical Evaluation Handbook*, there have been roughly 1,700 RY of commercial nuclear power operation. Over that entire period, the only commercial nuclear power plant fire which has approached inducing a core-melt was the Browns Ferry fire of 22 March 1975. Thus, a crude but reasonable point estimate for the frequency of a fire with at least the potential core-melt impact of Browns Ferry is

$$1/1,700 \text{ RY} = 6.0\text{E-}4/\text{RY}.$$

However, the Browns Ferry fire did not result in a core melt. Kazarians and Apostolakis (1977) estimated the maximum probability for core damage given the Browns Ferry fire to be 0.03, occurring 2.5 hours from the start of the fire and assuming all relief valves failed after 30 minutes and remained so for at least 12 hours. If we apply this conservative estimate to the point estimate for a Browns-Ferry-type fire, we calculate a conservative point estimate for the frequency of core damage from fire of

$$(0.03)(6.0\text{E-}4/\text{RY}) = 1.8\text{E-}5/\text{RY}.$$

Since the Browns Ferry fire represented one in which a fire barrier "failed" (the barrier had previously been breached by non-fire-related activities), this estimate would be a conservative one for core-melt frequency due to fire barrier failure. We note that it lies within the range ($1\text{E-}5/\text{RY}$ to

1E-4/RY) estimated by Lambright, et al., comparing well with the geometric mean (3E-5/RY) over that range.

Commercial nuclear power plants have made numerous improvements since the Browns Ferry fire to reduce the likelihood of a fire-induced core melt. We feel it would be reasonable to lower the Kazarians and Apostolakis conditional probability for core damage, which they considered a maximum anyway, from 0.03 to 0.003 (i.e., a factor of 10 reduction). This results in a point estimate for the core-melt frequency due to fire barrier failure of

$$(1.8\text{E-}5/\text{RY})/10 = 1.8\text{E-}6/\text{RY}$$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Adequacy of Fire Barriers (149)

2. Affected Plants (N):

All 134 LWRs are affected by this issue.

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

The average remaining lifetime is 28.3 yr/plant.

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

$$\Delta(\bar{F}D_R) = (19860 \text{ person-rem})(1.62E-6/\text{RY}) = 0.0322 \text{ person-rem/Ry}$$

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

Best Estimate (person-rem)	Error Bounds (person-rem)	
	Upper	Lower
1.2E+2	8.1E+2	0

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

As discussed in Table 3, it is estimated that utility labor to implement SIR would include approximately one man-month in radiation zones to inspect fire barrier penetration seals.

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

The dose rates given by Andrews, et al., (1983) for areas outside containment are used in these calculations. This dose rate was 2.5 mrem/hr.

$$D = (160 \text{ man-hr})(2.5 \text{ mrem/hr})(1E-3 \text{ rem/mrem}) = 0.4 \text{ person-rem/plant}$$

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

$$ND = (0.4 \text{ person-rem/plant})(134 \text{ plants}) = 5.4E+1 \text{ person-rem}$$

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

Utility labor in radiation zones for SIR operation and maintenance are expected to be minimal. For conservatism, no additional operation/maintenance labor in radiation zones was included in this analysis.

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

$$D_o = 0$$

11. Total Occupational Dose Increase for SIR Operation and Maintenance (NTD_o):

$$NTD_o = 0$$

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
5.4E+1	1.6E+2	1.8E+1

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Adequacy of Fire Barriers (149).

2. Affected Plants (N):

All 134 LWRs are affected by this issue.

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

The average remaining lifetime is 28.3 yr/plant.

Industry Costs (Steps 4 through 12):

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

$$\Delta(\bar{F}A) = (1.62E-6/RY)(\$1.65E+9) = \$2,670/RY$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.0E+7	\$6.8E+7	0

6. Per-Plant Industry Resources for SIR Implementation:

Industry resources for SIR implementation will be required for three main activities. The first is to search the plant for the potential vulnerabilities identified in this analysis. This search is estimated to require approximately 6 man-months of effort, predominantly in reviewing plant drawings, the existing fire hazards analyses, and a walk-through inspection of potentially susceptible areas.

The second main activity will be to inspect fire barriers for missing or degraded penetration seals. This is estimated to require 3 man-months of utility, of which, approximately 1 man-month will be spent in radiation zones.

The third main activity is to assess the reliability of fire barriers based in site-specific experience. This work will involve reviewing the existing fire hazards analyses, estimating the conditions likely to be experienced in an actual fire (principally considering the differential pressures across fire barriers, room ventilation, etc.), and performing

sensitivity studies using varying fire barrier effectiveness parameters. This study is expected to require approximately 6 man-months of effort.

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

$$\begin{aligned} I &= (15 \text{ man-months}) * (\$100,000/\text{man-yr}) / 12 \text{ man-months/man-yr} \\ &= \$125,000 \text{ (rounded)} \end{aligned}$$

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

$$NI = (\$125,000/\text{plant})(134 \text{ plants}) = \$1.7\text{E}+7$$

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

No significant incremental labor requirements for SIR operation and maintenance were identified. Per-plant industry labor for SIR operation and maintenance are assumed to be zero for conservatism.

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

$$I_o = 0$$

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

$$NTI_o = 0$$

12. Total Industry Cost (S_t):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.5E+7	\$2.2E+7	\$8.4E+6

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC development costs will be incurred for a study to evaluate the general reliability of U.S.-qualified fire barriers. The main tasks in this study will be to review plant designs and fire protection requirements to determine the general types of fire barriers in place in nuclear power plants and then categorize them. It is assumed that ten general categories of fire barriers will be established. This task will also develop plans for the qualification tests to be performed in the second main task. A total of 2 man-yr of NRC contractor labor are estimated for this task. The second main task will be to perform qualification tests on each general category of fire barriers. Tests will be performed on a total of 20 specimens (10 specimens from each general category and tests with and without a differential pressure across the fire barrier). The cost of each test is estimated to be \$30,000/specimen. The final task will be to evaluate the reliability of each general type fire barrier and compare their performance in tests with and without differential pressure. An

additional 1 man-yr of NRC contractor resources are estimated for this task.

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

$$\begin{aligned} C_D &= (3 \text{ man-yr})(\$100,000/\text{man-yr}) \\ &\quad + [(20 \text{ specimens})(2 \text{ tests})(\$30,000/\text{test})] \\ &= \$900,000 \end{aligned}$$

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

Implementation resources are anticipated to include NRC staff labor necessary to issue, review, and evaluate a Generic Letter or Information Notice on fire barriers. According to Sciacca (1989), NRC resources for issuance, review, and evaluation of a Generic Letter, (represented by an uncomplicated Technical Specification change) are approximately \$11,000/plant.

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

$$C = \$11,000/\text{plant}$$

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

$$NC = (134 \text{ plants})(\$11,000/\text{plant}) = \$1.4\text{E}+6$$

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

NRC labor requirements for SIR operation and maintenance are anticipated to be minimal. No significant NRC operation and maintenance support activities were identified. Therefore, these costs are assumed to be zero for conservatism.

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

$$C_o = 0$$

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

$$\overline{N}C_o = 0$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$2.4E+6	\$3.2E+6	\$1.5E+6

REFERENCES (For Issue 149)

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Lambright, J. A., et al. 1989. *Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fore Risk Including Previously Unaddressed Issues*. NUREG/CR-5088. Sandia National Laboratory, Albuquerque, New Mexico.

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Sciaccia, F. 1989. *Generic Cost Estimates*. NUREG/CR-4627, Rev. 2. Science and Engineering Associates, Inc., Albuquerque, New Mexico.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 150, Overpressurization of Containment Penetrations

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: Nuclear power plants are provided with a containment isolation function that rapidly closes valves inside and outside the reactor pressure boundary for all fluid penetrations that do not serve an accident mitigation system. A potential problem may exist when water becomes trapped between the inner and outer isolation valves. If the ambient temperature inside containment rises significantly, the water between the isolation valves will expand and the pressure may increase to point at which the penetrations will fail. The potential resolution to this issue is to first evaluate the stresses that may be produced within the penetration and then implement hardware fixes on potentially susceptible penetrations that will either prevent the water from becoming trapped or provide a means to relieve the pressure between the inner and outer isolation valves.

AFFECTED PLANTS: All plants (BWRs - 44; PWRs - 90)

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 5.6E+1

OCCUPATIONAL DOSES:

SIR Implementation =	6.4E+1
SIR Operation/Maintenance =	2.7E+3
Total of Above =	2.7E+3
Accident Avoidance =	7.0E-1

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	6.5
SIR Operation/Maintenance =	6.0
Total of Above =	13
Accident Avoidance =	0.056

NRC COSTS:

SIR Development =	0.05
SIR Implementation =	0.6
SIR Operation/Maintenance Review =	1.6
Total of Above =	2.3

OVERPRESSURIZATION OF CONTAINMENT PENETRATIONS

ISSUE 150

1.0 SAFETY ISSUE DESCRIPTION

Containment isolation is a function provided at all nuclear power plants to ensure containment of radioactive materials in the event of an accident or inadvertent release. This function is integrated into the design of all systems that penetrate the reactor pressure boundary. The containment isolation function is provided for all fluid penetrations that do not serve an accident mitigation system or engineered safeguards function. Double barriers are provided to ensure that no single failure of an active component can result in a loss of this function. The typical means of providing this double barrier system is to place valves inside and outside containment. When required, the valves are closed to prevent leakage of radioactive materials to the environment.

This issue is concerned with potential overpressurization of the containment piping penetrations that could arise from an accident that results in a significant increase in the containment temperature. Overpressurization could result from heating and expansion of water that may become trapped between the inner and outer containment isolation valves. As noted by Minners (1989), heating water at constant volume from 100°F and 100 psia to 200°F would increase the pressure to 3000 psia. This increase in pressure could fail the penetration or the isolation valves and provide a direct flow path for the potentially-contaminated containment atmosphere to the environment. This phenomenon is mitigated somewhat by the fact that the penetration itself will also expand due to the temperature increase as well as the possibility that the valves will not be leaktight.

PROPOSED RESOLUTION

The proposed resolution to this issue is to provide a mechanism for preventing water from becoming trapped or for relieving the pressure that could build up in the piping systems between the inner and outer containment isolation valves. It would be necessary for utilities to perform thermal and structural analyses of the penetration systems to determine which, if any, penetrations are susceptible to such a failure and then to install a pressure relief system if the possible pressure increase would result in failure of the penetration. The pressure relief system could consist of check valves inside the reactor building (which would prevent water from becoming trapped between the two isolation valves) or a means of providing pressure relief (such as a rupture disk or safety relief valve). In the latter case, it is believed to be necessary to provide a storage tank to contain blowdown (liquid or vapor) that would be forced through the pressure relief equipment should it operate.

AFFECTED PLANTS

Resolution of this issue is assumed to affect all commercial nuclear power plants, including those operating and under construction.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this issue are estimated in this section and summarized in Tables 1 and 2. Attachment 1 describes the basis for the risk and dose changes that are estimated in the Tables.

TABLE 1 Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Overpressurization of Containment Penetrations (150)

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

All PWRs and BWRs are assumed to be affected.

	<u>N</u>	<u>$\bar{T}$ (yr)</u>
PWRs	90	28.8
BWRs	44	27.4
All	134	28.3

3. Plants Selected for Analysis:

Representative PWR: Oconee 3 - Representative BWR: Grand Gulf 1

4. Parameters Affected by SIR:

The parameters affected by safety issue resolution are the probabilities of containment isolation failure, as discussed in Attachment 1.

5. Base-Case Values for Affected Parameters:

Oconee: Probability of containment leakage = 0.0073
Grand Gulf: Probability of containment leakage = 0.04

6. Affected Accident Sequences and Base-Case Frequencies:

A list of the affected accident sequences and their base-case frequencies is presented in Attachment 1.

7. Affected Release Categories and Base-Case Frequencies:

As shown in Attachment 1, the affected release categories and their base-case frequencies are shown below. Excess significant figures are provided in order to calculate the changes in core-melt frequency and public risks associated with safety issue resolution.

<u>Release Category</u>	<u>Frequency, RY⁻¹</u>
-------------------------	-----------------------------------

Oconee 3:

PWR-4	9.56E-8
PWR-5	4.61E-7

Total:	5.56E-7
--------	---------

Grand Gulf:

BWR-4	2.818E-6
-------	----------

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

$$\bar{F}_{PWR} = 5.56E-7/RY$$

$$\bar{F}_{BWR} = 2.818E-6/RY$$

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

$$W_{PWR} = 0.719 \text{ person-rem/RY}$$

$$W_{BWR} = 1.719 \text{ person-rem/RY}$$

10. Adjusted-Case Values for Affected Parameters:

The adjusted-case values for the affected parameters incorporate the possibility of overpressurization of containment penetrations as a potential failure mode for containment, as discussed in Attachment 1. The adjusted-case values were estimated as follows:

Oconee: Probability of containment leakage = 0.00744

Grand Gulf: Probability of containment leakage = 0.04014

11. Affected Accident Sequences and Adjusted-Case Frequencies:

The adjusted-case frequencies of the affected accident sequences are presented in Attachment 1.

12. Affected Release Categories and Adjusted-Case Frequencies:

<u>Release Category</u>	<u>Frequency, RY⁻¹</u>
-------------------------	-----------------------------------

Oconee 3:

PWR-4	9.75E-8
PWR-5	4.69E-7

Total:	5.669E-7
--------	----------

Grand Gulf:

BWR-4	2.823E-6
-------	----------

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

$$\bar{F}^*_{PWR} = 5.669E-7/RY$$

$$\bar{F}^*_{BWR} = 2.823E-6/RY$$

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

$$W^*_{PWR} = 1.506 \text{ person-rem/RY}$$

$$W^*_{BWR} = 1.742 \text{ person-rem/RY}$$

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

$$\Delta\bar{F}_{PWR} = 1.067E-8$$

$$\Delta\bar{F}_{BWR} = 5.19E-9$$

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

$$\Delta W_{PWR} = 1.38E-2$$

$$\Delta W_{BWR} = 3.17E-3$$

17. Total Public Risk Reduction, (ΔW)_{Total}:

Best Estimate (man-rem)	Error Bounds (man-rem)	
	<u>Upper</u>	<u>Lower</u>
5.6E+1	1.2E+5	0

ATTACHMENT 1 (To Table 1)

The containment isolation function for fluid systems that penetrate containment typically consists of valves located inside and outside the reactor pressure boundary. These valves close upon receiving a containment isolation signal to prevent potential releases of radioactive material to the environment.

Resolution of this issue affects a nuclear power plant's response to a core damage event and the ability to contain radioactive materials that may be released from the reactor pressure vessel. Containment isolation systems do not have a direct impact on the core damage frequency but do affect the containment failure probability given that a core damage event occurs. Therefore, the approach to developing estimates of the public risk reduction associated with this issue is to evaluate the effect of proposed issue resolution on the containment failure probabilities used in the Oconee and Grand Gulf PRAs as described by Andrews, et al., 1983.

The Oconee and Grand Gulf PRAs (Kolb, et al., 1981, and Hatch, et al., 1981, respectively) addressed failure of or leakage through containment penetrations as a potential failure mode for containment. These containment failure modes are the affected parameters for this issue. For Oconee, the probability of containment penetration leakage was estimated to be 0.0073 given a core damage sequence (containment failure mode "B"). This is the base-case value of the affected parameter for Oconee. For Grand Gulf, the containment penetration leakage failure mode was not explicitly evaluated because none of the accident sequences involving containment isolation failure were found to be among the dominant sequences from a public risk perspective (Hatch, et al., 1981). Therefore, the base-case value that was used in this analysis for the probability of containment isolation failure was developed as shown below.

The Grand Gulf PRA was based on the *Reactor Safety Study* (RSS; NRC 1979). The containment failure analysis for BWRs that was performed in the RSS included an assessment of the conditional probability of containment isolation failure (containment failure mode "B"). The probability of containment isolation failure is dependent upon the specific core damage sequence that occurs prior to containment failure. As a result, there is a range of containment isolation failure probabilities that depend upon prior success or failure of the various engineered safeguards functions at the plant. The representative probability for containment isolation failure of 0.04 was selected based on information in Appendix V of the RSS. This is not the lowest or highest value for this parameter that was given in the RSS but was selected as representative of the weighted average of the range of possible values based on the number of observations in the RSS that are lower and higher than this value.

The release categories associated with containment isolation failure for Oconee are PWR-4 and PWR-5. For Grand Gulf, the affected release categories were not explicitly stated in Andrews, et al., (1983), or Hatch, et al., (1981). It was noted in the RSS and in a recent PRA of Oconee (Sugnet, et al., 1984), that relatively high containment leakage, as could be experienced from failure of containment isolation, would prevent failure of the containment building from potential overpressure caused by hydrogen gas explosions. Consideration of the containment isolation failure mode was therefore applied to accident sequences

involving containment overpressure events caused by gas generation as shown in Appendix B of Andrews, et al., (1983). This involved developing a new base-case risk value for all BWR accident sequences that involve containment overpressure events. To accomplish this, the core damage sequences presented in Andrews, et al., (1983) that could result in containment overpressure were modified to incorporate the base-case containment isolation failure probability rather than the containment overpressure probability. This had the effect of creating a set of new accident sequences that included containment isolation failure events.

The adjusted-case values of the affected parameters were estimated by adding to the base-case values the probability of failure of the penetration system that would arise from overpressurization. A conservative approach was taken to develop a new containment leakage sequence of events that incorporates the potential for overpressurization. This sequence consists of the following events: (1) successful containment isolation; (2) water becomes trapped between inner and outer isolation valves; (3) containment heating causes the water between isolation valves to heat and expand; and (4) the expansion of water between isolation valves causes the penetration to fail. Based on this sequence of events, the probability of containment penetration rupture is the product of the following four terms:

1. p (penetration is susceptible to overpressurization) = $p[1]$
2. p (water becomes trapped between isolation valves) = $p[2]$
3. p (the inboard and outboard isolation valves are leak-tight) = $p[3]$
4. p (inboard and outboard isolation valves rupture simultaneously) = $p[4]$

The probability that a containment penetration is susceptible to this containment failure sequence ($p[1]$) depends on the types of penetrations and isolation valves that are at each plant. Only fluid penetrations are susceptible to this type of failure. In addition, according to Minners (1989), penetrations provided with check valves are not susceptible. To determine the value for $p[1]$, the description of containment penetrations given by Sugnet, et al., (1984) is assumed to be representative of both PWRs and BWRs. A total of 62 penetrations were listed, of which 36 were provided with check valves or were not fluid-carrying lines, and are not susceptible to this containment failure sequence. Therefore, the value of $p[1]$ is equal to approximately $26/62$ or 0.419.

No information is available to calculate the probability that water becomes trapped between the inboard and outboard isolation valves, $p[2]$. Containment isolation valve closures are timed such that one valve closes slightly sooner than the other in order to prevent water from becoming trapped between the valves. Therefore, this event could be caused by failure of the containment isolation system control logic or circuitry to function as intended or by failure of the valve to close when intended. However, due to a lack of specific information, $p[2]$ was assigned a conservative value of 0.5.

No credit was taken for the protection from overpressure that would be provided when one of the isolation valves is not leaktight. Therefore, $p[3]$ was set equal to 1.0 in this analysis. However, according to Pelto, Ames, and Gallucci (1985), there is approximately a 30% chance that one of the valves is not leaktight. This additional 30% reduction in vulnerability will not be factored into the analysis of the containment overpressurization probability.

The probability of failure for rupture of containment penetrations from overpressure, given that the penetration is leaktight and is full of water, was assumed to be represented by the failure rate for valves (also in the rupture failure mode). The value used in this analysis was $1\text{E-}7/\text{hr}$.

The failure probability for the containment isolation valves, $p[4]$, was approximated using the " $\lambda-\tau$ " approach where λ is the random failure rate for valve ruptures and τ is the mean dead time or mean time that an undetected fault could exist. For the penetrations, a fault could potentially exist for up to the test interval for most penetrations. These tests are performed during reactor shutdowns (Sugnet, et al., 1984), or approximately every 18 months. Thus the mean dead time is about 9 months, assuming a constant failure rate over the 18 month test interval. The resulting failure probability for isolation valve ruptures, $p[4]$, is therefore $6.5\text{E-}4$.

The probability of containment isolation failure is the product of $p[1]$, $p[2]$, $p[3]$, and $p[4]$, or approximately $1.4\text{E-}4$.

The base-case and adjusted-case accident sequences and their frequencies are shown on the following pages. Also shown are the base-case and adjusted-case public risk values. Note that excess significant figures are shown in order to calculate the changes in core damage frequency and public risks associated with safety issue resolution.

This approach is believed to be conservative for the following reasons. The containment failure sequence generated above also assumes that water will become trapped in at least one of the penetrations on each containment isolation signal. The probability of this occurrence is believed to be smaller than 1.0 because of the timing of valve closures programmed into the containment isolation function. Finally, penetration failure due to overpressure was considered to occur if either the inboard or outboard isolation valves were ruptured. This does not include the reduced failure probability that would arise from the requirement that both inboard and outboard isolation valves fail simultaneously. In reality, if this phenomenon were to occur, one of the valves is likely to fail before the other valve. This would relieve the pressure between the two valves and tend to prevent overpressurization failure of the second valve.

BWR ANALYSIS

Accident Sequence	Core Damage Frequency	Base-Case Overpressure Probability	Base-Case Sequence Frequency	Adjusted-Case Overpressure Probability	Adjusted-Case Sequence Frequency
BWR Release Category 4		p = 0.04	p = 0.0414		
T1PQI	1.60E-06	4.00E-02	6.40E-08	4.0140E-02	6.42E-08
T23PQI	3.70E-06	4.00E-02	1.48E-07	4.0140E-02	1.49E-07
T1PQE (a)	2.30E-07	5.40E-01	1.24E-07	5.4014E-01	1.24E-07
T23PQE (a)	5.40E-07	5.40E-01	2.92E-07	5.4014E-01	2.92E-07
SI	4.60E-06	4.00E-02	1.84E-07	4.0140E-02	1.85E-07
T1QW	6.20E-06	4.00E-02	2.48E-07	4.0140E-02	2.49E-07
T23QW	1.20E-05	4.00E-02	4.80E-07	4.0140E-02	4.82E-07
T23C	5.40E-06	4.00E-02	2.16E-07	4.0140E-02	2.17E-07
T1QUV (a)	1.90E-06	5.40E-01	1.03E-06	5.4014E-01	1.03E-06
Non-Dominant	9.10E-07	4.00E-02	3.64E-08	4.0140E-02	3.65E-08
TOTAL:			2.82E-06	2.82E-06	

Base-case affected core melt freq = 2.8182E-06 /reactor-yr
 Adj-case affected core melt freq = 2.8234E-06 /reactor-yr
 Change in core melt freq = 5.1912E-09 /reactor-yr
 Base-case affected public risk = 1.7191E+00 person-rem/reactor-yr
 Adj.-case affected public risk = 1.7223E+00 person-rem/reactor-yr
 Change in public risk = 3.1666E-03 person-rem/reactor-yr

(a) The potential for containment overpressure due to gas generation leading to a release category BWR-4 was included in the base-case Grand Gulf PRA results. The containment failure probability in the PRA was estimated to be 0.5. For this analysis, the base-case probability of containment leakage leading to a release category BWR-4 release for these sequences was assumed to be the $(0.5 + 0.04) = 0.54$. The adjusted-case containment failure probability was $(0.5 + 0.04 + 9.4E-4) = 0.541$.

PWR ANALYSIS

Accident Sequence	Core Damage Frequency	Base-Case Overpressure Probability	Base-Case Sequence Frequency	Adjusted-Case Overpressure Probability	Adjusted-Case Sequence Frequency
PWR Release Category 4		p = 0.0073		p = 0.00744	
T2MQFH	5.00E-06	7.30E-03	3.65E-08	7.4400E-03	3.72E-08
S3FH	4.20E-06	7.30E-03	3.07E-08	7.4400E-03	3.12E-08
S2FH	1.30E-06	7.30E-03	9.49E-09	7.4400E-03	9.67E-09
Non-dominant	2.60E-06	7.30E-03	1.90E-08	7.4400E-03	1.93E-08
TOTAL:			9.56E-08	9.75E-08	
PWR Release Category 5					
T2MLU	1.20E-06	7.30E-03	8.76E-09	7.4400E-03	8.93E-09
T1MLU	2.00E-06	7.30E-03	1.46E-08	7.4400E-03	1.49E-08
T1(B3)MLU	2.20E-06	7.30E-03	1.61E-08	7.4400E-03	1.64E-08
T2MQH	1.10E-05	7.30E-03	8.03E-08	7.4400E-03	8.18E-08
S3H	1.00E-05	7.30E-03	7.30E-08	7.4400E-03	7.44E-08
S1D	6.70E-06	7.30E-03	4.89E-08	7.4400E-03	4.98E-08
T2MLUO	8.10E-06	7.30E-03	5.91E-08	7.4400E-03	6.03E-08
T2KMU	7.80E-06	7.30E-03	5.69E-08	7.4400E-03	5.80E-08
S2D	2.00E-06	7.30E-03	1.46E-08	7.4400E-03	1.49E-08
S3D	1.40E-06	7.30E-03	1.02E-08	7.4400E-03	1.04E-08
T1MLUO	5.40E-06	7.30E-03	3.94E-08	7.4400E-03	4.02E-08
T3MLUO	1.10E-06	7.30E-03	8.03E-09	7.4400E-03	8.18E-09
T2MQD	1.50E-06	7.30E-03	1.10E-08	7.4400E-03	1.12E-08
Non-dominant	2.70E-06	7.30E-03	1.97E-08	7.4400E-03	2.01E-08
TOTAL:			4.61E-07	4.69E-07	

Base-case affected core-melt freq = 5.5626E-07/reactor-yr
 Adj.-case affected core-melt freq = 5.6693E-07/reactor-yr
 Change in core melt freq = 1.0668E-08/reactor-yr
 Base-case affected public risk = 7.1883E-01 person-rem/reactor-yr
 Adj.-case affected public risk = 7.3262E-01 person-rem/reactor-yr
 Change in public risk = 1.3786E-02 person-rem/reactor-yr

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Overpressurization of Containment Penetrations (150)

2. Affected Plants (N):

All PWRs and BWRs are assumed to be affected.

	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	47	43	90
BWRs	24	20	44

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

	<u>$\bar{T}$</u>		
	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	27.7	30	28.8
BWRs	25.2	30	27.4

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

$$\text{PWR} = (19,860 \text{ person-rem})(1.07\text{E-}8/\text{RY}) = 2.1\text{E-}4 \text{ person-rem/RY}$$

$$\text{BWR} = (19,860 \text{ person-rem})(5.19\text{E-}9/\text{RY}) = 1.0\text{E-}4 \text{ person-rem/RY}$$

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

<u>Best Estimate</u>	<u>Error Bounds (person-rem)</u>	
<u>(person-rem)</u>	<u>Upper</u>	<u>Lower</u>
0.7	580	0

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

The basis for the amount of labor in radiation zones to implement the SIR is discussed in Section 3.0. For backfit plants, labor in radiation zones was estimated at 5 man-hrs/penetration. SIR was estimated to be implemented on 7 penetrations/plant. The total labor requirements in radiation zones are therefore 35 man-hrs/plant. For forward-fit plants, no labor is performed in radiation zones.

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

The dose rate is assumed to be 25 mrem/hr, which is representative of the dose rate inside containment during reactor shutdowns (Andrews, et al., 1983).

$$\begin{aligned} D &= (35 \text{ person-hr/plant}) (25 \text{ mrem/hr}) = 875 \text{ person-mrem/plant} \\ &= 0.9 \text{ person-rem/plant} \end{aligned}$$

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

For backfit plants only.

$$ND = (47 \text{ PWRs} + 24 \text{ BWRs})(0.9 \text{ person-rem/plant}) = 64 \text{ person-rem}$$

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

Routine testing and inspection of the penetration pressure relief systems is assumed to occur once per 30 days, similar to testing of the containment isolation valves. Testing is assumed to be performed by a 2-person team and each test lasts about 10 minutes. Total labor requirements for the 7 penetration pressure relief systems are:

$$\begin{aligned} \text{Annual labor} &= (12 \text{ tests/yr})(2 \text{ persons})(10 \text{ min/test})(7 \text{ penetrations}) \\ &= 28 \text{ person-hr/Ry} \end{aligned}$$

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

$$D_o = (28 \text{ person-hr/Ry})(25 \text{ mrem/hr}) = 0.7 \text{ person-rem/Ry}$$

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

Operation/maintenance doses are incurred at all plants.

$$\begin{aligned} \bar{N}TD_o &= [(90 \text{ PWRs})(28.8 \text{ yr}) + (44 \text{ BWRs})(27.4 \text{ yr})] (0.7 \text{ person-rem/Ry}) \\ &= 2.7E+3 \text{ person-rem} \end{aligned}$$

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
2.7E+3	8.2E+3	9.0E+2

3.0 SAFETY ISSUE COSTS

This section presents the bases and assumptions that were used to derive estimates of utility labor in radiation zones, utility costs, and NRC costs associated with safety issue resolution (SIR).

There are four major elements of the proposed SIR. These are: (1) utilities perform analyses to determine if certain penetrations are vulnerable to overpressurization following containment heatup; (2) utilities complete safety analyses and quality-related activities associated with implementing SIR on vulnerable penetrations; (3) install pressure relief system; and (4) test/maintain pressure relief system. The cost and labor requirements associated with each element were estimated as follows.

- (1) Perform Overpressure Investigation: This element includes NRC labor to develop acceptable methods, data, and acceptance criteria for utilities to use when evaluating the vulnerability of penetrations to the overpressure phenomenon analyzed in the issue. This is estimated to require approximately 6 man-months of NRC labor. It is further estimated that approximately 1 man-month/plant is required to complete an initial assessment of penetration vulnerability.
- (2) Complete Safety Analyses and Quality-Related Activities: SIR involves installation of hardware inside containment to reduce the vulnerability to penetration overpressure events. Work of this nature requires a high degree of attention to quality assurance in the design of the penetrations' pressure relief systems. Therefore, it is estimated that approximately 2 man-wks of utility labor per penetration is required to complete the design and safety analysis activities associated with the pressure relief systems. The number of vulnerable penetrations was assumed to be 20% of the total penetrations without check valves that were listed in Sugnet, et al., (1984). A total of 75 valves on 37 penetrations were not provided with check valves. Twenty percent of 37 valves is approximately 7 penetrations. Consequently, a total of 7 pressure-relief systems/plant must be designed and installed by the utilities. At 2 man-wks/penetrations, a total of 14 man-wks of utility labor will be required to design and analyze the pressure relief systems.
- (3) Install Pressure Relief Systems: Installation of new hardware within containment at operating plants (i.e., backfit) requires labor in radiation zones that would not be experienced at plants that are under construction (i.e., forward-fit).

The material costs for the hardware fixes consist of: (1) a pressure relief valve or rupture disk to be installed between the inner and outer containment isolation valves; (2) a liquid storage tank to receive the blowdown from the pressure relief system; (3) valve and tank instrumentation; and (4) interconnecting piping. These costs are estimated as follows:

(1)	Pressure relief equipment	\$ 400
(2)	Tank (including fittings)	400
(3)	Instrumentation	100
(4)	Connecting piping	Included above
	TOTAL:	<u>\$ 900</u>

Installation labor was estimated at 5 man-hrs/system, including labor for pipefitters, welders, radiation monitoring staff, and instrument technicians. Therefore, the total labor requirements for this element at backfit plants amounts to 0.9 man-wk/plant. For forward-fit plants, the hardware costs are approximately the same as for backfits; however, labor would be more efficient because the containment is not yet a radiation zone and space is likely to be less of a problem. Therefore, a 50% reduction in labor requirements was estimated for forward-fit plants (i.e., 0.5 man-wk/plant).

- (4) Test/Maintain Pressure Relief System: Periodic testing of the pressure relief systems are assumed to be performed at the same interval as the containment penetration inspections or on a 31 day interval. Assuming test activities are performed by a two-person team and it required 10 continuous minutes to perform each test, a total of 28 man-hr/Ry are required for these activities.

It should be noted that the cost estimates developed here were made under the assumption that the containment isolation systems were located away from the containment building wall. This is not the case for many of the isolation valves, which are located adjacent to the containment walls. This assumption tends to minimize the costs of SIR because the costs would clearly be higher in the case in which the containment structure must be modified to accommodate the pressure relief system. As a result, the cost estimates provided in the analysis are believed to be conservatively low.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Overpressurization of Containment Penetrations (150)

2. Affected Plants (N):

All PWRs and BWRs are assumed to be affected.

	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	47	43	90
BWRs	24	20	44

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

	<u>$\bar{T}$</u>		
	<u>Backfit</u>	<u>Forward-fit</u>	<u>All plants</u>
PWRs	27.7	30	28.8
BWRs	25.2	30	27.4

Industry Costs (Steps 4 through 12):

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

$$\begin{aligned} \text{PWR} &= (\$1.65\text{E}+9)(1.07\text{E}-8/\text{RY}) = \$17.6/\text{RY} \\ \text{BWR} &= (\$1.65\text{E}+9)(5.19\text{E}-9/\text{RY}) = \$8.5/\text{RY} \end{aligned}$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$5.6E+4	\$4.8E+7	0

6. Per-Plant Industry Resources for SIR Implementation:

Backfit plants

(1) Evaluate penetration pressures	4 man-wk/plant
(2) Complete design/safety analyses/QA	14 man-wk/plant
(3) Install SIR	0.9 man-wk/plant
(4) Hardware	\$6300/plant

Forward-fit plants

(1) Evaluate penetration pressures	4 man-wk/plant
(2) Complete design/safety analyses/QA	14 man-wk/plant
(3) Install SIR	0.5 man-wk/plant
(4) Hardware	\$6300/plant

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

$$\begin{aligned} \text{Backfit plants: } I &= (18.9 \text{ man-wk/plant})(\$2270/\text{man-wk}) + \$6300 \\ &= \$4.9\text{E}+4/\text{plant} \end{aligned}$$

$$\begin{aligned}\text{Forward-fit plants: } I &= (18.5 \text{ man-wk/plant})(\$2270/\text{man-wk}) + \$6300 \\ &= \$4.8\text{E}+4/\text{plant}\end{aligned}$$

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

$$\begin{aligned}NI &= (47 \text{ backfit PWRs} + 24 \text{ backfit BWRs})(\$4.9\text{E}+4/\text{plant}) \\ &\quad + (43 \text{ forward-fit PWRs} + 20 \text{ forward-fit BWRs})(\$4.8\text{E}+4/\text{plant}) \\ &= \$6.5\text{E}+6\end{aligned}$$

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

As discussed in Section 3.0, operation and maintenance labor were estimated to require approximately 28 person-hr/Ry (0.7 person-wk/Ry).

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

$$\begin{aligned}I_o &= (0.7 \text{ person-wk/Ry})(\$2270/\text{person-wk}) \\ &= \$1589/\text{Ry}\end{aligned}$$

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

$$\begin{aligned}NTI_o &= [(90 \text{ PWRs})(28.8 \text{ yr}) + (44 \text{ BWRs})(27.4 \text{ yr})](\$1589/\text{Ry}) \\ &= \$6.0\text{E}+6\end{aligned}$$

12. Total Industry Cost (S_r):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.3E+7	\$1.7E+7	\$8.6E+6

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

As discussed in Section 3.0, NRC resources for SIR development were estimated at 24 man-wks of NRC labor.

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

$$CD = (24 \text{ man-wks})(\$2270/\text{man-wk}) = \$5.4\text{E}+4$$

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

It is estimated that approximately 2 man-wk of NRC labor are required to review and evaluate the utility's calculations of the stresses within the penetrations and to review the design, safety analyses, and QA documentation for the penetration pressure relief systems.

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

$$C = (2 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$4.5\text{E}+3/\text{plant}$$

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

$$\begin{aligned} \text{NC} &= (90 \text{ PWRs} + 44 \text{ BWRs})(\$4.5\text{E}+3/\text{plant}) \\ &= \$6.0\text{E}+5 \end{aligned}$$

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

NRC labor for this activity is anticipated to minimal. It is estimated that 1 man-hr/Ry will be sufficient for NRC review of each penetration isolation system. For seven such systems, the annual NRC labor requirement is estimated to be 0.18 man-wk/Ry.

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

$$C_o = (0.18 \text{ man-wk/Ry})(\$2270/\text{man-wk}) = \$409/\text{Ry}$$

20. Total NRC Cost for Review of SIR Operation and Maintenance ($\overline{\text{N}}\overline{\text{T}}\text{C}_o$):

$$\begin{aligned} \overline{\text{N}}\overline{\text{T}}\text{C}_o &= [(90 \text{ PWRs})(28.8 \text{ yr}) + (44 \text{ BWRs})(27.4 \text{ yr})](\$409/\text{Ry}) \\ &= \$1.6\text{E}+6 \end{aligned}$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$2.3E+6	\$3.2E+6	\$1.5E+6

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 151, Reliability of ATWS Recirculation Pump Trip

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: The reliability of ATWS RPT's in some BWR's has been abnormally low. Malfunctioning of the RPT could result in the reactor remaining critical during transients which require a reactor shutdown. A resolution has been proposed that provides a trip signal to a redundant recirculation pump breaker to increase the probability that the recirculation pumps will trip during an ATWS.

AFFECTED PLANTS: BWR: Operating = 25 Planned = 0
 PWR: Operating = 0 Planned = 0

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 1.2E+04

OCCUPATIONAL DOSES:

SIR Implementation = 0
SIR Operation/Maintenance = 0
Total of Above = 0
Accident Avoidance = 35

COST RESULTS (\$10⁶)

INDUSTRY COSTS:

SIR Implementation = 11.3
SIR Operation/Maintenance = 1.5
Total of Above = 12.8
Accident Avoidance = 2.9

NRC COSTS:

SIR Development = 0.11
SIR Implementation Support = 0.28
SIR Operation/Maintenance Review = 0
Total of Above = 0.38

RELIABILITY OF ATWS RECIRCULATION PUMP TRIP (RPT)

ISSUE 151

1.0 ISSUE DESCRIPTION

This issue is concerned with the malfunctioning of the recirculation pump trip (RPT) during an anticipated transient without scram (ATWS) event due to failure of the recirculation pump field breakers to trip on demand. The failures have occurred in plants equipped with General Electric (GE) AKF-25 breakers. Failures of the breakers were caused by binding of the trip latch mechanism and misadjustment of the breakers' mechanical linkage.

As a result of the failures, GE issued a Service Information Letter (SIL) dated December 23, 1986. The letter states that failures can be attributed to circuit breaker misadjustment or lubrication problems, and it suggests corrective actions and improved maintenance practices to improve the breaker performance. The Office of Nuclear Reactor Regulation (NRR) has issued Information Notice 87-12 to all BWR licensees to alert them of the potential problems with the GE AKF-25 circuit breakers. There is not sufficient data at this time to determine if the actions recommended by GE will be sufficient to remedy the breaker problems.

ISSUE RESOLUTION

For purposes of this analysis, this proposed safety issue resolution (SIR) is one involving the installation of a redundant ATWS trip signal to interrupt current to the recirculation pumps. Specifically, a new trip coil, initiated by an ATWS signal, would be installed in each recirculation pump motor-generator (MG) set drive motor. Thus, during an ATWS event, both the recirculation pump field breaker and the MG set supply breaker would receive trip signals, increasing the RPT reliability.

AFFECTED PLANTS

This issue affects only plants that use GE AKF-25 breakers for the RPT. The breakers were used only in the BWR 3 (seven total) and BWR 4 plants (nineteen total). The Pilgrim plant (a BWR 3 plant) has already implemented the resolution and, thus, would not be affected by this change. Therefore, there are a total of 25 plants affected by this issue.

2.0 SAFETY ISSUE RISK AND DOSE

Results of the analyses for public risk and occupational dose associated with the Safety Issue Resolution (SIR) are shown in Tables 1 and 2, respectively.

PUBLIC RISK REDUCTION

For the public risk reduction, the affected parameter is the failure to render the reactor subcritical (C). In this case, the value of C is affected by the reliability of the RPT breaker to actuate on demand. Table 1 summarizes the results of the analysis of increasing the reliability by adding a redundant RPT to a different recirculation pump breaker.

OCCUPATIONAL DOSE

The resolution results in a decrease of occupational dose because the probability of core melt is reduced, and implementation of the resolution does not result in any additional occupational radiation exposure. The results of the analysis of occupational exposures is summarized in Table 2.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Reliability of ATWS Recirculation Pump Trip (Issue 151)

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

Only BWR 3 and 4 plants are affected by this issue. Pilgrim is excluded, as it has already implemented the changes.

	$\frac{N}{}$	$\frac{\bar{T}(\text{yr})}{}$
BWR 3's	6	15.4
BWR 4's	19	30
Total	25	26.5

3. Plant Selected for Analysis:

Grand Gulf 1 - Representative BWR (see Andrews, et al., 1983).

4. Parameter Affected by SIR:

This issue affects event "C", the failure to render the reactor subcritical. The failure probability of event "C" is the product of:

1. failure of the reactor protection system and
2. failure of the RPT or failure of the operator to take appropriate action to shutdown the reactor, given RPS failure.

The probability of failure of the reactor protection system is $7.7\text{E-}6$. The second failure is dominated by the operator failure to take appropriate action, and is equal to 0.1 (see Andrews, et al., 1983, Appendix B).

5. Base-Case Values for Affected Parameters:

The base-case value of Event "C" was modified in this analysis to reflect the operating data relevant to the RPT. As described in Attachment 1, the base-case value for "C", given the degraded reliability of the AKF-25 breakers, is $(7.7\text{E-}6) \times (0.15) = 1.16\text{E-}6$.

6. Affected Accidents Sequences and Base-Case Frequencies:

T_{23C} δ (BWR-2) $8.04\text{E-}06/\text{ry}$

7. Affected Release Categories and Base-Case Frequencies:

BWR-2 = $8.04\text{E-}06/\text{ry}$

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

$\bar{F}_{\text{BWR}} = 8.26\text{E-}06/\text{ry}$

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

$$W_{BWR} = 58.6 \text{ man-rem/ry}$$

10. Adjusted-Case Values For Affected Parameters:

SIR involves the addition of a redundant trip, without any increased reliability that might result from changing to a more reliable breaker or from improved maintenance and testing. This results in an adjusted-case probability of breaker-trip failures estimated at $(0.05 * 0.05) = .0025/\text{Demand}$ (assuming the redundant trip subsystems are completely independent). The adjusted-case value of "C" then becomes: $\{(0.1 + 0.0025) * 7.7E-06\}$, and is equal to $7.89E-07$. This is discussed more completely in Attachment 1.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

$$T_{23}C \quad \delta \text{ (BWR-2)} \quad 5.62E-06/\text{ry}$$

12. Affected Release Categories and Adjusted-Case Frequencies:

$$BWR-2 = 5.62E-06/\text{ry}$$

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

$$\bar{F}^* = 5.62E-06/\text{ry}$$

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

$$W^* = 39.87 \text{ man-rem/ry}$$

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

$$\Delta\bar{F}^* = 2.64E-06/\text{ry}$$

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

$$\Delta W^* = 18.75 \text{ man-rem/ry}$$

17. Total Public Risk Reduction, (ΔW_{TOTAL}):

Best Estimate (man-rem)	Error Bounds (man-rem)	
	Upper	Lower
1.24E+04	1.16E+06	0

ATTACHMENT 1 (To Table 1)

A simplified fault tree was constructed to examine the probability of failure to render the reactor subcritical, event "C" in Andrews, et al., (1983, Appendix B). This event occurs if the reactor protection system (RPS) fails AND the recirculation pump fails to trip OR the operator fails to take appropriate action to shutdown the reactor, given the RPS failure. The simplified fault tree for this event is shown in Figure 1. The figure also presents the original WASH-1400 values (designated "a" on the fault tree) as well as the base-case (designated "b") and adjusted-case values (designated "c").

The original WASH-1400 value for the basic event "RPT fails to trip" was modified to reflect a significantly higher RPT breaker failure probability than was used in WASH-1400. The modified value was incorporated into the base-case. The original WASH-1400 breaker failure probability was 0.003 per demand and was insignificant relative to the probability of operator failure to manually initiate the standby liquid control system or to manually initiate control rod insertion, given RPS failure (estimated in WASH-1400 to be 0.1 per demand). The revised RPT breaker failure probability is 0.5/demand, based on data described by Novak (1988). Therefore, the revised base-case value for event "C" was estimated to be $(7.7E-6) * (0.1 + 0.05) = 1.16E-6/\text{demand}$.

The adjusted-case value for event "C" reflects the installation of a redundant ATWS trip signal to the recirculation pump. Therefore, the event "recirculation pump fails to trip" requires failure of both RPT subsystems. The failure probability of this event becomes $(0.05 * 0.05) = 0.0025$, assuming the RPT subsystems are independent. This takes no credit for potential increases in reliability that could result from improved test and maintenance programs or from changing to a more reliable RPT breaker (this was not part of the SIR). That is, the probability of failure of each RPT breaker was not changed from the base-case historical value of 0.05/demand. The adjusted-case value of event "C" then becomes $(7.7E-6) * (0.1 + 0.0025) = 7.78E-7/\text{demand}$.

FIGURE 1: Simplified Fault Tree for Failure to Render Reactor Subcritical

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification of Safety Issue:

Reliability of ATWS Recirculation Pump Trip (Issue 151)

2. Affected Plants (N):

Only BWR 3 and 4 plants are affected by this issue. Pilgrim is excluded, as it has already implemented the changes.

	<u>N</u>
BWR 3's	6
BWR 4's	<u>19</u>
Total	25

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

	<u>$\bar{T}(\text{yr})$</u>
BWR 3's	15.4
BWR 4's	<u>30</u>
Average	26.5

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

$$\Delta\bar{F}D_R = (19,860 \text{ man-rem}) (2.64\text{E-}06/\text{ry}) = 5.24\text{E-}02 \text{ man-rem/ry}$$

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

<u>Best Estimate</u>	<u>Error Bounds (man-rem)</u>	
<u>(man-rem)</u>	<u>Upper</u>	<u>Lower</u>
35	650	0

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

The labor necessary to implement this SIR will all be conducted outside of radiation areas. No exposure will experienced as a result of this SIR.

3.0 SAFETY ISSUE COSTS

Utility modification cost estimates have been obtained from conversations with staff at Pilgrim, where the SIR has already been implemented. Best estimates were used to estimate NRC resource requirements and for industry operation and maintenance costs. The resulting industry and NRC cost estimates are presented in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification of Safety Issue:

Reliability of ATWS Recirculation Pump Trip (Issue 151)

2. Affected Plants (N):

Only BWR 3 and 4 plants are affected by this issue. Pilgrim is excluded, as it has already implemented the SIR.

	<u>N</u>
BWR 3's	6
BWR 4's	<u>19</u>
Total:	25

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

	<u>$\bar{T}$</u>
BWR 3's	15.4
BWR 4's	<u>30</u>
Average	26.5

Industry Costs (Steps 4 through 12)

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

$$\Delta(\bar{F}A)_{\text{BWR}} = (\$1.65\text{E}+09)(2.64\text{E}-06/\text{ry}) = \$4.36\text{E}+03/\text{ry}$$

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

<u>Best Estimate (\$)</u>	<u>Error Bounds (\$)</u>	
	<u>Upper</u>	<u>Lower</u>
2.9E+06	5.4E+7	0

6-7. Steps Related to Per-Plant Industry Cost for SIR Implementation (I):

The costs for SIR implementation will most likely be different for every plant. The amounts below were obtained from conversations with utility staff and are believed representative of implementation costs.

- Engineering = \$3.87E+05
- Labor = \$4.6E+04
- Hardware = \$1.0E+04
- Total I = \$4.53E+05/plant

8. Total Industry Cost for SIR Implementation:

$$NI = (\$4.53E+05/\text{plant})(25 \text{ plants}) = \$1.13E+7$$

9-10. Steps Related to Per-Plant SIR Operation and Maintenance Costs:

It is expected that the installed redundant trip will only be operated for testing purposes, and, thus, operation costs are negligible. In addition, the maintenance and testing procedures for the redundant trip and breaker will be very similar to those of the existing RPT, and they should, therefore, require very little additional development time. The estimated amounts are shown below.

- Testing (0.5 man-weeks @ \$2270/man-week) = \$1135/ry
- Maintenance (0.5 man-weeks @ \$2270/man-week) = \$1135/ry
- Total I_o = \$2270/ry

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

$$NTI_o = (\$2270/\text{ry})(25 \text{ plants})(26.5 \text{ years}) = \$1.5E+06$$

12. Total Industry Cost (S_I):

<u>Best Estimate</u>	<u>Error Bounds</u>	
	<u>Upper</u>	<u>Lower</u>
\$1.28E+07	\$2.42E+07	\$1.40E+06

NRC Costs (Steps 13 through 21)

13-14. NRC Resources for SIR Development:

It is estimated that SIR development will require approximately one man-year of contractor labor to complete an evaluation of the proposed SIR and any potential alternatives (e.g., enhanced test/maintenance or replacement of the degraded RPT breakers with more reliable models). The study would also include a preliminary review of plant designs to develop an indication of the technical feasibility of the proposed modifications. SIR development also includes NRC issuance of a generic letter to affected plants that was estimated to cost about \$11,000 based on information in NUREG/CR-4627 (Sciaccia 1989).

$$C_o = (1 \text{ man-year})(\$1.0E+05/\text{man-year}) + \$11,000 = \$1.11E+05$$

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

It is anticipated that an average of 5 man-weeks of NRC labor for each affected plant will be required to review proposed plant modifications to incorporate a redundant RPT.

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

$$C = (5 \text{ man-weeks/plant})(\$2270/\text{man-week}) = \$1.14\text{E}+04/\text{plant}$$

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

$$NC = (25 \text{ plants})(\$1.14\text{E}+04/\text{plant}) = \$2.84\text{E}+05$$

18-20. Per-Plant and Total NRC Labor and Cost for Review of SIR Operation and Maintenance:

It is estimated that additional NRC effort required to monitor maintenance and testing of the second RPT would be negligible. Therefore, no additional NRC labor or costs will be incurred as a result of this SIR.

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Error Bounds</u>	
	<u>Upper</u>	<u>Lower</u>
\$3.95E+05	\$7.0E+05	\$9.0E+04

REFERENCES (for Issue 151)

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

Memorandum for T. King from K. Kniel, "Request for Prioritization of New Generic Safety Issue 'Reliability of Recirculation Pump Trip (RPT) During an ATWS,'" March 17, 1989.

Novak, T. M., Memorandum for F. J. Miraglia entitled, "Reliability of Non-Safety Related Field Breakers During ATWS Events." August 12, 1988. U. S. Nuclear Regulatory Commission, Washington D.C.

Sciacca, F. 1989. *Generic Cost Estimates*. NUREG/CR-4627, Rev. 2. Prepared by Science and Engineering Associates, Inc. for the U. S. Nuclear Regulatory Commission, Washington D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NUMBER/TITLE: 152, Assure Qualification of Isolation Valves for Extreme Conditions

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: Concerns have been raised that Generic Letter 89-10 requires that, in testing and qualifying safety-related motor-operated isolation valves, a licensee need only meet a design basis which may not require valve closure under the high differential pressure and flow conditions resulting from a downstream break. Resolution of GI-152 is assumed to consist of four steps:

1. NRC amends Generic Letter 89-10.
2. Licensees review their design bases for compliance. Half of all licensees presumably find that they are in compliance.
3. Remaining licensees initiate analyses to assess valve operability. Presumably, 80% demonstrate compliance via these analyses.
4. The remaining 20% (i.e., 10% of all licensees) perform hardware modifications to the isolation valves and additional analyses to bring them into compliance.

AFFECTED PLANTS: All 112 operating LWRs (75 PWRs and 37 BWRs), although only 11 (7 PWRs and 4 BWRs) perform hardware modifications.

RISK/DOSE RESULTS (person-rem):

PUBLIC RISK REDUCTION	=	110
OCCUPATIONAL DOSES:		
SIR Implementation	=	66
SIR Operation/Maintenance	=	0
Total of Above	=	66
Accident-Avoidance	=	0.69

COST RESULTS (\$10⁶):

INDUSTRY COSTS:		
SIR Implementation	=	3.9
SIR Operation/Maintenance	=	0
Total of Above	=	3.9
Accident-Avoidance	=	0.057
NRC COSTS:		
SIR Development	=	0.018
SIR Implementation Support	=	1.6
SIR Operation/Maintenance Review	=	0
Total of Above	=	1.6

ASSURE QUALIFICATION OF ISOLATION VALVES FOR EXTREME CONDITIONS

ISSUE 152

1.0 SAFETY ISSUE DESCRIPTION

Generic Issue 152 (GI-152) has evolved from concerns raised regarding possible inadequacies in the resolutions to Generic Issues II.E.6.1 and 87 (GI-II.E.6.1 and GI-87), specifically Generic Letter 89-10. The NRC is concerned that this Generic Letter requires that, in testing and qualifying safety-related motor-operated isolation valves, a licensee need only meet a design basis which may not require valve closure under the high differential pressure and flow conditions resulting from a downstream break.

This concern was originally raised by the ACRS with regard to GI-87, "HPCI Steam Line Break Without Isolation." Remick (1989) states:

"...This Generic Letter [89-10] requires each licensee to undertake a program to identify safety-related valves that may not perform adequately under design-basis conditions. Unfortunately...the design basis for the HPCI steam line and other valves in some plants may not specify the type of heavy duty that is of concern. This means simply that the heavy duty loads for these valves do not have to be considered in the program required by Generic Letter 89-10, and the deficiencies addressed by GI-87 will not be remedied.

"Unless Generic Letter 89-10 is amended to require updating or revision of the design basis for valves of this type, we do not consider the requirements of Generic Letter 89-10 as a complete resolution of GI-87."

Minners (1989) states:

"The two subject issues [GI-II.E.6.1 and GI-87] require only that the valves be properly tested and qualified to meet their original (or current, if modified) design basis. Therefore, if that design basis does not include the proper requirement, then the Generic Letter would not require that the valve be capable of the needed operation.

"If such valves exist, there would be a risk associated with continued operation of the plant, and there would be a cost associated with backfitting new specifications including the proper requirements, and then demonstrating or upgrading the valves to meet those new requirements."

Taylor (1989), the Executive Director for Operations, addresses the concerns with respect to GI-87:

"The subject generic issue was initiated due to concern that, if an HPCI turbine steam supply line were to break outside containment, its isolation valve(s) might be unable to isolate the resulting high flow. This could expose safety-related equipment (LPCI/RHR pumps, for example) to a harsh environment beyond its design basis. The HPCI steam line break could therefore result in a LOCA without any ECCS."

SAFETY ISSUE RESOLUTION

Based on the preceding information, resolution of GI-152 is assumed to consist of the following steps:

1. The NRC amends Generic Letter 89-10 to require updating or revision of the design basis as discussed by Remick in his 11/20/89 memo to Carr.
2. To comply with this amendment, all licensees review their design bases with regard to operability of safety-related motor-operated isolation valves. Half of all licensees presumably find that they are in compliance and report so to the NRC. No further action is required by these licensees or by the NRC with respect to them.
3. The remaining half of the licensees find that they may not be in compliance and initiate analyses to assess valve operability. Presumably, 80% of these demonstrate compliance via these analyses and report so to the NRC. No further action is required by these licensees or by the NRC with respect to them.
4. The remaining 20% from this latter half of the licensees (i.e., 10% of all licensees), find that they are not in compliance with amended Generic Letter 89-10 and report so to the NRC. These licensees are assumed to perform hardware modifications to the isolation valves and additional analyses to bring them into compliance, reporting so to the NRC. Any further action by the licensees or the NRC with regard to them (e.g., subsequent valve inspection or requalification following valve replacement) are presumed to be part of the standard inspection and reporting procedures already in place. Thus, no further action is foreseen by the licensees or the NRC in response to GI-152.

The estimation of the public risk, occupational dose, and costs associated with resolution of this safety issue follows the procedure described by Andrews, et al., (1983). The numbers of plants and their remaining lives are based on the most recent PNL updates.

2.0 SAFETY ISSUE RISK AND DOSE

From the preceding discussion, it is evident that any possible reduction in the public risk and change in the occupational dose can arise only at the 10% of all plants which perform hardware modifications. In terms of the numbers of affected plants, the above assumptions translate into the following:

1. Of the 112 operating LWRs (75 PWRs and 37 BWRs), all perform Step 2. Half of these (38 PWRs and 19 BWRs) find and report that they are in compliance and take no further action.
2. Of the remaining 37 PWRs and 18 BWRs, all perform Step 3. Eighty percent of these (30 PWRs and 14 BWRs) demonstrate and report that they are in compliance (based on the analyses performed). These plants take no further action.
3. The remaining 7 PWRs and 4 BWRs (~10% of all LWRs) perform Step 4. They subsequently demonstrate and report that they are in compliance (after the appropriate hardware modifications and additional analyses). These plants take no further action. Only at these plants can a reduction in public risk and a change in occupational dose be realized.

A change in public risk is assumed to arise from the elimination of the need for certain components (motor-operated valves and pumps) to operate under the potentially harsh environments resulting from a failure to isolate a steam line break. The quantification of this reduction is presented in Table 1 and Attachment 1. The change in occupational dose is quantified in Table 2.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Assure Qualification of Isolation Valves for Extreme Conditions (GI-152)

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

Only 10% of the 112 LWRs (75 PWRs and 37 BWRs) are assumed to perform hardware modifications capable of reducing the risk. Thus, 7 PWRs and 4 BWRs are assumed affected, with average remaining lives of 25.8 yr and 24.1 yr, respectively.

3. Selected Analysis Plants:

Oconee 3 - representative PWR
Grand Gulf 1 - representative BWR

4. Affected Parameters:

Oconee - D, E, A1, B1, C1, W, X (only in accident sequences initiated by S_3 , see Attachment 1)
Grand Gulf - VGA1, VGA2, VGB1, VGB2, SA, SB, LA2, LB2, SSA, SSB, PA27, PB27 (only in accident sequences initiated by S_1 , (see Attachment 1)

5. Affected Parameters' Base-Case Values:

Oconee (see Attachment 1):

$D = E = 0.0112$
 $A1 = C1 = W = X = 0.0081$
 $B1 = 0.0186$
 $D \cdot E = 1.25E-4$
 $W \cdot X = 6.56E-5$
 $B \cdot W = C \cdot X = 2.67E-5$
 $D \cdot X = E \cdot W = 9.07E-5$

Grand Gulf (see Attachment 1):

$VGA1 = VGA2 = VGB1 = VGB2 = SA = SB = 0.0029$
 $LA2 = LB2 = SSA = SSB = 0.0022$
 $PA27 = PB27 = 8.9E-4$
 $VGA2 \cdot VGB2 = VGA2 \cdot VGB1 = VGA1 \cdot VGB2 = SA \cdot SB = 8.41E-6$
 $VGB2 \cdot SSA = VGA2 \cdot SSB = VGA2 \cdot LB2 =$
 $VGB2 \cdot LA2 = VGB1 \cdot SSA = VGA1 \cdot SSB =$
 $VGA1 \cdot VGB1 = VGA1 \cdot LB2 = VGB1 \cdot LA2 = 6.38E-6$
 $SSA \cdot SSB = LB2 \cdot SSA = LA2 \cdot SSB = LA2 \cdot LB2 = 4.84E-6$

6. Affected Accident Sequences and Base-Case Frequencies:

Oconee:

$S_3H-\gamma$ (PWR-3)	=	1.99E-7/RY
$S_3H-\beta$ (PWR-5)	=	2.91E-9/RY
$S_3H-\epsilon$ (PWR-7)	=	1.99E-7/RY
$S_3FH-\gamma$ (PWR-2)	=	7.75E-8/RY
$S_3FH-\beta$ (PWR-4)	=	1.13E-9/RY
$S_3FH-\epsilon$ (PWR-6)	=	7.75E-8/RY
$S_3D-\gamma$ (PWR-3)	=	3.09E-7/RY
$S_3D-\beta$ (PWR-5)	=	4.51E-9/RY
$S_3D-\epsilon$ (PWR-7)	=	3.09E-7/RY

Grand Gulf:

SI- α (BWR-1)	=	2.46E-9/RY
SI- δ (BWR-2)	=	2.46E-7/RY

7. Affected Release Categories and Base-Case Frequencies:

Oconee:

PWR-2	=	7.75E-8/RY
PWR-3	=	5.08E-7/RY
PWR-4	=	1.13E-9/RY
PWR-5	=	7.42E-9/RY
PWR-6	=	7.75E-8/RY
PWR-7	=	5.08E-7/RY

Grand Gulf:

BWR-1	=	2.46E-9/RY
BWR-2	=	2.46E-7/RY

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

Oconee	=	1.18E-6/RY
Grand Gulf	=	2.48E-7/RY

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

Oconee	=	3.14 person-rem/RY
Grand Gulf	=	1.76 person-rem/RY

10. Affected Parameters' Adjusted-Case Values:

Oconee (see Attachment 1):

D = E	=	0.0102
A1 = C1 = W = X	=	0.0074
B1	=	0.0169
D•E	=	1.04E-4

$$\begin{aligned}
 W \bullet X &= 5.48E-5 \\
 B \bullet W = C \bullet X &= 2.44E-5 \\
 D \bullet X = E \bullet W &= 7.55E-5
 \end{aligned}$$

Grand Gulf (see Attachment 1):

$$\begin{aligned}
 VGA1 = VGA2 = VGB1 = VGB2 = SA = SB &= 0.0026 \\
 LA2 = LB2 = SSA = SSB &= 0.002 \\
 PA27 = PB27 &= 8.1E-4 \\
 VGA2 \bullet VGB2 = VGA2 \bullet VGB1 = VGA1 \bullet VGB2 = SA \bullet SB &= 6.76E-6 \\
 VGB2 \bullet SSA = VGA2 \bullet SSB = VGA2 \bullet LB2 = \\
 \quad VGB2 \bullet LA2 = VGB1 \bullet SSA = VGA1 \bullet SSB = \\
 \quad VGA1 \bullet VGB1 = VGA1 \bullet LB2 = VGB1 \bullet LA2 &= 5.20E-6 \\
 SSA \bullet SSB = LB2 \bullet SSA = LA2 \bullet SSB = LA2 \bullet LB2 &= 4.00E-6
 \end{aligned}$$

11. Adjusted-Case Frequencies of Affected Accident Sequences:

Oconee:

$$\begin{aligned}
 S_3H-\gamma \text{ (PWR-3)} &= 1.66E-7/RY \\
 S_3H-\beta \text{ (PWR-5)} &= 2.42E-9/RY \\
 S_3H-\epsilon \text{ (PWR-7)} &= 1.66E-7/RY \\
 S_3FH-\gamma \text{ (PWR-2)} &= 6.75E-8/RY \\
 S_3FH-\beta \text{ (PWR-4)} &= 9.86E-10/RY \\
 S_3FH-\epsilon \text{ (PWR-6)} &= 6.75E-8/RY \\
 S_3D-\gamma \text{ (PWR-3)} &= 2.66E-7/RY \\
 S_3D-\beta \text{ (PWR-5)} &= 3.88E-9/RY \\
 S_3D-\epsilon \text{ (PWR-7)} &= 2.66E-7/RY
 \end{aligned}$$

Grand Gulf:

$$\begin{aligned}
 SI-\alpha \text{ (BWR-1)} &= 2.07E-9/RY \\
 SI-\delta \text{ (BWR-2)} &= 2.07E-7/RY
 \end{aligned}$$

12. Adjusted-Case Frequencies of Affected Release Categories:

Oconee:

$$\begin{aligned}
 PWR-2 &= 6.75E-8/RY \\
 PWR-3 &= 4.32E-7/RY \\
 PWR-4 &= 9.86E-10/RY \\
 PWR-5 &= 6.30E-9/RY \\
 PWR-6 &= 6.75E-8/RY \\
 PWR-7 &= 4.32E-7/RY
 \end{aligned}$$

Grand Gulf:

$$\begin{aligned}
 BWR-1 &= 2.07E-9/RY \\
 BWR-2 &= 2.07E-7/RY
 \end{aligned}$$

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

Oconee	=	1.01E-6/RY
Grand Gulf	=	2.09E-7/RY

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

Oconee	=	2.68 person-rem/RY
Grand Gulf	=	1.48 person-rem/RY

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

Oconee	=	1.7E-7/RY
Grand Gulf	=	3.9E-8/RY

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

Oconee	=	0.46 person-rem/RY
Grand Gulf	=	0.28 person-rem/RY

17. Total Public Risk Reduction (ΔW_{Total}):

Best Estimate	=	110 person-rem
Lower Bound	=	0
Upper Bound	=	2.2E+4 person-rem

ATTACHMENT 1

OCONEE (PWR)

A steam line break is assumed to correspond to an S_3 LOCA. If this LOCA is not isolated, the potential for introducing a harsh environment into containment exists. This harsh environment may adversely affect the operation of certain components needed to mitigate the LOCA. The affected components are assumed to be motor-operated valves (MOV) and pumps, specifically for failure modes designated as "hardware" or "control circuitry," found in accident sequences initiated by an S_3 LOCA. For Oconee, the affected parameters, along with their original failure probabilities, are as follow:

D:	Normally-closed (NC) MOV		
	Hardware	=	0.001
	Control circuitry	=	<u>0.0064</u>
			<u>0.0074</u>
	Pump		
	Hardware	=	0.001
	Control circuitry	=	<u>0.0018</u>
			<u>0.0028</u>
E:	same as D		
A1:	NC MOV	=	0.0074 (as in D)
B1:	2 NC MOVs	=	0.0074 each (as in D)
	Pump		
	Hardware	=	0.001
	Control circuitry	=	<u>0.0011</u>
			<u>0.0021</u>
C1:	same as A1		
W:	same as A1		
X:	same as A1		

It was assumed that, as originally evaluated, these failure probabilities did not account for an increased failure potential under harsh environmental conditions. Therefore, given an S_3 LOCA, the base-case failure probabilities are assumed to be 10% higher than their original values, i.e.,

D:	NC MOV	=	(1.1)(0.0074)	=	0.0081
	Pump	=	(1.1)(0.0028)	=	<u>0.0031</u>
					<u>0.0112</u>
E:	same as D				
A1:	NC MOV	=			0.0081 (as in D)
B1:	2 NC MOVs	=	(2)(1.1)(0.0074)	=	0.0163
	Pump	=	(1.1)(0.0021)	=	<u>0.0023</u>
					<u>0.0186</u>

C1: same as A1

W: same as A1

X: same as A1

The following pairs need not be corrected for multiple maintenance contributions as in the original evaluation. Their base-case values become:

$$D \bullet E: (0.0112)^2 = 1.25E-4$$

$$W \bullet X: (0.0081)^2 = 6.56E-5$$

$$B \bullet W: 0.0033(0.0081) = 2.67E-5$$

C • X: same as B • W

$$D \bullet X: 0.0112(0.0081) = 9.07E-5$$

E • W: same as D • X

Resolution of the safety issue is assumed to return these failure probabilities to their original values, in effect a 10% reduction in their base-case values. Thus, the adjusted-case values become:

$$D: 0.0074 + 0.0028 = 0.0102$$

E: same as D

A1: 0.0074 (as in D)

$$B1: 2(0.0074) + 0.0021 = 0.0169$$

C1: same as A1

W: same as A1

X: same as A1

$$D \bullet E: (0.0102)^2 = 1.04E-4$$

$$W \bullet X: (0.0074)^2 = 5.48E-5$$

$$B \bullet W: 0.0033(0.0074) = 2.44E-5$$

C • X: same as B • W

$$D \bullet X: 0.0102(0.0074) = 7.55E-5$$

E • W: same as D • X

GRAND GULF (BWR)

A steam line break is assumed to correspond to an S LOCA. If this LOCA is not isolated, the potential for introducing a harsh environment into containment exists. This harsh environment may adversely affect the operation of certain components needed to mitigate the LOCA. The affected components are assumed to be motor-operated valves (MOV) and pumps, specifically for failure modes designated as "hardware" or "control circuitry," found in accident sequences initiated by an S LOCA. For Grand Gulf, the affected parameters, along with their original failure probabilities, are as follow:

VGA1:	2 NC MOVs		
	Hardware	=	0.001
	Control circuitry	=	<u>0.0003</u>
			<u>0.0013</u> each
VGA2:	normally-open (NO) must-close MOV		
	Hardware	=	0.001
	Control circuitry	=	<u>0.0003</u>
			<u>0.0013</u>
	NC MOV	=	0.0013 (as in VGA1)
VGB1:	same as VGA1		
VGB2:	same as VGA2		
SA:	same as VGA1		
SB:	same as VGA1		
LA2:	Pump		
	Hardware	=	0.001
	Control circuitry	=	<u>0.001</u>
			<u>0.002</u>
LB2:	same as LA2		
SSA:	same as LA2		
SSB:	same as LA2		
PA27:	pump fails to run ~30 hrs	=	8.1E-4
PB27:	same as PA27		

It was assumed that, as originally evaluated, these failure probabilities did not account for an increased failure potential under harsh environmental conditions. Therefore, given an S LOCA, the base-case failure probabilities are assumed to be 10% higher than their original values, i.e.,

VGA1:	2 NC MOVs	=	(2)(1.1)(0.0013)	=	0.0029
VGA2:	NO must-close MOV	=	(1.1)(0.0013)	=	0.0014
	NC MOV	=	(1.1)(0.0013)	=	<u>0.0014</u>
					<u>0.0029</u>
VGB1:	same as VGA1				
VGB2:	same as VGA2				
SA:	same as VGA1				
SB:	same as VGA1				
LA2:	pump	=	(1.1)(0.002)	=	0.0022
LB2:	same as LA2				
SSA:	same as LA2				
SSB:	same as LA2				
PA27:	pump fails to run ~30 hrs	=	(1.1)(8.1E-4)	=	8.9E-4
PB27:	same as PA27				

The following pairs need not be corrected for multiple maintenance contributions as in the original evaluation. Their base-case values become:

VGA2•VGB2:	(0.0029) ²	=	8.41E-6
VGB2•SSA:	(0.0029)(.0022)	=	6.38E-6
VGA2•SSB:	same as VGB2•SSA		
VGA2•VGB1:	same as VGA2•VGB2		
VGA1•VGB2:	same as VGA2•VGB2		
VGA2•LB2:	same as VGB2•SSA		
VGB2•LA2:	same as VGB2•SSA		
SSA•SSB:	(0.0022) ²	=	4.84E-6
VGB1•SSA:	same as VGB2•SSA		
VGA1•SSB:	same as VGB2•SSA		
LB2•SSA:	same as SSA•SSB		
LA2•SSB:	same as SSA•SSB		

VGA1•VGB1: same as VGB2•SSA
 VGA1•LB2: same as VGB2•SSA
 VGB1•LA2: same as VGB2•SSA
 SA•SB: same as VGA2•VGB2
 LA2•LB2: same as SSA•SSB

Resolution of the safety issue is assumed to return these failure probabilities to their original values, in effect a 10% reduction in their base-case values. Thus, the adjusted-case values become:

VGA1: $(2)(0.0013) = 0.0026$
 VGA2: $0.0013 + 0.0013 = 0.0026$
 VGB1: same as VGA1
 VGB2: same as VGA2
 SA: same as VGA1
 SB: same as VGA1
 LA2: 0.002
 LB2: same as LA2
 SSA: same as LA2
 SSB: same as LA2
 PA27: $8.1E-4$
 PB27: same as PA27
 VGA2•VGB2: $(0.0026)^2 = 6.76E-6$
 VGB2•SSA: $0.0026(0.002) = 5.20E-6$
 VGA2•SSB: same as VGB2•SSA
 VGA2•VGB1: same as VGA2•VGB2
 VGA1•VGB2: same as VGA2•VGB2
 VGA2•LB2: same as VGB2•SSA
 VGB2•LA2: same as VGB2•SSA
 SSA•SSB: $(0.002)^2 = 4.00E-6$

VGB1•SSA: same as VGB2•SSA
VGA1•SSB: same as VGB2•SSA
LB2•SSA: same as SSA•SSB
LA2•SSB: same as SSA•SSB
VGA1•VGB1: same as VGB2•SSA
VGA1•LB2: same as VGB2•SSA
VGB1•LA2: same as VGB2•SSA
SA•SB: same as VGA2•VGB2
LA2•LB2: same as SSA•SSB

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:
Assure Qualification of Isolation Valves for Extreme Conditions (152)
2. Affected Plants (N):
7 PWRs and 4 BWRs
3. Average Remaining Lives of Affected Plants ($\bar{T}$):
25.8 yr for PWRs and 24.1 yr for BWRs
4. Per-Plant Occupational Dose Reduction Due to Accident Avoidance, $\Delta(FD_R)$:

PWR	-	(19,900 person-rem)(1.7E-7/RY)	=	0.0034 person-rem/RY
BWR	-	(19,900 person-rem)(3.9E-8/RY)	=	7.8E-4 person-rem/RY
5. Total Occupational Dose Reduction Due to Accident Avoidance (ΔU):

Best Estimate	=	0.69 person-rem
Lower Bound	=	0
Upper Bound	=	28 person-rem
6. Per-Plant Utility Labor in Radiation Zones for SIR Implementation:

It is assumed that, at each of the 11 LWRs performing hardware modifications, the equivalent of 6 person-wk of labor will be required in radiation zones.
7. Per-Plant Occupational Dose Increase for SIR Implementation (D):

Since labor inside containment should be required, a radiation field of 0.025 rem/hr is assumed. The occupational dose increase becomes:

$$D = (0.025 \text{ rem/hr})(6 \text{ person-wk/plant})(40 \text{ person-hr/person-wk})$$
$$= 6.0 \text{ person-rem/plant}$$
8. Total Occupational Dose Increase for SIR Implementation (ND):
$$ND = (11 \text{ plants})(6.0 \text{ person-rem/plant}) = 66 \text{ person-rem}$$
- 9-11. Occupational Dose Increase for SIR Operation and Maintenance:

No action subsequent to implementation is foreseen as attributable to this issue. Therefore, no occupational dose will accrue.
12. Total Occupational Dose Increase (G):

Best Estimate	=	66 person-rem
Lower Bound	=	22 person-rem
Upper Bound	=	200 person-rem

3.0 SAFETY ISSUE COSTS

Industry and NRC costs are developed for each of the three LWR groups discussed above. Consistent with the preceding assumptions, no costs are associated with operation or maintenance with respect to this issue. Only implementation and NRC development costs exist. Costs are quantified in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Assure Qualification of Isolation Valves for Extreme Conditions (152)

2. Affected Plants (N):

All 112 operating LWRs (75 PWRs and 37 BWRs) are assumed to review their design bases for compliance with the amendment to Generic Letter 89-10. Half of these (38 PWRs + 19 BWRs = 57 LWRs) find that they are in compliance, report so to the NRC, and take no further action. The remaining 55 LWRs (37 PWRs and 18 BWRs) conduct analyses to determine if they are in compliance. Eighty percent of these (30 PWRs + 14 BWRs = 44 LWRs) demonstrate compliance via these analyses, report so to the NRC, and take no further action. The remaining 11 LWRs (7 PWRs and 4 BWRs) find that they are not in compliance and perform hardware modifications and additional analyses to bring them into compliance. They report so to the NRC and take no further action.

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

For the 7 PWRs and 4 BWRs performing hardware modifications, the average remaining lives are 25.8 yr and 24.1 yr, respectively.

Industry Costs (Steps 4 Through 12)

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

PWR	-	$(\$1.65E+9)(1.7-7/RY)$	=	\$280/RY
BWR	-	$(\$1.65E+9)(3.9-8/RY)$	=	\$64/RY

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

Best Estimate	=	\$5.7E+4
Lower Bound	=	0
Upper Bound	=	\$2.3E+6

6. Per-Plant Industry Resources for SIR Implementation:

Design Basis Review and Report to NRC (All 112 LWRs):

Labor = 6 person-wk/plant (In his 12/27/89 memo to Michelson, Taylor states that review of the design bases would be a "significant burden on licensees that would require considerable staff time and resources.")

Additional Analyses and Report to NRC (55 LWRs):

Labor = 12 person-wk/plant

Hardware Modifications, Additional Analyses, and Report to NRC (11 LWRs):

Labor = 24 person-wk/plant

Equipment (cost estimated directly in next step)

No Additional Downtime (work can be performed during regularly scheduled outages)

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

All 112 LWRs:

Labor = (6 person-wk/plant)(\$2270/person-wk) = \$1.4E+4/plant

55 LWRs:

Labor = (12 person-wk/plant)(\$2270/person-wk) = \$2.7E+4/plant

11 LWRs:

Labor = (24 person-wk/plant)(\$2270/person-wk) = \$5.4E+4/plant

Equipment = \$2.0E+4/plant
\$7.4E+4/plant

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

(112 plants)(\$1.4E+4/plant) = \$1.6E+6

(55 plants)(\$2.7E+4/plant) = \$1.5E+6

(11 plants)(\$7.4E+4/plant) = \$8.1E+5
\$3.9E+6

9-11. Industry Labor and Cost for SIR Operation and Maintenance:

Any industry operation and maintenance activities associated with this issue are assumed to be performed as part of activities already in place, such as standard inspection and reporting procedures. No industry labor or cost is foreseen for operation and maintenance.

12. Total Industry Cost (S_i):

Best Estimate = \$3.9E+6

Lower Bound = \$2.0E+6

Upper Bound = \$5.9E+6

NRC Costs (Steps 13 Through 21)

13. NRC Resources for SIR Development:

The NRC would amend Generic Letter 89-10 and issue the amendment to all licensees. Review and approval by the ACRS would be sought since they originated concern over the issue. The estimate to accomplish this is 8 person-wk of equivalent NRC staff labor.

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

$$\text{Labor} = (8 \text{ person-wk})(\$2270/\text{person-wk}) = \$1.8\text{E}+4$$

15. Per-Plant NRC Labor to Support SIR Implementation:

The NRC must respond to initial reports of compliance/non-compliance with the amendment to Generic Letter 89-10 from all 112 LWRs. This response should soon become standard as the reports arrive, averaging 2 person-wk/plant of NRC staff labor. The NRC must respond to subsequent reports with submitted analyses from 55 of these LWRs. This response is estimated to require 6 person-wk/plant of NRC staff labor. Eleven of these LWRs will perform hardware modifications and additional analyses to establish compliance. The NRC response to these should be somewhat plant-specific, averaging an estimated 12 person-wk/plant of equivalent NRC staff labor (some contractor support may be utilized).

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

All 112 LWRs:

$$\text{Labor} = (2 \text{ person-wk/plant})(\$2270/\text{person-wk}) = \$4500/\text{plant}$$

55 LWRs:

$$\text{Labor} = (6 \text{ person-wk/plant})(\$2270/\text{person-wk}) = \$1.4\text{E}+4/\text{plant}$$

11 LWRs:

$$\text{Labor} = (12 \text{ person-wk/plant})(\$2270/\text{person-wk}) = \$2.7\text{E}+4/\text{plant}$$

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

NC = (112 plants)(\\$4500/plant)	=	\$5.0E+5
(55 plants)(\\$1.4E+4/plant)	=	\$7.7E+5
(11 plants)(\\$2.7E+4/plant)	=	\$3.0E+5
		<u>\$1.6E+6</u>

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

Any industry operation and maintenance activities associated with this issue are assumed to be performed as part of activities already in place, such as standard inspection and reporting procedures. No industry labor or cost is foreseen for operation and maintenance. Thus, no NRC labor or cost is foreseen for review.

21. Total NRC Cost (S_N):

Best Estimate	=	\$1.6+6
Lower Bound	=	\$8.0+5
Upper Bound	=	\$2.4+6

REFERENCES (for Issue 152)

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

Minners, W. 1989. Memorandum to W. Morris dated December 4, 1989. U.S. Nuclear Regulatory Commission, Washington D.C.

Remick, L. 1989. Memorandum to W. Carr dated November 20, 1989. U.S. Nuclear Regulatory Commission, Washington D.C.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 154, Adequacy of Emergency and Essential Lighting

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: This issue addresses potential deficiencies related to the design and maintenance of emergency and essential lighting systems at nuclear power plants. Questions have arisen regarding the adequacy of criteria for ensuring adequate lighting levels to perform necessary tasks, which may include tasks required to mitigate the effects of abnormal or design basis events concurrent with a loss of offsite power. Current guidance relates only to emergency lighting that is required for safe shutdown in the event of a fire (Gillespie 1990). The proposed resolution to this issue involves NRC development of guidance on appropriate areas to be provided with emergency lighting, adequate illumination levels, and preventive maintenance criteria. Nuclear plant licensees may be required to install additional emergency lighting and upgrade existing lighting systems to comply with the new requirements.

AFFECTED PLANTS: This issue affects all PWRs and all BWRs, including operating plants and plants under construction.

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 1.8E+3

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	1.1E+1

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	5.7E+0
SIR Operation/Maintenance =	8.6E+0
Total of Above =	1.4E+1
Accident Avoidance =	9.3E-1

NRC COSTS:

SIR Development =	1.1E-1
SIR Implementation =	1.8E+0
SIR Operation/Maintenance Review =	3.4E+0
Total of Above =	5.4E+0

ADEQUACY OF EMERGENCY AND ESSENTIAL LIGHTING

ISSUE 154

1.0 SAFETY ISSUE DESCRIPTION

This issue addresses potential deficiencies related to the design and maintenance of emergency and essential lighting systems at nuclear power plants. The essential lighting system is the portion of the normal plant lighting system that is backed up by diesel generators. The essential lighting system, which is rated non-Class 1E but receives power from Class 1E sources, provides illumination in the event that the normal lighting system fails. Essential lighting is provided for the control room, shutdown panel area, main walkways and stairways, for Class 1E equipment, Class 1E switchgear rooms, and in areas used for safe shutdown of the plant. The essential lighting system receives power from Class 1E redundant, independent AC buses, each of which is capable of providing 50% of the essential lighting in vital areas.

Plants are also provided with DC emergency lighting which receive power from redundant Class 1E batteries through inverters. This system provides illumination for the control room, auxiliary electrical equipment rooms, at stairways, and at points leading to plant exits. This system is normally deenergized and is automatically energized upon loss of AC power to the normal system. In addition, plants are provided with a standby, self-contained battery-operated lighting system that is capable of providing additional backup lighting for 8 hours if the DC emergency lighting fails. These units are located throughout the plant in areas necessary to provide for evacuation of personnel and in areas required for safe shutdown.

The current regulatory bases for emergency and essential lighting are found in 10 CFR 50, Appendix R, Section III.J, as well as Appendix E. These regulations, which address emergency lighting relative to fire protection, require emergency lighting units with at least an 8-hr battery power supply in all areas used for operation of safe shutdown equipment and in access and egress routes thereto. Emergency lighting guidance is provided in 10 CFR 50 Appendix R, and in the Standard Review Plan, Sections 9.5.1 and 9.5.3.

The issue has arisen as a result of an event at the Palo Verde Nuclear Generating Station (PVNGS) in which inadequate lighting conditions exacerbated an unrelated reactor trip (Gillespie 1990). The reactor trip occurred as a result of a grid-induced load rejection and was followed by reactor/turbine trip, main steam isolation, containment isolation, and safety injection. During this event, half of the steam bypass control system malfunctioned, the atmospheric dump valves (ADVs) failed to operate from both the control room and the remote shutdown panel, and the emergency lighting for the main steam support structure (MSSS) failed.

The failure of the emergency lighting system hampered operators in their attempt to recover from the ADV failures. After remote operation of the ADVs failed, operators attempted to establish local control of the ADVs, which are located in the MSSS. Normal lighting in the MSSS was not available. Backup emergency lighting was found to be inadequately positioned in one MSSS room to

allow the operators to accomplish their tasks. The lighting in the second MSSS room was not functioning. Because of a burned-out light bulb, operators were not able to restore essential lighting in the second MSSS room.

An augmented inspection team (AIT) examined the circumstances surrounding the Palo Verde event and noted the following (Gillespie 1990):

- the AIT noticed discrepancies in the implementation of essential and emergency lighting requirements
- inadequate preventive maintenance on the emergency and essential lighting in the MSSS contributed to the inadequate lighting conditions

In addition, plant personnel observed that, in some instances, the lighting was found to be inadequate to perform the required tasks because the design did not require emergency lights or because the emergency lights provided inadequate illumination. In other instances, the plants did not conform to the design requirements (Gillespie 1990).

PROPOSED RESOLUTION

Gillespie (1990) stated the following potential inadequacies in the current guidance on emergency and essential lighting:

1. Current guidance relates only to emergency lighting that is required for safe shutdown in the event of a fire.
2. Illumination level requirements are vague and subjective. Existing references in the guidance to the criteria in the Illuminating Engineering Society Handbook may not be adequate.
3. Current guidance does not address the need for emergency lighting to execute tasks which may be required to mitigate the effects of abnormal or design basis events concurrent with a loss of offsite power.
4. 10 CFR 50.53 does not address the need for emergency lighting to execute tasks which may be required to mitigate the effects of a station blackout event.

The proposed resolution to this issue involves NRC development of guidance on appropriate areas to be provided with emergency lighting, adequate illumination levels, and preventive maintenance criteria. Nuclear plant licensees may be required to install additional emergency lighting and upgrade existing lighting systems to comply with the new requirements.

AFFECTED PLANTS

This safety issue affects all plants, including PWRs and BWRs. Plants operating and under construction are affected.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this issue are estimated in this section and summarized in Tables 1 and 2. Attachment 1 describes the basis for the risk and dose changes that are estimated in the Tables.

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

Adequacy of Emergency and Essential Lighting (154).

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

All PWRs and BWRs are assumed to be affected.

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

3. Plants Selected for Analysis:

Representative PWR: Oconee-3. The results for Oconee-3 are scaled to approximate the results for a BWR.

4. Parameters Affected by SIR:

The parameters that are affected by SIR are related to the operator recovery actions to accomplish tasks in areas that might not be well illuminated. For this analysis, it was assumed that emergency lighting was adequate in the control room but might not be adequate in other areas of the plant. It was observed in the Oconee-3 PRA that operators have the capability to initiate core cooling and inventory makeup from the Standby Shutdown Facility (SSF) to recover from some core damage sequences. Parameter RESSFSI appears in dominant cut sets of the Oconee-3 PRA and was therefore used as the basis for this analysis.

5. Base-Case Values for Affected Parameters:

The original Oconee-3 PRA value of parameter RESSFSI was modified to establish the base-case value for this analysis. As described in Attachment 1, the base-case value of RESSFSI was calculated to be 0.0393/demand.

6. Affected Accident Sequences and Base-Case Frequencies:

This analysis was performed using the assumption that the ratio of the base-case to adjusted-case values of RESSFSI is equal to the ratio of the base-case to adjusted-case affected core melt frequencies of Oconee-3. The basis for this assumption is described in detail in Attachment 1. The modified base-base case values were derived from the original Oconee-3 PRA

frequencies for the affected accident sequences (see Step 11), as described in Attachment 1. The modified base-case values are shown below:

<u>Sequence</u>	<u>Frequency, RY⁻¹</u>
T ₁₂ BU	1.51E-5
T ₁₀ BU	4.83E-6
T ₆ BU	4.73E-6
T ₄ BU	4.13E-7
T ₅ BU	2.22E-7
TBLX	1.61E-7
TQU	8.26E-8
TOTAL:	<u>2.56E-5</u>

7. Affected Release Categories and Base-Case Frequencies:

All the accident sequences listed in Step 6 were assigned to a release category that is represented by a weighted-average consequence estimate, as discussed in Attachment 1.

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

$$\bar{F}_{PWR} = 2.56E-5/RY \qquad \bar{F}_{BWR} = 1.15E-5/RY$$

The analysis was performed for the Oconee-3 PWR and then scaled to calculate the results for the PWR (See Attachment 1).

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

$$W_{PWR} = .63.9 \text{ person-rem/RY} \qquad W_{BWR} = 76.7 \text{ person-rem/RY}$$

10. Adjusted-Case Values for Affected Parameters:

The adjusted-case value of parameter RESSFSI was calculated to be 0.0390/demand (See Attachment 1).

11. Affected Accident Sequences and Adjusted-Case Frequencies:

The dominant cut sets from the Oconee-3 accident sequences were reviewed to identify cut sets which contained recovery events that occur outside the control room. The sequences that involved such recovery actions and their frequencies (which represent the adjusted-case) are as follows:

<u>Sequence</u>	<u>Frequency, RY⁻¹</u>
T ₁₂ BU	1.5E-5
T ₁₀ BU	4.8E-6
T ₆ BU	4.7E-6
T ₄ BU	4.1E-7
T ₅ BU	2.2E-7
TBLX	1.6E-7
TQU	<u>8.1E-8</u>
TOTAL:	<u>2.54E-5</u>

12. Affected Release Categories and Adjusted-Case Frequencies:

All the accident sequences listed in Step 11 were assigned to a release category that is represented by a weighted-average consequence estimate, as discussed in Attachment 1.

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

$$\bar{F}_{PWR}^* = 2.54E-5/R\bar{Y} \qquad \bar{F}_{BWR}^* = 1.14E-5/R\bar{Y}$$

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

$$W_{PWR}^* = 63.4 \text{ person-rem} \qquad W_{BWR}^* = 76.1 \text{ person-rem/R}\bar{Y}$$

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

$$\Delta\bar{F}_{PWR} = 1.8E-7/R\bar{Y} \qquad \Delta\bar{F}_{BWR} = 8.1E-8/R\bar{Y}$$

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

$$\Delta W_{PWR} = 4.5E-1 \text{ person-rem/R}\bar{Y} \qquad \Delta W_{BWR} = 5.4E-1 \text{ person-rem/R}\bar{Y}$$

17. Total Public Risk Reduction, (ΔW)_{Total}:

Best Estimate (person-rem)	Error Bounds (person-rem)	
	Upper	Lower
1.8E+3	7.7E+6	0

ATTACHMENT 1

Emergency and essential lighting failures and their effects are not explicitly addressed in existing Probabilistic Risk Assessments (PRAs). As a result, a different approach than that described by Andrews, et al., (1983) was used in this analysis to estimate the public risk reduction associated with safety issue resolution (SIR). In general, emergency and essential lighting is important to restoring equipment that may have failed to operate properly during abnormal and design-basis events. These systems are, in general, not contributors to the frequency of occurrence of these events nor are they contributors to mechanical failures or malfunctions of the equipment. Instead, the emergency and essential lighting systems provide an important service to operators that may be called upon to restore or recover a failed or malfunctioning component. In this sense, emergency and essential lighting systems are important accident management features. Therefore, this analysis focuses on the probabilities of operator recovery actions in response to abnormal and design-basis events. Based on engineering judgement and the design/operation of the lighting systems, it is concluded that emergency lighting systems in the control rooms are not as vulnerable to failure as other plant areas. Therefore, operator recovery actions that occur in the control room are not examined further in this analysis.

The Oconee-3 PRA (Sugnet, et al., 1984) was reviewed in detail to identify operator recovery events that could be affected by failure of emergency and essential lighting systems in plant areas other than the control room. Among the dominant cut sets, only recovery events involving operator actions in the Standby Shutdown Facility (SSF) were included as basic events. The SSF is a separate, hardened structure which is capable of providing core cooling if needed in the event of failures of normal and emergency cooling systems. The SSF is provided with the auxiliary service water system (ASWS), which provides a backup supply of water to the steam generators, and the reactor coolant volume control system (RCVCS), which is designed to maintain sufficient inventory in the reactor coolant system to sustain natural circulation. Electrically-operated equipment in the SSF can be powered from Class 1E AC buses which are backed up by a diesel generator in the SSF.

It was stated in Sugnet, et al., (1984) that recovery events are important to the estimation of the total core damage frequency. Application of recovery events were stated to have reduced the core damage frequency from internal events by more than order of magnitude. It was further observed that the dominant cut set in the highest frequency core damage sequence (a sequence initiated by a loss of service water) included an operator recovery action from the SSF. This recovery action was assigned a failure probability of 0.1 so inclusion of this recovery action in this dominant cut set reduced the total sequence frequency by about an order of magnitude. Based on this observation, it is assumed that the affected portion of the total core damage frequency is proportional to the probability of non-recovery. The affected core-damage frequency is the sum of the frequencies of accident sequences which contain cut sets with recovery actions that occur outside the control room. Therefore, the ratio of the non-recovery probabilities before and after SIR is proportional to the ratio of the base-case and adjusted-case affected core damage frequencies. This assumption forms the basis for the public risk reduction analysis.

Sugnet, et al., (1984) indicated that only a preliminary analysis was performed to develop an estimate of the non-recovery probability from a loss of service water by initiating seal injection from the SSF. For this analysis, a more detailed treatment of this recovery event was performed to determine the base-case and adjusted-case non-recovery probabilities. Since emergency lighting failures were not addressed in the Oconee PRA, the base-case needs to be adjusted to include the contribution to the core damage frequency that results from such failures. The adjusted-case represents the plant core damage frequency after SIR.

The adjusted-case affected core damage frequency was calculated by summing the frequencies of the accident sequences which contain cut sets with basic events involving recovery actions outside the control room. This involved reviewing the dominant cut sets that are given by Sugnet, et al., (1984) and identifying the specified recovery actions. The accident sequences which contained these basic events and their frequencies are listed below:

<u>Sequence</u>	<u>Frequency, RY⁻¹</u>
T ₁₂ BU	1.5E-5
T ₁₀ BU	4.8E-6
T ₆ BU	4.7E-6
T ₄ BU	4.1E-7
T ₅ BU	2.2E-7
TBLX	1.6E-7
TQU	<u>8.1E-8</u>
TOTAL:	<u>2.54E-5</u>

The total adjusted-case affected core-damage frequency is therefore estimated to be 2.53E-5/RY.

The base-case and adjusted-case non-recovery probabilities were derived using the method described by Weston, et al., (1987). In this method, the total failure probability for the recovery action is the probability of either failing to diagnose the appropriate action or failing to perform the recovery action successfully. The following expression is used:

$$P(NR) = P(ND) + P(NA) - P(ND)P(NA)$$

where $P(NR)$ = failure probability for the recovery action
 $P(ND)$ = failure probability for diagnosing the required action within time T_D
 $P(NA)$ = failure probability for physically accomplishing the action within time T_A

The time parameters, T_A and T_D were derived based on the recovery time for the action to initiate seal injection from the SSF at Oconee-3 that was described above. This event, which must be accomplished within 30 minutes after the loss of service water, was designated event RESSFSI in the Oconee PRA. The time parameters are calculated using the following expression:

$$T_D = T_M - T_A$$

where T_M is defined as the maximum allowable time to accomplish the recovery action.

For the recovery action, $T_M = 30$ minutes. It was assumed that a maximum of fifteen minutes is required for operators to reach the SSF and then manually initiate seal injection. Therefore:

$$\begin{aligned} T_D &= (30 - 15) \text{ minutes} \\ &= 15 \text{ minutes} \end{aligned}$$

The probability $P(ND)$ was obtained from a lookup table given by Weston, et al., (1987). For time $T_D = 15$ minutes, $P(ND) = 0.039$. Since the proposed SIR does not affect the operator's ability to diagnose the need to activate the SSF, this parameter value is common to both the base- and adjusted cases.

The probability $P(NA)$ was derived by constructing a Human Reliability Analysis (HRA) event tree using the techniques described by Swain and Guttman (1983). In general, after the situation has been successfully diagnosed, the control room operator will instruct another operator to physically travel to the SSF and manually initiate seal injection. The control room operator will be monitoring the seal injection indications in the control room which will provide feedback as to the success of the SSF operator. The HRA event tree developed for this recovery action is shown in Figure 1.

The base-case failure probability for the action phase of this event, $P(NA)$, was derived from Figure 1 as follows:

$$\begin{aligned} P(NA) &= X_1 + X_2 + X_3 + X_4 \\ &= 0.0 + 3E-6 + 3E-6 + 3E-4 \\ &= 3.1E-4 \end{aligned}$$

	HEP	Source
ε = Mechanical, physical failure prevents operator from getting message to SSF	Negligible	NUREG/CR-1278
A = Error in message from control room operator	0.001	NUREG/CR-1278
B = Operator fails to monitor feedback	0.003	NUREG/CR-1278
C = SSF operator fails to understand message	0.001	NUREG/CR-1278
D = Operator fails to monitor feedback	0.003	NUREG/CR-1278
E = SSF operator fails to initiate seal injection (Base-case)	0.1	Sugnet 1984
E' = SSF operator fails to initiate seal injection (Adjusted-case)	0.003	NUREG/CR-1278
F = Operator fails to monitor feedback	0.003	NUREG/CR-1278

Figure 1: HRA Event Tree for Recovery Action

Similarly, the adjusted-case failure probability, $P(NA)'$, was calculated to be:

$$\begin{aligned} P(NA)' &= X_1 + X_2 + X_3 + X_4' \\ &= 0.0 + 3E-6 + 3E-6 + 9E-6 \\ &= 1.5E-5 \end{aligned}$$

As shown, the difference between the base-case and adjusted-case values appears in subevent E on Figure 1, "SSF operators fail to initiate seal injection." The values of E (base-case) and E' (adjusted-case) for this analysis were assigned using the following logic. The base-case human error probability (HEP), E, was assumed to be equivalent to the Oconee PRA value for event RESSFSI or 0.1. This value was taken from the *PRA Procedures Guide* (NRC 1983) as stated by Sugnet, et al., (1984). The adjusted-case HEP was assigned a value of 0.003, based on the HEP for selecting an incorrect valve from a control console that is clearly and unambiguously marked (Swain and Guttman 1983). It is believed that adequate lighting in the SSF would create conditions in which the control markings can be clearly read. Conversely, for the situation in which the emergency lighting is not adequate, a relatively high HEP of 0.1 is warranted.

The base-case and adjusted case failure probabilities, $P(ND)$ and $P(ND)'$, respectively, were calculated as follows:

$$\begin{aligned} P(ND) &= (0.039) + (3.1E-4) - (0.039)(3.1E-4) \\ &= 3.93E-2 \end{aligned}$$

$$\begin{aligned} P(ND)' &= (0.039) + (1.5E-5) - (0.039)(1.5E-5) \\ &= 3.90E-2 \end{aligned}$$

The base-case and adjusted-case non-recovery failure probabilities were used to calculate the base-case and adjusted-case core-melt frequencies. As discussed previously, the ratio of the base-case to adjusted-case failure probabilities was assumed to be approximately equal to the ratio of the base-case to adjusted-case core damage frequencies. Therefore, the following expression was used:

$$\frac{P(ND)}{P(ND)'} = \frac{CMF}{CMF'}$$

$$CMF = [P(ND)/P(ND)'] CMF'$$

where CMF = Base-case core melt frequency
 CMF' = Adjusted-case core melt frequency

For this analysis, the adjusted-case core melt frequency is represented by the original Oconee-3 core melt frequency, $5.4E-5$. Therefore:

$$\begin{aligned} CMF &= [(3.93E-2)/(3.90E-2)] (2.54E-5/RY) \\ &= 2.56E-5/RY \end{aligned}$$

The next step in the analysis is to calculate the base-case and adjusted-case public risk estimates by combining the core melt frequencies with the consequences of the core-melt, should it occur. Recovery events occur in accident sequences which encompass all of the potential release categories for Oconee-3. Consequently, the approach used here to calculate the public risk estimates was

to calculate a weighted-average consequence value and multiply this by the core-melt frequencies calculated above. The weighted-average consequence estimate was obtained by dividing the total Oconee-3 plant risk given by Andrews, et al., (1983), 207 person-rem/RY, by the total core-melt frequency, $8.2\text{E-}5/\text{RY}$. The resulting weighted-average consequence value was calculated to be $2.5\text{E}+6$ person-rem. The base-case and adjusted-case public risk estimates are therefore 63.88 and 63.43 person-rem/RY, respectively.

The base-case and adjusted-case core melt frequencies and public risk estimates for BWRs were calculated by scaling the results for Oconee-3 to a generic BWR using the scaling relationships given by Andrews, et al., (1983). These scaling relationships are:

$$\text{CMF}_{\text{BWR}} = 0.45 \text{ CMF}_{\text{PWR}}$$

$$\text{RISK}_{\text{BWR}} = 1.2 \text{ RISK}_{\text{PWR}}$$

Using these relationships, the base-case and adjusted-case core melt frequencies for a generic BWR were estimated to be $1.15\text{E-}5/\text{RY}$ and $1.14\text{E-}5/\text{RY}$, respectively. The base-case and adjusted-case public risks were estimated to be 76.65 and 76.11 person-rem/RY, respectively. Excess significant figures were maintained in these calculation in order to illustrate the differences between the base-case and adjusted-case.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Adequacy of Emergency and Essential Lighting (154).

2. Affected Plants (N):

	<u>N</u>
PWRs	90
BWRs	44
All plants	134

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

	<u>$\bar{T}$</u>
PWRs	28.8 yr
BWRs	27.4 yr
All plants	28.3 yr

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

$$\Delta(\bar{F}D_R)_{PWR} = (19860 \text{ person-rem})(1.8E-7/RY) = 3.6E-3 \text{ person-rem/RY}$$

$$\Delta(\bar{F}D_R)_{BWR} = (19860 \text{ person-rem})(8.1E-8/RY) = 1.6E-3 \text{ person-rem/RY}$$

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

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
1.1E+1	9.5E+3	0

6-12 Steps Related to Per-Plant Utility Labor in Radiation Zones for SIR Implementation, Operation and Maintenance:

SIR does not involve labor in radiation zones. All of the upgrades and enhancements to the emergency lighting system, including improved preventive maintenance provisions, would occur in areas outside of the containment building in low radiation zones or in areas in which the radiation dose rate is zero. The proposed SIR focuses on lighting required to accomplish manual recovery actions under abnormal and design-basis accident conditions and does not apply to potential improvements inside containment, which is not accessible during such conditions. Therefore, the occupational doses due to SIR implementation, operation, and maintenance are zero.

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Adequacy of Emergency and Essential Lighting (154)

2. Affected Plants (N):

	<u>N</u>
PWRs	90
BWRs	44
All Plants	134

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

	<u>$\bar{T}$</u>
PWRs	28.8 yr
BWRs	27.4 yr
All plants	28.3 yr

Industry Costs (Steps 4 through 12):

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

$$\begin{aligned} \text{PWR} &= (\$1.65\text{E}+9)(1.8\text{E}-7/\text{RY}) = \$297/\text{RY} \\ \text{BWR} &= (\$1.65\text{E}+9)(8.1\text{E}-8/\text{RY}) = \$134/\text{RY} \end{aligned}$$

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

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$9.3E+5	\$7.9E+8	0

6. Per-Plant Industry Resources for SIR Implementation:

SIR implementation includes an initial assessment of each plant's emergency and essential lighting systems, preparation of safety documentation, and procurement and installation of improvements to existing lighting systems, where needed. The total labor requirements for these activities are estimated to be 10 man-wks. An additional equipment cost of \$20,000/plant is estimated for procurement of improved lighting systems. See Attachment 2 for additional details.

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

$$I = (10 \text{ man-wk/plant})(\$2270/\text{man-wk}) + \$20,000 = \$42,700/\text{plant}$$

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

$$NI = (\$42,700/\text{plant})(134 \text{ plants}) = \$5.7\text{E}+6$$

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

Labor requirements to perform the enhanced preventive maintenance and testing program were estimated in Attachment 2 to be 1 man-wk/RY.

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

$$I_o = (1 \text{ man-wk/RY})(\$2270/\text{man-wk}) = \$2270/\text{RY}$$

11. Total Industry Cost for SIR Operation and Maintenance ($\overline{NI}_o$):

$$\overline{NI}_o = (\$2270/\text{RY})(134 \text{ plants})(28.3 \text{ yr}) = \$8.6\text{E}+6$$

12. Total Industry Cost (S_I):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.4E+7	\$2.0E+7	\$9.1E+6

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC resources to develop the proposed resolution is estimated to be about 6 man-months of contractor labor to develop emergency lighting system guidance (including design guidance as well as guidance for an acceptable preventive maintenance program) plus an additional 6 man-months to examine in detail the costs and benefits of the proposed guidance. These estimates were increased by 10% to account for NRC staff labor requirements to monitor the contract(s). Therefore, the total labor requirements are estimated at 13.2 man-months.

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

$$C_D = (13.2 \text{ man-months})(\$8330/\text{man-month}) = \$110,000$$

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

NRC resources are needed to prepare and issue guidance to the industry on acceptable emergency and essential lighting systems, review and approve proposed plant modifications, and prepare safety evaluation reports. As discussed in Attachment 2, the total NRC labor requirements for these activities are estimated to be 6 man-wks/plant for the latter two activities plus 3 man-wks to prepare and issue the proposed guidance. All these estimates include NRC technical staff labor plus management and legal reviews.

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

$$\begin{aligned} C &= (6 \text{ man-wk/plant})(\$2270/\text{man-wk}) + (3 \text{ man-wk})(\$2270/\text{man-wk}) \\ &= \$13,620/\text{plant} + \$6800 \end{aligned}$$

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

$$\begin{aligned} NC &= (\$13,620/\text{plant})(134 \text{ plants}) + \$6800 \\ &= \$1.8\text{E}+6 \end{aligned}$$

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

NRC labor for support of SIR implementation are estimated in Attachment 2 to amount to approximately 2 man-day/RY.

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

$$\begin{aligned} C_o &= [(2 \text{ man-day/RY})/(5 \text{ man-day/man-wk})] (\$2270/\text{man-wk}) \\ &= \$908/\text{RY} \end{aligned}$$

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

$$\overline{N}C_o = (\$908/\text{RY})(134 \text{ plants})(28.3 \text{ yr}) = \$3.4\text{E}+6$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$5.4E+6	\$7.3E+6	\$3.4E+6

ATTACHMENT 2 (To Table 3)

This Attachment outlines the NRC and industry cost estimates and their bases.

NRC Development

The NRC is assumed to undertake a research program to support development of the SIR. This research program is projected to include a survey of plant emergency lighting system designs (including interfaces with other plant systems such as electric power and maintenance programs), development of guidance on adequate illumination levels, emergency lighting system designs, and preventive maintenance programs, coordination with the Illumination Engineering Society in the area of standards development, and a more detailed evaluation of the costs and benefits of the proposed SIR. A total resource requirement of 1 man-yr contractor labor plus 0.1 man-yr of NRC labor was estimated for these activities.

NRC Implementation

Implementation costs would include costs associated with preparation, review, and approval of revised emergency lighting system guidance (3 man-wk), review and approval of utility actions/plans to respond to revised lighting guidance (2.5 man-wk/plant), and preparation of license amendments, safety evaluation reports, etc., to document the changes at the plants (3.5 man-wk/plant). The estimated resource requirements were taken from Sciacca (1987).

NRC Support of SIR Operation and Maintenance

This cost category includes NRC labor to periodically review and evaluate the emergency lighting systems and enhanced preventive maintenance programs. A nominal 2 man-day/RV was estimated for this activity.

Industry Implementation Costs

Implementation of the proposed SIR involves a number of activities at the plants. First, it was assumed that each utility would investigate their plant's emergency lighting systems, including a walk-around inspection of the emergency lighting system, review existing guidance and requirements, and develop proposed recommendations to enhance the operation and maintenance of emergency lighting systems. This is estimated to require 4 man-wks of utility labor. After this initial assessment, the utilities would then have to prepare a safety analysis of the proposed changes and revise maintenance procedures, if necessary. These activities are estimated to require an additional 4 man-wks including quality assurance activities, and management and legal reviews. The proposed SIR also includes installation of additional emergency lighting, where needed. This is estimated to require an additional \$20,000/plant for procurement of additional battery-powered lights plus 2 man-wk/plant to install the additional lights.

Industry Operation and Maintenance

The proposed SIR is assumed to include improved preventive maintenance and testing programs to ensure that emergency and essential lighting systems will operate as intended. The enhanced program is assumed to include periodic testing

of the lighting systems as well as replacement or repair of lighting system components (e.g., batteries, light bulbs) that are at or near their design lifetimes. An additional 1 man-wk/RV is estimated for these activities, which includes the costs for procurement and installation of replacement components when needed.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: GSI 158, Performance of Safety-Related, Power-Operated Valves Under Design-Basis Conditions

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: Nuclear power plants contain four types of power-operated valves (POVs), including motor-operated valves (MOVs), solenoid-operated valves (SOVs), air-operated valves (AOVs), and hydraulic-operated valves (HOVs). Inservice testing of these valves presently is performed under static conditions, i.e., zero differential pressure and flow conditions. Testing under static conditions does not always reveal valve performance under design-basis conditions. As a result, there is some uncertainty with regard to the operability of these valves when demanded to function under accident conditions. The resolution to this issue involves a combination of design reviews, improved surveillance/maintenance programs, valve testing, and adjusting valve actuation setpoints. A review of the design basis for operation of each POV is also part of the proposed resolution.

AFFECTED PLANTS: This issue affects all operating and planned plants.

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION =	2.4E+04
OCCUPATIONAL DOSES:	
SIR Implementation =	2.7E+02
SIR Operation/Maintenance =	0
Total of Above =	2.7E+02
Accident Avoidance =	1.9E+02

COST RESULTS (\$10⁶):

INDUSTRY COSTS:	
SIR Implementation =	1.2E+01
SIR Operation/Maintenance =	5.4E+00
Total of Above =	1.8E+01
Accident Avoidance =	1.5E+01
NRC COSTS:	
SIR Development =	3.0E-01
SIR Implementation =	1.0E+00
SIR Operation/Maintenance Review =	2.4E+00
Total of Above =	3.7E+00

NOTE: Intermediate results may not add exactly to totals due to rounding errors.

PERFORMANCE OF SAFETY-RELATED POWER-OPERATED VALVES
UNDER DESIGN-BASIS CONDITIONS

ISSUE 158

1.0 SAFETY ISSUE DESCRIPTION

This prioritization analysis is for Generic Safety Issue (GSI) 158, "Performance of Safety-Related Power-Operated Valves (POVs) Under Design Basis Conditions." There are four different types of POVs in nuclear power plants, including motor-operated valves (MOVs), air-operated valves (AOVs), solenoid-operated valves (SOVs), and hydraulic-operated valves (HOVs). Inservice testing requirements for these valves are found in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Presently, inservice testing is performed under static conditions, i.e., zero pressure differential and flow conditions. Static testing does not always reveal the performance of these valves under design-basis conditions. Therefore, there is some uncertainty regarding the operability of these valves when demanded to function during an accident.

The scope of this issue is limited to HOVs, AOVs, and SOVs. Issues related to MOVs are being addressed elsewhere (see Information Notice 90-40, "Results of NRC-Sponsored Testing of Motor-Operated Valves," dated June 5, 1990). The issue is also limited to the operability of valve operators and valves. The support systems for the valves, e.g., AC power, DC power, and Instrument Air Systems, are not included in the analysis. Furthermore, Marino (1993) states that relief and safety valves should not be included.

Operating experience reveals that POV failures have resulted from design, installation, and maintenance errors. POV failures are important to plant safety because they can result in unacceptable plant conditions such as safety system failures, unacceptable blowdown to vital plant areas, and pump deadheading due to loss of net positive suction head (NPSH). This is similar to the effects of MOV malfunctions. Therefore, the proposed resolution to this issue was developed by modifying the NRC's action plan for improving MOV performance (see Generic Letter 89-10) to account for design differences in the other types of POVs. The proposed resolution to this issue is described in the following paragraphs:

- a. Review and document the design basis for each safety-related AOV, SOV, and HOV.
- b. Establish correct valve settings (e.g., air flow control that regulates the rate of opening or closing of AOVs) based on results of (a) above.
- c. Change individual valve settings as appropriate. Test the valves at design-basis differential pressure and flow conditions where possible. This may not be possible for all safety-related POVs due to plant configurations. Provide documentation of the tests and justification for the tests that cannot be practically performed.
- d. Prepare or revise procedures to ensure that correct switch settings are determined and maintained throughout the life of the plant.

- e. Each safety-related POV failure and corrective action taken should be analyzed or justified and documented.

Note that these actions do not require hardware changes. It is assumed that the installed valves possess significant safety margins that encompass the changes in switch settings, etc., without exceeding the valves' failure limits.

2.0 SAFETY ISSUE RISK AND DOSE

- Public risk reduction and occupational dose are presented in Tables 1 and 2. Attachment 1 describes the assumptions and approach used for this analysis.

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

Performance of Safety-Related Power-Operated Valves Under Design-Basis Conditions (158)

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

All PWRs and BWRs are assumed to be affected.

	<u>N</u>	<u>$\bar{T}$ (yr)</u>
PWRs	74	24.7
BWRs	37	23.3
All plants	111	24.2

The number of plants and average remaining lifetimes were taken from Weakley, S. A., October 27, 1992. Memorandum to R. H. V. Gallucci entitled, "Update of U. S. Nuclear Power Plant's Average Remaining Lifetimes for Backfit VIA Calculations." Pacific Northwest Laboratory, Richland, Washington.

3. Plants Selected for Analysis:

The Oconee nuclear power plant is the representative plant for this analysis. Andrews, et al., (1983) provided the probabilistic risk assessment (PRA) information for the analysis.

4. Parameters Affected by SIR:

All AOVs, SOVs, and HOVs included in the Oconee PRA are affected by this issue. These parameters and their values are listed in Step 5.

5. Base-Case Values for Affected Parameters:

Air-operated valves were identified by Andrews, et al., (1983) to be present in the Oconee plant's emergency feedwater system (EFWS), hereinafter referred to as the Auxiliary Feedwater (AFW) System. HOVs and SOVs were not included in the dominant cut sets in the Oconee PRA (see Kolb, et al., 1981 and Andrews, et al., 1983). Since POVs are used in more than the AFW system, a review of the Sequoyah PRA (Bertucio and Brown 1990) was conducted to identify other systems that contain these types of valves. Modifications were then made to Oconee PRA to model the contributions from POVs in systems other than the AFW system (see Attachment 1). The affected parameters and base-case values were determined to be:

D (LP/CSIS Trn A) :	0.023	E (LP/CSIS Trn B) :	0.023
CONST1 (AFW) :	2.1E-4	CONST2 (AFW) :	6.3E-4
F1 (LPSWS, Trn B) :	0.0034	G1 (LPSWS, Trn A) :	0.0156

The bases for these parameter values are provided in Attachment 1.

NOTE: LPSWS = Low Pressure Service Water System
LP/CSIS = Low Pressure/Containment Spray Injection System

6. Affected Accident Sequences and Base-Case Frequencies:

The calculations in this prioritization analysis were performed using the OCONEE computer code.¹ Affected accident sequences and base-case frequencies were calculated directly by the code. This information is provided below (per Reactor-yr):

T_2MLU	=	1.42E-06	T_2MLUO	=	2.13E-05
T_1MLU	=	2.13E-06	S_2D	=	3.06E-07
T_2MQH	=	7.80E-07	T_1MLUO	=	1.42E-05
S_3H	=	7.09E-07	T_3MLUO	=	2.90E-06
S_1D	=	4.70E-06			

These values do not include the contributions from non-dominant minimal cut sets.

7. Affected Release Categories and Base-Case Frequencies:

The calculations in this prioritization analysis were performed using the OCONEE computer code. Affected release categories and base-case frequencies calculated by the code are shown below (per Reactor-yr):

PWR-1 = 5.50E-08	PWR-5 = 3.69E-07
PWR-3 = 2.39E-05	PWR-7 = 2.75E-05

These values include the contributions from non-dominant minimal cut sets.

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

$\bar{F}_{PWR} = 5.18E-05/RY$	$\bar{F}_{BWR} =$ Not calculated
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9. Base-Case, Affected Public Risk (W):

$W_{PWR} = 129.8$ person-rem/RY	$W_{BWR} =$ Not calculated
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10. Adjusted-Case Values for Affected Parameters:

The adjusted-case values for the affected parameters are derived in Attachment 1. The results are as follows:

¹ Daling, P. M. and T. B. Powers. 1991. Users Manual for the GRAND GULF and OCONEE PRA Computer Codes. Unpublished Draft Report. Pacific Northwest Laboratory, Richland, Washington.

D (LP/CSIS Trn A) :	0.022	E (LP/CSIS Trn B) :	0.022
CONST1 (AFW) :	1.84E-4	CONST2 (AFW) :	5.95E-04
F1 (LPSWS, Trn B) :	0.0032	G1 (LPSWS, Trn A) :	0.0154

11. Affected Accident Sequences and Adjusted-Case Frequencies:

The information provided below was taken from the OCONEE computer code output (per Reactor-yr):

T_2 MLU	=	1.26E-06	T_2 MLU0	=	1.99E-05
T_1 MLU	=	2.01E-06	S_2 D	=	2.80E-07
T_2 MQH	=	7.00E-07	T_1 MLU0	=	1.33E-05
S_3 H	=	6.37E-07	T_3 MLU0	=	2.70E-06
S_1 D	=	4.49E-06			

These values do not include the contributions from non-dominant minimal cut sets.

12. Affected Release Categories and Adjusted-Case Frequencies:

The affected release categories and adjusted-case frequencies were calculated directly by the code. The results are as follows:

PWR-1	=	5.25E-08	PWR-5	=	3.44E-07
PWR-3	=	2.22E-05	PWR-7	=	2.56E-05

These values include the contributions from non-dominant minimal cut sets.

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

$\bar{F}^*_{PWR}$	=	4.83E-05/Ry	F^*_{BWR}	=	Not calculated
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14. Adjusted-Case, Affected Public Risk (W^*):

W^*_{PWR}	=	120.9 person-rem/Ry	W^*_{BWR}	=	Not calculated
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15. Reduction in Core-Melt Frequency ($\Delta\bar{F}$):

$\Delta\bar{F}_{PWR}$	=	3.49E-06/Ry	$\Delta\bar{F}_{BWR}$	=	Not calculated
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16. Per-Plant Reduction in Public Risk (ΔW):

ΔW_{PWR}	=	8.9 person-rem/Ry	ΔW_{BWR}	=	Not calculated
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17. Total Public Risk Reduction, (ΔW)_{Total}:

Best Estimate (person-rem)	Error Bounds (person-rem)	
	Upper	Lower
2.4E+04	1.1E+07	0

These results were calculated assuming that the change in risk for a PWR is equal to that for a BWR. Uncertainty calculations are described in Andrews, et al., (1983, page 3.23).

ATTACHMENT 1 (To Table 1)

This attachment provides a detailed description of the approach that was used to calculate the base-case and adjusted-case CDF and public risks associated with this issue. As will be discussed, modifications were made to the Oconee PRA described in Andrews, et al., (1983) to enable the effects of the proposed resolution to be quantified. Therefore, the resulting PRA models no longer are directly applicable to a specific plant design.

BASE-CASE ANALYSIS

Air-operated valves were identified by Andrews, et al., (1983) to be present in the Oconee plant's emergency feedwater (EFW) system (referred to as the Auxiliary Feedwater [AFW] system in this analysis). HOVs and SOVs were not included in the dominant cut sets in the Oconee PRA (see Kolb, et al., 1981 and Andrews, et al., 1983). Since POVs are used in more than the AFW system, a review of the Sequoyah PRA (Bertucio and Brown 1990) was conducted to identify other systems that may contain other examples of these types of valves. The additional systems at Sequoyah that include POVs are:

- AOVs were found in the Essential Service Water (ESW), Low Pressure Injection/Recirculation (LPI/R) and Containment Spray (CS) systems.
- HOVs were found in the AFW system.
- SOVs were found in the Primary Pressure Relief System (PPRS) and are also used as Main Steam Isolation Valves (MSIVs).

Therefore, modifications to the Oconee PRA were necessary to effectively model the CDF and risks associated with AOVs, HOVs, and SOVs. It should be noted that SOVs did not appear in any systems other than the PPRS at Sequoyah and were not evident in any systems at Oconee. However, relief and safety valves were excluded from the scope of this issue because they are being addressed elsewhere. As a result, SOVs are not addressed further in this analysis.

From the above information, the affected parameters at Oconee were determined to be:

- Parameters D and E: Failure of Train A and Train B of the Low Pressure/Containment Spray Injection System (LP/CSIS), respectively.
- Parameters CONST1 and CONST2: Failure of the AFW system. CONST2 is for loss of offsite power (LOOP) transients and CONST1 is for transients other than LOOP.
- Parameters F1 and G1: Failure of Train A and Train B, respectively, of the Low Pressure Service Water (LPSW) system.

Parameters CONST1 and CONST2 already contained instances of AOVs and did not require modifications to the original PRA to model the base-case. For parameters D and E, a MOV was assumed to be changed to an AOV to model the base-case, similar to the configuration at Sequoyah in which one AOV is present in

each train of the LPI system. Modifications to parameters F1 and G1 were necessary to incorporate AOVs into the base-case model. These changes are explained further below.

Parameters D and E

The modifications to Parameters D and E consisted of replacing one normally-closed MOV in each train with an AOV. The unavailability of this valve, as given in the Oconee PRA (Andrews, et al., 1983, p. A.10) was calculated as follows:

Hardware	-	0.001
Plugged	-	0.0001
Control circuitry	-	0.0064
Maintenance outage	-	<u>0.0021</u>
		0.0096

The remaining contributors to the values for D and E, including the failure probabilities for check valves, manual valves, and a pump, were unaffected and therefore not changed. The base-case values are 0.023/demand for both parameters D and E. Modifications were made to the AOV failure probability shown above to model the effects of the proposed resolution (i.e., to develop the adjusted-case values for affected parameters). These modifications will be discussed later.

Parameters CONST1 and CONST2

The base-case values for parameters CONST1 and CONST2 were not changed because the Oconee AFW system models already contained AOVs. Therefore, the base-case value for CONST1 is 2.1E-04/demand and for CONST2 is 6.3E-04/demand. Modifications were made to these values to model the effects of the proposed resolution (i.e., for the adjusted-case).

Parameters F1 and G1

The Oconee model for events F1 and G1 did not include any POVs. The original analysis of these events (Andrews, et al., 1983, p. A.15) indicated the failure probabilities for both of these parameters were the sum of the failure probabilities of two pumps. The failure probability for F1 was the sum of the sum of the failure probabilities of two normally-operating pumps. The failure probability for each pump was given to be 7.2E-04/demand so the failure probability for F1 was (7.2E-04 + 7.2E-04) or 1.4E-03/demand. This was modified to include the failure probability for an AOV in each pump line. Bertucio and Brown (1990) gave the failure probability for an AOV to be 1E-03/demand. Therefore, the modified base-case failure probability for F1 is:

$$F1 = (7.2E-04 + 1E-03) * 2 = 3.44E-03/demand$$

Similarly, the failure probability for G1 was given by Andrews, et al., (1983) to be the sum of the failure probabilities for two normally-idle pumps. The failure probability for each pump was given to be 0.0068/demand so the total failure probability for G1 was calculated to be (0.0068 * 2) or 0.014/demand. The modified base-case value for G1 was calculated similarly to that for F1, i.e., an AOV was assumed to be located in each pump line. Therefore, the modified base-case failure probability for G1 is:

$$G1 = (0.0068 + 1E-03) * 2 = 1.56E-02/\text{demand}.$$

ADJUSTED-CASE ANALYSIS

The adjusted-case represents the situation in which the plants have implemented the proposed resolution. The effects of the proposed resolution will be to reduce the failure probability of the POVs in the affected parameters. The amount of improvement in POV reliability was assumed to be a factor of 4 reduction in the contributions of operator errors and hardware failures to the total valve failure probability. This reduction is consistent with reductions taken in other hardware-related safety issues (see Emrit, et al., 1983). Maintenance contributions were assumed to be unaffected. This is a conservative assumption in that maintenance outage failure probabilities should increase as a result of an enhanced testing/maintenance program. This would tend to offset the improvements in POV hardware and operator error probabilities that will result from implementation of the proposed resolution. The calculations that were performed to develop the adjusted-case values for the affected parameters are described below.

Parameters D and E

The adjusted-case failure probability for the AOV that was assumed to replace the MOV in the LP/CSIS system was assumed to improve as a result of the proposed resolution. The contributions to hardware and operator error probabilities were assumed to be reduced by a factor of 4. Therefore, the adjusted-case failure probability for this AOV was calculated as follows:

Hardware	-	0.001 * 0.25	=	0.00025
Plugged	-	0.0001 * 0.25	=	0.000025
Control circuitry	-	0.0064	=	0.0064
Maintenance outage	-	<u>0.0021</u>	=	<u>0.0021</u>
		0.0096	=	0.0089
		[Base-case]		[Adjusted-case]

The adjusted-case value for the AOV was substituted into the calculations for parameter D and the resulting adjusted-case value for D was calculated to be 0.022/demand. The calculations are identical for event E.

Parameters CONST1 and CONST2

The calculations for CONST1 and CONST2 are somewhat complicated (see Andrews, et al., 1983, Section 6.1.2.1 and pages A.19 to A.24) and will not be repeated here. The approach was the same as that used for parameters D and E, i.e., recalculating the AOV failure probability assuming the factor of four decrease in AOV failure probabilities and then recalculating CONST1 and CONST2 using the new values for the AOVs. The adjusted-case values for these parameters were calculated to be 1.84E-04/demand and 5.95E-04/demand for CONST1 and CONST2, respectively.

Parameters F1 and G1

The base-case failure probabilities for AOVs were specified in the Oconee PRA in terms of the contributions of hardware failures, operator errors, etc. The first step in this analysis was to establish the contributing failure causes

and their probabilities for the AOVs added to parameters F1 and G1. Andrews, et al., (1983, page A.22) provided the following breakdown for an AOV:

Operator error	-	0.001	(14.5% of total failure probability)
Plugged	-	0.0001	(1.5%)
Maintenance	-	<u>0.0058</u>	(84.1)
TOTAL:		<u>0.0069</u>	

The fractions due to the three failure causes were assumed to be applicable to parameters F1 and G1. Therefore, after normalizing to the original (base-case) failure probability of the AOV that was added to F1 and G1 (failure probability = $1\text{E-}03/\text{demand}$), the contributions due each of these causes is as follows:

Operator error	-	$(0.145 * 1\text{E-}03)$	=	$1.45\text{E-}04$
Plugged	-	$(0.015 * 1\text{E-}03)$	=	$1.45\text{E-}05$
Maintenance	-	$(0.841 * 1\text{E-}03)$	=	<u>$8.41\text{E-}04$</u>
TOTAL:				<u>$1\text{E-}03/\text{demand}$</u>

To model the effects of the proposed resolution, the operator error and plugged failure causes were reduced by a factor of 4. The revised failure probabilities for the AOVs are per demand. The calculations leading to this value are illustrated below:

Operator error	-	$(1.45\text{E-}04 * 0.25)$	=	$3.63\text{E-}05$
Plugged	-	$(1.45\text{E-}05 * 0.25)$	=	$3.63\text{E-}06$
Maintenance	-	$(8.41\text{E-}04 * 1.0)$	=	<u>$8.41\text{E-}04$</u>
TOTAL:				<u>$8.81\text{E-}04/\text{demand}$</u>

Using the adjusted-case failure probability for the AOV, the adjusted-case failure probabilities for parameters F1 and G1 are as follows:

F1	=	$(7.2\text{E-}04 + 8.81\text{E-}04) * 2$	=	$3.20\text{E-}03/\text{demand}$
G1	=	$(6.68\text{E-}03 + 8.81\text{E-}04) * 2$	=	$1.51\text{E-}03/\text{demand}$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Performance of Safety-Related Power-Operated Valves Under Design-Basis Conditions (158)

2. Affected Plants (N):

	<u>N</u>
PWRs	74
BWRs	37
All plants	111

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

	<u>$\bar{T}$</u>
PWRs	24.7 yr
BWRs	23.3 yr
All plants	24.2 yr

See Table 1, Step 2, for derivation of these values.

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

$$\Delta\bar{F}D_R \text{ (PWR)} = (19,860 \text{ person-rem})(3.5\text{E-}06/\text{RY}) = 6.9\text{E-}02 \text{ person-rem/Ry}$$

The value for D_R , the best estimate of the occupational dose due to reactor cleanup and repair following an accident, was taken from Andrews, et al., (1983, page D.6).

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

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
1.9E+02	1.7E+04	0

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

SIR implementation primarily involves administrative and review activities that will take place in non-radiation areas. However, two activities will require work in radiation zones, namely POV testing and revising switch settings where necessary. To estimate the amount of labor in radiation zones, it was assumed that a total of twenty valves are affected; ten are located in containment and ten outside containment. Valve testing is assumed to require a two-person team 4 hours to complete the activity. Changing valve setting is assumed to require a two-person team 0.5 hr/valve.

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

D was calculated in four parts, as shown below:

Testing

Inside containment (10 valves)(2 persons)(4 hr)(25 mRem/hr)

Outside containment (10 valves)(2 persons)(4 hr)(2.5 mRem/hr)

Change switch settings

Inside containment (10 valves)(2 persons)(0.5 hr)(25 mRem/hr)

Outside containment (10 valves)(2 persons)(0.5 hr)(2.5 mRem/hr)

$$D = 2000 + 200 + 250 + 25 = 2.48 \text{ person-rem/plant}$$

Dose rates inside and outside containment were taken from Andrews, et al., (1983).

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

$$ND = (2.48 \text{ person-rem/plant})(111 \text{ plants}) = 2.75E+02 \text{ person-rem}$$

9. Per-Plant Utility Labor in Radiation Zones for SIR Operation and Maintenance:

No additional utility labor in radiation zones, beyond the presently performed testing and maintenance, results from SIR operation and maintenance. Therefore, the additional doses due to SIR operation and maintenance are zero.

10. Per-Plant Radiation Dose Increase for SIR Operation and Maintenance

$$D_0 = 0$$

11. Total Radiation Dose Increase for SIR Operation and Maintenance

$$NTD_0 = 0$$

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
2.75E+02	8.2E+02	9.2E+01

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions (158).

2. Affected Plants (N):

	<u>N</u>
PWRs	74
BWRs	37
All Plants	111

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

	<u>$\bar{T}$</u>
PWRs	24.7 yr
BWRs	23.3 yr
All plants	24.2 yr

See Table 1, Step 2, for derivation of these values.

Industry Costs (Steps 4 through 12):

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, $\Delta(\bar{F}A)$:

$$\Delta(\bar{F}A)_{\text{PWR}} = (\$1.65\text{E}+09)(3.5\text{E}-06/\text{RY}) = \$5.8\text{E}+03/\text{RY}$$

The value for A, the best estimate of the industry cost for reactor cleanup, repair, and replacement power following a core damage accident, was taken from Andrews, et al., (1983, page E.2).

5. Total Industry Cost Savings Due to Accident Avoidance (ΔH):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
1.5E+07	1.4E+09	0

6. Per-Plant Industry Resources for SIR Implementation:

Industry resources for implementation of SIR are needed to complete administrative and design review activities, revise procedures, perform POV tests, and revise POV switch settings. The proposed resolution of this issue is similar to the MOV program described in Generic Letter 89-10. The administrative, design review, and procedural activities were

estimated in GL 89-10 to require 2000 person-hrs per plant. Resources for POV testing and revising valve switches were estimated in Table 2, Step 7 at about 225 person-hrs/plant. Therefore, the total labor requirements for SIR implementation were estimated to be about 2225 person-hrs/plant.

7. Per-Plant Industry Cost for SIR Implementation (I):

$$\begin{aligned} I &= (2225 \text{ person-hrs/plant})(\$50/\text{person-hr}) \\ &= \$1.11\text{E}+05/\text{plant} \end{aligned}$$

Basic labor rates were taken from Andrews, et al., (1983). The burdened, hourly labor rate was calculated using \$100,000/person-yr annual rate and assuming that one person-yr = 2000 person-hrs.

8. Total Industry Cost for SIR Implementation (NI):

$$NI = (\$1.11\text{E}+05/\text{plant})(111 \text{ plants}) = \$1.23\text{E}+07$$

9. Per-Plant Industry Resources for SIR Operation and Maintenance:

The proposed resolution includes a periodic review of valve data to detect valve failure trends. The trends could provide the basis for revisions to testing frequencies and for assessing the adequacy of the revised valve settings. It was estimated that this periodic review will require approximately 1 person-wk/Ry (40 person-hr/Ry).

10. Per-Plant Industry Costs for SIR Operation and Maintenance:

$$I_o = (40 \text{ person-hr/Ry})(\$50/\text{person-hr}) = \$2000/\text{Ry}$$

11. Total Industry Cost for SIR Operation and Maintenance

$$NTI_o = (\$2000/\text{Ry})(111 \text{ plants})(24.2 \text{ yr}) = \$5.37\text{E}+06$$

12. Total Industry Cost (S_I):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$1.8E+07	\$2.4E+07	\$1.1E+07

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC resources are needed to support a research and development program designed to test POVs in a manner similar to the MOV tests described in GL 89-10. The POV study is estimated to require approximately 30 man-months of contractor labor and 6 man-months of NRC labor. The testing equipment developed for the MOV study was assumed to be applicable to the POV study so procurement of additional testing hardware was assumed to be unnecessary.

14. Total NRC Cost for SIR Development (C_D):

$$C_D = (36 \text{ man-month})(\$8,330/\text{man-month}) = \$3.00\text{E}+05$$

15. Per-Plant NRC Labor for Support of SIR Implementation:

NRC resource requirements to support SIR implementation were estimated to be 4 person-wk per plant.

16. Per-Plant NRC Cost for Support of SIR Implementation (C):

$$C = (4 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$9080/\text{plant}$$

17. Total NRC Cost for Support of SIR Implementation (NC):

$$NC = (\$9080/\text{plant})(111 \text{ plants}) = \$1.01\text{E}+06$$

18. Per-Plant NRC Labor for Review of SIR Operation and Maintenance:

NRC resource requirements for support of SIR implementation were estimated to be approximately 2 man-day/RY.

19. Per-Plant NRC Cost for Review of SIR Operation and Maintenance C_o :

$$\begin{aligned} C_o &= [(2 \text{ man-day/RY})/(5 \text{ man-day/man-wk})] (\$2270/\text{man-wk}) \\ &= \$908/\text{RY} \end{aligned}$$

20. Total NRC Cost for Review of SIR Operation and Maintenance ($N\bar{T}C_o$):

$$N\bar{T}C_o = (\$908/\text{RY})(111 \text{ plants})(24.2 \text{ yr}) = \$2.44\text{E}+06$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$3.7E+06	\$5.1E+06	\$2.4E+06

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 160, Spurious Actuation of Instrumentation Upon Restoration of Power

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: This issue is related to reenergization of instruments, after a loss of power to single channel interlocks or single channel permissives (or simultaneous loss of power to multiple channels (e.g., loss of offsite power)). Spurious actuations of ESF components have occurred upon restoration of power when instrumentation voltage comparison devices which feed pressure isolation valves sense a rapid restoration of the reference voltage and delays in signal voltage restoration due to circuitry delays. The voltage difference sensed by the comparison circuit exceeds the set point value as the signal voltage returns to the full normal value, within the normal tolerance band of the reference voltage. Under such conditions, and with the appropriate other permissives such as power available to the ESF pump bus, the pressure isolation valves can open leaving a check valve as the only barrier preventing an interfacing system LOCA (ISLOCA). This issue could also contribute to the likelihood of a Low Temperature Overpressurization (LTOP) event.

AFFECTED PLANTS: This issue affects all operating and planned plants.

RISK/DOSE RESULTS (man-rem):

PUBLIC RISK REDUCTION = 4.7E+3

OCCUPATIONAL DOSES:

SIR Implementation =	1.8E+3
SIR Operation/Maintenance =	1.4E+4
Total of Above =	1.6E+4
Accident Avoidance =	2.0E+1

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation =	6.7E+1
SIR Operation/Maintenance =	0
Total of Above =	6.7E+1
Accident Avoidance =	1.6E+0

NRC COSTS:

SIR Development =	5.0E-1
SIR Implementation =	1.0E+0
SIR Operation/Maintenance Review =	2.4E+0
Total of Above =	4.0E+0

SPURIOUS ACTUATION OF INSTRUMENTATION UPON RESTORATION OF POWER

ISSUE 160

1.0 SAFETY ISSUE DESCRIPTION

The prioritization analysis is for Generic Safety Issue 160, "Spurious Actuation of Instrumentation Upon Restoration of Power." The potential for inadvertent ESF actuations to contribute to interfacing system loss of coolant accidents (ISLOCA, GSI 105), and low-temperature overpressurization (LTOP, GSI 94) accident sequences are addressed. Safety equipment such as pressure isolation valves unexpectedly change positions. Spurious actuations of instrumentation upon restoration of power can cause ESF actuations when this instrumentation has been placed on an interruptable bus, which is often the case based on the "fail safe" response of the instrument upon a loss of power. The response of such instruments to a power restoration may cause undesirable response of ESF equipment.

Investigations of inadvertent ESF equipment actuation events at operating nuclear power plants has indicated these actuations can occur when ESF circuitry provides an actuation signal based on comparison of a reference voltage to a signal voltage. When power is lost to such instruments (i.e., due to test, maintenance, power supply or other component failures, or other loss of power conditions), the reference voltage and signal voltage are restored at different rates after power is restored. The reference voltage is typically restored almost immediately, but return of signal voltage is delayed due to the normal circuitry delays. This delay sets up a difference in the ESF actuation comparison circuit or device which can cause ESF equipment actuation when other permissives, interlocks, and requirements are satisfied (e.g., power on the ESF equipment bus). If the ESF actuation setpoint is for a condition of signal voltage less than reference voltage, then the setpoint will be met upon return of power. If, on the other hand, the ESF actuation setpoint is for a condition of signal voltage greater than reference voltage, then the setpoint will not be met upon return of power. In a Boiling Water Reactor (BWR), ESF actuation circuits such as Reactor Low Water Level and Low Reactor Pressure are typically designed to actuate when signal voltage is less than reference voltage by the setpoint amount. This condition can contribute to the ISLOCA event sequence frequency when the ESF actuation causes normally closed pressure isolation valves to open inadvertently.

Spurious actuations of ESF components have also been caused when energizing DC equipment during maintenance and testing activities. High output voltage may be experienced after battery chargers are re-energized, resulting in a trip of the charger. Chargers may overshoot their normal output voltage if they are re-energized when there is no electrical load on the charger. The charger trip may result in an overvoltage trip of the AC/DC inverter which feeds the ECCS trip units. After the trip units are reset, they recognize the change of state as an ECCS initiation signal and the trip units initiate various ECCS responses, including initiating one channel of the core spray system, low pressure coolant injection system, high pressure coolant injection system, LOCA load shedding of the affected vital AC bus, and start of an emergency diesel generator (EDG). In addition, the LPCI isolation valves are opened, providing the opportunity for an ISLOCA event.

Potential solutions to resolve this issue of spurious actuations of instrumentation upon restoration of power causing inadvertent ESF actuations, in particular inadvertent opening of pressure isolation valves and corresponding increased risk of ISLOCA and LTOP accident sequences are as described below.

1. Improve analog transmitter trip unit power supply reliability.
2. improve operator training and procedures to respond to ISLOCA and LTOP events.
3. Improve the design of power supplies and ESF actuation circuitry so that spurious actuation that opens pressure isolation valves does not occur upon restoration of power.
4. Implement auctioneering circuit from two analog transmitter trip unit voltage converters, one converter taking input from a 120V AC instrument bus and the other converter taking input from a 125V DC source.
5. Use a diverse interlock for pressure isolation valve opening. For example, use a contact from the RHR pump motor breaker to signal that the RHR pump is running, where this signal is not dependent on any power supply.

2.0 SAFETY ISSUE RISK AND DOSE

- Public risk reduction and occupational dose are presented in Tables 1 and 2. Attachment 1 describes the assumptions and approach used for this analysis.

TABLE 1. Public Risk Reduction Worksheet

1. Title and Identification Number of Safety Issue:

Spurious Actuation of Instrumentation Upon Restoration of Power (160)

2. Affected Plants (N) and Average Remaining Lives ($\bar{T}$):

All PWRs and BWRs are assumed to be affected.

	<u>N</u>	<u>$\bar{T}$ (yr)</u>
PWRs	74	24.7
BWRs	37	23.3

The number of plants and average remaining lifetimes were taken from Weakley, S. A., October 27, 1992. Memorandum to R. H. V. Gallucci entitled, "Update of U. S. Nuclear Power Plant's Average Remaining Lifetimes for Backfit VIA Calculations." Pacific Northwest Laboratory, Richland, Washington.

3. Plants Selected for Analysis:

A generic plant was used in this analysis. Data was taken from Bertucio and Brown (1990) and Sugnet, et al., (1984) to address potential ISLOCA sequences and from Gore, et al., (1988) to address LTOP sequences.

4. Parameters Affected by SIR:

The issue was not addressed in the existing PRAs. Therefore, new accident sequence analyses based on the existing PRAs were performed. See Attachment 1 for an explanation of the calculational procedure.

5. Base-Case Values for Affected Parameters:

See Attachment 1.

6. Affected Accident Sequences and Base-Case Frequencies:

See Attachment 1 for the derivation of the following values.

<u>Accident Sequence</u>	<u>Base-Case Affected CDF, per RY</u>	
	<u>PWR</u>	<u>BWR</u>
LTOP	1.1E-6	NA
ISLOCA	2.9E-8	2.9E-8
Total:	1.1E-6	2.9E-8

7. Affected Release Categories and Base-Case Frequencies:

A generic release category and consequence value was applied to both LTOP and ISLOCA sequence frequencies (See Attachment 1).

8. Base-Case, Affected Core-Melt Frequency ($\bar{F}$):

$$\bar{F}_{PWR} = 1.1E-6/RY \qquad \bar{F}_{BWR} = 2.9E-8/RY$$

9. Base-Case, Affected Public Risk (W):

See Attachment 1 for the derivation of the following values.

$$W_{PWR} = 5.1E+0 \text{ person-rem/RY} \qquad W_{BWR} = 1.3E-1 \text{ person-rem/RY}$$

10. Adjusted-Case Values for Affected Parameters:

An 50% reduction in the initiating event frequency was estimated to result from implementation of the resolution to this issue (See Attachment 1).

11. Affected Accident Sequences and Adjusted-Case Frequencies:

<u>Accident Sequence</u>	<u>Adjusted-Case Affected CDF, per RY</u>	
	<u>PWR</u>	<u>BWR</u>
LTOP	5.5E-7	NA
ISLOCA	<u>1.5E-8</u>	<u>1.5E-8</u>
Total:	<u>5.6E-7</u>	<u>1.5E-8</u>

See Attachment 1.

12. Affected Release Categories and Adjusted-Case Frequencies:

See Step 7.

13. Adjusted-Case, Affected Core Melt Frequency ($\bar{F}^*$):

$$\bar{F}^*_{PWR} = 5.6E-7/RY \qquad \bar{F}^*_{BWR} = 1.5E-8/RY$$

14. Adjusted-Case, Affected Public Risk (W^*):

See Attachment 1 for the derivation of the following values.

$$W^*_{PWR} = 2.5E+0 \text{ person-rem/RY} \qquad W^*_{BWR} = 6.5E-2 \text{ person-rem/RY}$$

15. Reduction in Core-Melt Frequency ($\Delta\bar{F}$):

$$\Delta\bar{F}_{PWR} = 5.4E-7/RY \qquad \Delta\bar{F}_{BWR} = 1.4E-8/RY$$

16. Per-Plant Reduction in Public Risk (ΔW):

$$\Delta W_{PWR} = 2.6E+0 \text{ person-rem/RY} \qquad \Delta W_{BWR} = 6.5E-2 \text{ person-rem/RY}$$

17. Total Public Risk Reduction, $(\Delta W)_{\text{Total}}$:

Best Estimate (person-rem)	Error Bounds (person-rem)	
	<u>Upper</u>	<u>Lower</u>
4.7E+3	2.8E+5	0

ATTACHMENT I

The risk associated with this issue for light water reactors is estimated based on the Oconee PRA (Sugnet, et al., 1984; Gore, et al., 1988; and Andrews et al., 1983). The key assumptions and calculational procedure are described in this attachment. The information requirements and final results are presented as required by Andrews, et al., (1983).

An ISLOCA is characterized by a failure of valves that protect low pressure injection (LPI) system piping from the high pressures of the reactor coolant system (RCS). Specific design configurations are plant-specific. For this analysis, the LPI pipe segments attached to the RCS were assumed to contain two check valves and a normally-closed motor-operated valve (MOV). The ECCS actuation system logic was assumed to open the MOV to allow LPI flow to the reactor vessel when system pressures are below the LPI pump shutoff head. An ISLOCA can occur only if both check valves and the MOV are opened simultaneously and the RCS pressure is above the rated pressure of the LPI piping downstream of the outboard check valve. This would allow the high pressure primary coolant to flow from the reactor vessel into low-pressure-rated LPI piping, possibly causing failure of the LPI piping and loss of reactor coolant through the failed LPI system.

Low-temperature overpressure (LTOP) transients may be caused when water is inadvertently injected into the reactor vessel. Such an occurrence could increase the pressure in the reactor vessel, thus increasing the likelihood of a reactor vessel failure. Spurious actuations of ESF systems upon restoration of power may cause an inadvertent safety injection which could result in an overpressurization. Therefore, this issue may increase the probability of an LTOP event. LTOP is applicable only to PWRs.

Base-Case Analysis

The key assumptions that were used to derive the estimates in Table 1 are as follows:

1. It was assumed that the ISLOCA frequency estimated in the original PRAs does not include contribution from spurious actuation of ESF equipment due to a loss and restoration of power as described above.
2. It was assumed that the industry experience gathered through licensee event reports and reported in NRC Information Notice 90-22, "Unanticipated Equipment Actuations Following Restoration of Power to Rosemount Transmitter Trip Units", dated March 23, 1990 captures all instances of spurious actuations of ESF equipment due to a loss and restoration of power. The events are described more fully by Murley (1991). For this bounding case evaluation, the number of events described by Murley (1991) was doubled to account for possible unreported or otherwise unknown events. This effectively doubles the frequency of spurious actuations of ESF equipment.
3. It was assumed that the occurrence of ISLOCA and LTOP event sequences results in core damage and containment bypass.

4. It was assumed that the ISLOCA-related issue presents an equal challenge to PWRs and BWRs even though the industry data described in NRC Information Notice 90-22 describes only BWR experiences. LTOP affects only PWRs.

The base case risk affected core damage frequency and public risk associated with this issue was determined as follows. The industry information for spurious actuations of instrumentation upon restoration of power as described in NRC Information Notice 90-22 (see Murley 1991) and records of the total number of reactor operation years during the time period of these incidents was used to determine the frequency of this incident. The total reactor-years through 1991 were estimated based on the elapsed time between each reactor's date of commercial operation through 1991. The number of operating years for each plant type were calculated to be:

<u>Plant Type</u>	<u>Operating Reactor-yrs</u>
Westinghouse	649
Combustion Engineering	185
Babcock and Wilcox	<u>138</u>
TOTAL PWR:	<u>972</u>
General Electric	<u>523</u>
TOTAL BWR:	<u>523</u>
TOTAL (ALL PLANTS):	1,495

A total of seven events were identified by Murley (1991), including four events at one plant, 2 at a second plant, and one at a third. This number of events was multiplied by a factor of two to account for possibly missing event data and other uncertainties. Therefore, a total of 14 events was estimated to have occurred through 1991. Since the ISLOCA sequence is applicable only when the RCS pressure is greater than the LPI system pressure, a factor of 0.8 was applied to the number of operating years (Sugnet, et al., 1984, p. F-11). Therefore, the frequency of spurious actuations of ESF components upon restoration of power was calculated to be:

$$\begin{aligned}\text{Frequency} &= (14 \text{ events}) / (1495 \text{ RY} * 0.8) \\ &= 0.0117/\text{RY}\end{aligned}$$

The opportunity exists for operators to recover from the spurious ESF actuations before the malfunctions progress to ISLOCA or LTOP events. All seven of the events described by Murley (1991) were prevented from progressing to ISLOCA or LTOP events. Assuming that the next event that occurs will not be prevented by operators, the operator error probability will be 1/8 or 0.125.

The next step was to estimate the probabilities that the unrecovered spurious actuation events will progress to core damage. For an ISLOCA to occur, the valves on the LPI injection lines must be inadvertently opened or fail open (rupture or leak). A typical LPI injection line contains a motor-operated valve (MOV) and two check valves. The initiating event was assumed to cause the MOV to

open, leaving the two check valves to protect the low pressure lines from the high-pressure RCS. The failure probability for each check valve, $1E-4/\text{demand}$, was taken from Bertucio and Brown (1990, p. 4.9-24) and was used to calculate the failure probability for the two-valve system as follows.

The potential for common cause failure of the two check valves exists and must be accounted for in the calculations. The "Beta Factor" method was used to calculate the total unavailability of the two-valve system. In this method, the failure probability for the two-valve system is calculated as follows:

$$U(2 \text{ Valves}) = U(\text{Valve } 1) * U(\text{Valve } 2) + \beta U(\text{Valve})$$

where β = Factor accounting for common cause failure of 2 valves
 U = Unavailability

Since $U(\text{Valve } 1) = U(\text{Valve } 2)$, the formula can be rewritten:

$$U(2 \text{ Valves}) = U(\text{Valve})^2 + \beta U(\text{Valve})$$

The Beta factor for a 2-valve system was taken from Bertucio and Brown (1990, p. 4.9-8). This document did not quantify Beta factors for a 2 check valve system so the Beta factor for common cause failure of 2 air-operated valves was used. This Beta factor was 0.10. Therefore, the total unavailability of the 2 check valve system was calculated to be:

$$\begin{aligned} U(2 \text{ valves}) &= (1E-4)^2 + (0.1)(1E-4) \\ &= 1E-5/\text{demand} \end{aligned}$$

The base-case affected core damage frequency (CDF) for ISLOCA sequences was calculated by combining the frequency of spurious actuations, operator recovery failure probability, and check valve failure probabilities. Since there are two injection lines, the resulting frequency was doubled. The ISLOCA affected CDF was calculated to be:

$$\begin{aligned} \text{ISLOCA CDF} &= 2 (0.0117/\text{RY})(0.125)(1E-5) \\ &= 2.9E-8/\text{RY} \end{aligned}$$

The base-case affected CDF for LTOP events was calculated using information provided by Gore, et al., (1988). Gore, et al., presents estimates of reactor vessel failure probabilities and public risks associated with LTOP events. Overpressurization frequencies and spectra developed by Gore, et al., (1988) were assumed to be applicable to the accident sequences initiated by spurious actuations of ESF components upon restoration of power. Therefore, the "base-case" data in Table 9.1 of Gore, et al., (1988) was used in this prioritization analysis. Modifications were made to the table to account for the frequency of spurious ESF equipment actuations and for the average remaining lifetimes of affected plants. The revised table is presented in Table 1.

The total affected CDF is the sum of the ISLOCA and LTOP sequence frequencies for PWRs and the ISLOCA frequency for BWRs, which are not vulnerable to LTOP events. The affected CDFs are as follows:

<u>Accident Sequence</u>	<u>Base-Case Affected CDF, per RY</u>	
	<u>PWR</u>	<u>BWR</u>
LTOP	1.1E-6	NA
ISLOCA	<u>2.9E-8</u>	<u>2.9E-8</u>
Total:	<u>1.1E-6</u>	<u>2.9E-8</u>

The base-case affected public risk was calculated using the consequence assessment results provided by Gore, et al., (1988). The total 30-yr integrated radiation dose of 4.5E+6 person-rem/event was given in Table 9.1 of Gore, et al., (1988). This consequence estimate was multiplied by the affected CDF values given above to calculate the affected public risk values shown below:

<u>Accident Sequence</u>	<u>Base-Case Affected CDF, per RY</u>	
	<u>PWR</u>	<u>BWR</u>
LTOP	5.0E+0	NA
ISLOCA	<u>1.3E-1</u>	<u>1.3E-1</u>
Total:	<u>5.1E+0</u>	<u>1.3E-1</u>

Adjusted-Case Analysis

The adjusted case CDF and public risk values were determined as follows. First, it was estimated that the initiator frequency can be reduced by as much as 50% as a result of implementation of the safety issue resolution described previously. This fifty percent reduction represents that portion of the spurious actuations of instrumentation upon restoration of power which are eliminated by the potential resolution. The remaining fifty percent is not eliminated by the resolution. This fifty percent reduction includes the reduction from hardware improvements as well as improved operating and maintenance procedures to deal with spurious ESF actuations. This 50% reduction is applicable across the board to both LTOP and ISLOCA frequencies. Therefore, the adjusted-case CDF and public risk values were calculated by multiplying the base-case affected CDF and public risk values by 0.5. The results are as follows:

	<u>Adjusted-Case Affected CDF, per RY</u>		<u>Adjusted-Case Affected Risk, person-rem/RY</u>	
	<u>PWR</u>	<u>BWR</u>	<u>PWR</u>	<u>BWR</u>
LTOP	5.5E-7	NA	2.5E+0	NA
ISLOCA	<u>1.5E-8</u>	<u>1.5E-8</u>	<u>6.5E-2</u>	<u>6.5E-2</u>
Total:	<u>5.6E-7</u>	<u>1.5E-8</u>	<u>2.5E+0</u>	<u>6.5E-2</u>

TABLE 1

Overpressure Mitigation System	Challenge Frequency*	Operator Error*	Overpressure Spectrum		NIVFP Value by Pressure	VFP Contribution by Pressure	Category Total VFP	Reactor-ysrs from 1992 to EOL	Average VFP per plant per year
			psi	%					
PORV	0.0117	0.125	2500	9	3.79	4.99E-04 ^(a)	^(b) 5.25E-04	775.2 ^(c)	6.77E-07 ^(d)
			1400	9	0.19	2.50E-05			
			850	13	0.0037	7.03E-07			
			<600	69	0.0002	2.02E-07			
RHR	0.0117	0.125	2500	14	0.68	1.39E-04	1.40E-04	361.6	3.86E-07
			850	14	0.0009	1.84E-07			
			<600	72	0.0001	1.05E-07			
PORV + N2	0.0117	0.125	850	5	0.0002	1.46E-08	2.85E-08	146.4	1.95E-10
			<600	95	0.00001	1.39E-08			
TOTAL							6.64E-04		1.06E-06

NIVP = Normalized integrated vessel fracture probability; EOL = End-of-life

Source: Gore, et al., 1988, Table 9.1. Values calculated in this prioritization analysis are indicated by asterisk (*).

- (a) Values in this column were calculated by multiplying together the challenge frequency, operator error probability, overpressure spectrum percentage, and NIVP value by pressure.
- (b) Values in this column are the sum of the VFP contributions by pressure.
- (c) Values in this column were taken from Gore, et al., (1988) and then adjusted to account for the 6 yrs of operating life that has occurred between the data presented by Gore, et al., (1986) and now. A 20% decrease in the remaining operational lifetimes was applied to the data presented by Gore, et al.
- (d) Values in this column were calculated by dividing the Category total VFP by the reactor-ysrs from 1991 to EOL.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Spurious Actuation of Instrumentation Upon Restoration of Power (160)

2. Affected Plants (N):

	<u>N</u>
PWRs	74
BWRs	37
All plants	111

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

	<u>$\bar{T}$</u>
PWRs	24.7 yr
BWRs	23.3 yr

See Table 1, Step 2, for derivation of these values.

4. Per-Plant Occupational Dose Reduction due to Accident Avoidance, $\Delta(\bar{F}D_R)$:

$$\begin{aligned}\Delta\bar{F}D_R \text{ (PWR)} &= (19,900 \text{ person-rem})(5.4E-7/\text{RY}) \\ &= 1.08E-2 \text{ person-rem/Ry}\end{aligned}$$

$$\begin{aligned}\Delta\bar{F}D_R \text{ (BWR)} &= (19,900 \text{ person-rem})(1.4E-8/\text{RY}) \\ &= 2.79E-4 \text{ person-rem/Ry}\end{aligned}$$

5. Total Occupational Dose Reduction Due to Accident Avoidance (ΔU):

<u>Best Estimate</u> <u>(person-rem)</u>	<u>Error Bounds (person-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
2.0E+1	2.4E+2	0

6. Per-Plant Utility Labor in Radiation Zones for SIR Implementation:

It was estimated that utility labor in radiation zones will be required for installation of new equipment. A total of 3 man-wk/plant was estimated. A 75% utilization factor was assumed for labor in radiation zones.

$$\text{Labor} = (3 \text{ man-wk/plant})(40 \text{ man-hr/man-wk})/(0.75) = 160 \text{ man-hr/plant}$$

7. Per-Plant Occupational Dose Increase for SIR Implementation (D):

It was assumed that the labor occurs in a 100 mrem/hr radiation zone.

$$D = (160 \text{ man-hr/plant})(100 \text{ mrem/hr})(\text{rem}/1000 \text{ mrem}) = 16 \text{ man-rem/plant}$$

8. Total Occupational Dose Increase for SIR Implementation (ND):

$$ND = (111 \text{ plants})(16 \text{ man-rem/plant}) = 1.8E+3 \text{ person-rem}$$

9. Per-Plant Utility Labor in Radiation Zones for SIR Operation and Maintenance:

It was estimated that one additional man-week of NRC labor in radiation zones will be required for examination of equipment installed or exchanged as a result of SIR. Assuming a 75% utilization factor, the incremental labor in radiation zones for SIR operation and maintenance was estimated to be 53 man-hr/Ry.

10. Per-Plant Radiation Dose Increase for SIR Operation and Maintenance:

$$D_o = (53 \text{ man-hr/ry})(0.1 \text{ rem/hr}) = 5.3 \text{ person-rem/Ry}$$

11. Total Radiation Dose Increase for SIR Operation and Maintenance:

$$\begin{aligned} N\bar{D}_o &= (74 \text{ PWRs})(24.7 \text{ yr})(5.3 \text{ person-rem/Ry}) + \\ &\quad (37 \text{ BWRs})(23.3 \text{ yr})(5.3 \text{ person-rem/Ry}) \\ &= 1.4E+4 \text{ person-rem} \end{aligned}$$

12. Total Occupational Dose Increase (G):

<u>Best Estimate</u> <u>(man-rem)</u>	<u>Error Bounds (man-rem)</u>	
	<u>Upper</u>	<u>Lower</u>
1.6E+4	4.8E+4	5.3E+3

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Spurious Actuation of Instrumentation Upon Restoration of Power (160).

2. Affected Plants (N):

	<u>N</u>
PWRs	74
BWRs	37
All Plants	134

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

	<u>$\bar{T}$</u>
PWRs	24.7 yr
BWRs	23.3 yr

See Table 1, Step 2, for derivation of these values.

Industry Costs (Steps 4 through 12):

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, $\Delta(\bar{F}A)$:

$$\Delta(\bar{F}A)_{PWR} = (\$1.65E+9)(5.4E-7/RY) = \$8.9E+2/RY$$

$$\Delta(\bar{F}A)_{BWR} = (\$1.65E+9)(1.4E-8/RY) = \$2.3E+1/RY$$

5. Total Industry Cost Savings Due to Accident Avoidance (ΔH):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$ 1.6E+6	\$ 2.0E+7	0

6. Per-Plant Industry Resources for SIR Implementation:

Resource requirements to implement the proposed SIR are estimated as follows:

- Design - 0.5 person-yr/plant
- Training and installation - 2 person-yr/plant
- Additional downtime - 0
- Equipment costs - \$300,000/plant
- Test: 0.5 person-yr/plant

7. Per-Plant Industry Cost for SIR Implementation (I):

$$\begin{aligned} I &= (3 \text{ person-yr/plant})(\$100,000/\text{person-yr}) + \$300,000 \\ &= \$6.0\text{E}+5/\text{plant} \end{aligned}$$

Basic labor rates were taken from Andrews, et al., (1983).

8. Total Industry Cost for SIR Implementation (NI):

$$NI = (\$6.0\text{E}+5/\text{plant})(111 \text{ plants}) = \$6.7\text{E}+7$$

9-11 Steps Leading to Per-Plant Industry Labor and Costs for SIR Operation and Maintenance

Labor requirements to perform the enhanced training and inspection program were assumed to be integrated with or to replace existing programs. Therefore, no additional labor is assumed for SIR operation and maintenance.

12. Total Industry Cost (S_i):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$6.7E+7	\$1.0E+8	\$3.3E+7

NRC Cost (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC resources to develop the proposed resolution is estimated to be about 60 man-months of contractor and NRC staff labor to further evaluate this issue, develop potential resolutions, and perform regulatory analyses (i.e., determine the costs and benefits of the proposed resolution).

14. Total NRC Cost for SIR Development (C_o):

$$C_o = (60 \text{ man-months})(\$8330/\text{man-month}) = \$500,000$$

Basic labor rates were taken from Andrews, et al., (1983).

15. Per-Plant NRC Labor for Support of SIR Implementation:

NRC resource requirements to support SIR implementation were estimated to be 4 person-wk per plant..

16. Per-Plant NRC Cost for Support of SIR Implementation (C):

$$C = (4 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$9080/\text{plant}$$

17. Total NRC Cost for Support of SIR Implementation (NC):

$$NC = (\$9080/\text{plant})(111 \text{ plants}) = \$1.0\text{E}+6$$

18. Per-Plant NRC Labor for Review of SIR Operation and Maintenance:

NRC resource requirements for support of SIR implementation were estimated to be approximately 2 man-day/RY.

19. Per-Plant NRC Cost for Review of SIR Operation and Maintenance C_o:

$$\begin{aligned} C_o &= [(2 \text{ man-day/RY}) / (5 \text{ man-day/man-wk})] (\$2270/\text{man-wk}) \\ &= \$908/\text{RY} \end{aligned}$$

20. Total NRC Cost for Review of SIR Operation and/Maintenance (N $\bar{T}$ C_o):

$$\begin{aligned} N\bar{T}C_o &= (\$908/\text{RY})(74 \text{ PWRs})(24.7 \text{ yr}) + (\$908/\text{RY})(37 \text{ PWRs})(23.3 \text{ yr}) \\ &= \$2.4\text{E}+6 \end{aligned}$$

21. Total NRC Cost (S_N):

<u>Best Estimate</u>	<u>Upper Bound</u>	<u>Lower Bound</u>
\$4.0E+6	\$5.3E+6	\$2.6E+5

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ISSUE SUMMARY WORK SHEET

ISSUE NUMBER/TITLE: 167, Hydrogen Storage Facility Separation

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: The resolution of Generic Issue 106 excluded evaluation of the potential risk from large hydrogen storage facilities outside the reactor, auxiliary, and turbine buildings. Subsequent studies raised concern as to its magnitude; thus, Generic Issue 167 was identified. Resolution is assumed to be the construction of concrete enclosing walls around the hydrogen storage facility to serve as a blast shield in the event of an explosion.

AFFECTED PLANTS: Sixteen percent of currently operating plants are assumed to not comply with EPRI standards for separation from explosion effects, i.e., (0.16)(110 LWRs) $\approx$ 18 LWRs are affected.

RISK/DOSE RESULTS (person-rem):

PUBLIC RISK REDUCTION = 1,220

OCCUPATIONAL DOSES:

SIR Implementation = 0

SIR Operation/Maintenance = 0

Total of Above = 0

Accident Avoidance = 515

COST RESULTS (\$10⁶):

INDUSTRY COSTS:

SIR Implementation = 1.94

SIR Operation/Maintenance = 0

Total of Above = 1.94

Accident Avoidance = 42.7

NRC COSTS:

SIR Development = 0.100

SIR Implementation Support = 0.241

SIR Operation/

Maintenance Review = 0

Total of Above = 0.341

HYDROGEN STORAGE FACILITY SEPARATION

ISSUE 167

1.0 SAFETY ISSUE DESCRIPTION

In 1988, the U.S. Nuclear Regulatory Commission (NRC) completed the prioritization analysis of Generic Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas" (NRC 1988). In December 1993, the Office of Nuclear Reactor Regulation (NRR) identified a new generic issue based on exclusions from Issue 106 (NRC 1993):

The resolution of this issue [106] included evaluation of the risk from: (1) the storage and distribution of hydrogen for the volume control tank in PWRs and the main electric generator in BWRs and PWRs; (2) other sources of hydrogen such as battery rooms, the waste gas system in PWRs and the offgas system in BWRs; and (3) small, portable bottles of combustible gases used in maintenance, testing, and calibration. However, ... the potential risk from large hydrogen storage facilities outside the reactor, auxiliary, and turbine buildings was not addressed.

Studies subsequent to that for Issue 106 raised concern as to the magnitude of the previously excluded risk (Ballif, et al., 1990; EG&G Idaho, 1993). This prioritization analysis addresses this concern.

Background

As reported in SCIE-EGG-103-89, *Draft Technical Evaluation Report on U.S. Commercial Power Reactor Hydrogen Tank Farms and Their Compliance with Separation Distance Safety Criteria* (Ballif, et al., 1990),

... [A]t the Trojan Nuclear Plant, April 17, 1989, [NRC] inspectors identified a potential safety problem concerning the storage of 32,000 standard cubic feet (scf) of hydrogen gas on the control room roof. The 32,000 scf was made up of four 8,000-scf tanks. This discovery raised concerns about possible similar hazards in the storage of hydrogen at other nuclear facilities.

NRR issued Information Notice 89-44, "Hydrogen Storage on the Roof of the Control Room," on May 2, 1989. Each NRC regional office was to determine whether the nuclear plants in its region had similar safety-related concerns.

Ballif, et al., reviewed the information compiled by the five NRC regional offices and issued preliminary report SCIE-EGG-103-89. The storage of gaseous or liquid hydrogen at 119 power plants was investigated. Possible accident scenarios resulting from a fireball, explosion, or presence of unburned hydrogen gas in ventilation air intakes were examined. Explosion was identified as the scenario posing the greatest risk potential. As per Ballif, et al., this analysis focuses on explosion, with all quantification performed relative to this accident only.

Ballif, et al., assessed the status of the 119 power plants with respect to hydrogen tank farm separation compliance as per the Electric Power Research Institute (EPRI) Report NP-5283-SR-A, *Guidelines for Permanent BWR Hydrogen Water Chemistry Installations* (EPRI 1987). Sixteen percent were found to be out of compliance with respect to explosion hazard. For the current population of power plants (110), this translates into 18 LWRs out of compliance with the EPRI requirements.²

Ballif, et al., reviewed nuclear power industry experience, data published by the hydrogen industry and related industries (such as liquified natural gas), and surveyed the hydrogen supply industry to estimate the frequency of a hydrogen tank explosion. They deemed the data from the nuclear power industry to be most suitable, and estimated a frequency of 0.001/tank farm-year for a hydrogen tank farm explosion. This admittedly conservative value³ was reasonably comparable to published values of failure incidents from the liquified natural gas industry (0.0024/facility-year), which they expected to exhibit a higher rate than nuclear plant installations.

² Note that the permanently shutdown Trojan PWR is no longer included among this population. Thus, the subsequent analysis does not address hydrogen storage tanks located atop the control room roof.

³ Limited response from the survey of the hydrogen supply industry yielded a rough estimate of $1\text{E-}5$ /tank-year for a "generic hydrogen tank" rupture.

2.0. SAFETY ISSUE RISK AND DOSE

Of the types of accidents analyzed in risk assessments, hydrogen tank farm explosions would seem most similar to those classified as "external events." Furthermore, since such explosions would cause large pressure forces, or possibly missiles, to be exerted upon building walls, they would appear most similar to tornadoes among the types of external events. However, unlike what is assumed for tornadoes, hydrogen tank farm explosions would not exert pressure on all site buildings at one time. Thus, the consequences from hydrogen tank farm explosions are not expected to exceed those from tornadoes, although the risk could be greater due to higher incident frequencies (typical tornado frequencies are on the order of $1\text{E-}5/\text{reactor-year [ry]}$).

A review of recently available risk assessments yielded the Individual Plant Examination (IPE) for Oconee 3 as most amenable for the risk analysis in this prioritization analysis (Duke Power 1990). A fairly detailed assessment of the tornado risk was performed, building on that from the earlier Oconee-3 risk assessment in 1984 (NSAC and Duke Power 1984). An overview of the tornado risk assessment from the Oconee-3 IPE has been provided in Appendix 1. An added advantage to the selection of Oconee 3 is that Ballif, et al., provided a description of the Oconee plant 48,000-scf hydrogen tank farm, which they deemed as representative.

The tornado accident sequences from the Oconee-3 IPE are redefined to estimate the baseline risk from hydrogen tank farm explosions in Appendix 2. The results yield a baseline core damage frequency of $6.25\text{E-}5/\text{ry}$ and a baseline risk of 2.94 person-rem/ry. Table 1 completes the calculations. A resolution for this issue is assumed to be the construction of concrete enclosing walls around the hydrogen storage facility. This structure would serve as a blast shield in the event of an explosion, essentially eliminating the risk. Additional detail on the enclosure is provided in Section 3.0. Occupational dose calculations are summarized in Table 2.

TABLE 1. Public Risk Reduction

1. Title and Identification Number of Safety Issue:

Hydrogen Storage Facility Separation (167)

2. Affected Plants (N) and Average Remaining Lives ($\bar{T}$):

As per Ballif, et al., (1990), 16% of currently operating plants are assumed to not comply with EPRI standards for separation from explosion effects. Thus, $(0.16)(110 \text{ LWRs}) \approx 18 \text{ LWRs}$ with average remaining lives of ~23 yr are assumed affected.

3. Plants Selected For Analysis:

Oconee 3 (representative LWR)

4-9. Base-Case Parameters, Values, Frequencies, and Public Risk:

See Appendix 1. The base-case core-damage frequency and public risk have been calculated to be:

Core-Damage Frequency	=	6.25E-5/ry
Public Risk	=	2.94 person-rem/ry

10-14. Adjusted-Case Parameters, Values, Frequencies, and Public Risk:

Resolution is assumed to essentially eliminate the risk. Thus, all adjusted-case values are zero.

15. Core-Damage Frequency Reduction (ΔF):

Base-case - Adjusted-case = $6.25\text{E-}5/\text{ry} - 0 = 6.25\text{E-}5/\text{ry}$

16. Per-Plant Public Risk Reduction (ΔW):

Base-case - Adjusted-case = $2.94 \text{ person-rem/ry} - 0 = 2.94 \text{ person-rem/ry}$

17. Total Public Risk Reduction [$(\Delta W)_{\text{Total}}$]:

Best Estimate	=	$(2.94 \text{ person-rem/ry})(18 \text{ reactors})(23 \text{ yr})$
	=	$1.22\text{E+}3 \text{ person-rem}$
Lower Bound	=	0
Upper Bound	=	$3.65\text{E+}4 \text{ person-rem}$

TABLE 2. Occupational Dose

1. Title and Identification Number of Safety Issue:

Hydrogen Storage Facility Separation (167)

2. Affected Plants (N):

18 LWRs

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

~23 yr

4. Per-Plant Occupational Dose Reduction Due to Accident Avoidance [$\Delta(FD_R)$]:

$$(1.99E+4 \text{ person-rem})(6.25E-5/\text{ry}) = 1.24 \text{ person-rem/ry}$$

5. Total Occupational Dose Reduction Due to Accident Avoidance (ΔU):

Best Estimate	=	(1.24 person-rem/ry)(18 reactors)(23 yr)
	=	5.15E+2 person-rem
Lower Bound	=	0
Upper Bound	=	3.09E+3 person-rem

6-12. Utility Radiation Zone Labor and Occupational Dose Increase:

Since the enclosing structure would be constructed outside any radiation zone, no utility radiation zone labor or occupational dose increase is foreseen.

3.0. SAFETY ISSUE COSTS

Ballif, et al., (1990) reported that the 48,000-scf Ocone tank farm consists of six tanks with a footprint of 45 ft by 30 ft. It is surrounded by an exclusion fence and is always lighted. The concrete enclosure is assumed to have rectangular dimensions comparable to this footprint. The 1993 *Means Building Construction Cost Data* manual listed costs for thick, smooth gray architectural precast concrete 10-ft high and 6-in thick (R.S. Means 1993). For a 20-ft length, the cost was \$14.95/ft² (area); for a 30-ft length, the cost was \$14.60/ft². To form an enclosure at least 45 ft by 30 ft, one 20-ft and one 30-ft length would be needed for each long side and one 30-ft length for each short side. Ten feet should be sufficiently high to protect the surroundings from horizontal blast effects. However, a thickness comparable to that of site building walls (~18 in) would be desired. Thus, the total number of precast concrete wall panels becomes:

$$\begin{aligned} 30\text{-ft panels} &= (4 \text{ per perimeter})(3 \text{ at 6-in thickness each}) = 12 \\ 20\text{-ft panels} &= (2 \text{ per perimeter})(3 \text{ at 6-in thickness each}) = \frac{6}{18} \end{aligned}$$

In footprint, the enclosure would be 50 ft by 30 ft, yielding a total wall panel area of:

$$(2)(50 \text{ ft} + 30 \text{ ft})(10 \text{ ft})(3 \text{ panels}) = 4.80\text{E}+3 \text{ ft}^2.$$

At ~\$15/ft², the cost of this enclosure becomes:

$$(\$15/\text{ft}^2)(4.80\text{E}+3 \text{ ft}^2) = \$7.20\text{E}+4.$$

R.S. Means (1993) stated that, "[i]f the work is to be subcontracted, add the general contractor's markup, approximately 10%." In addition, the enclosure will have to be anchored in place and penetrated for piping and access. Combined with the general contractor's markup, these factors are assumed to increase the cost of the enclosure by ~50%, bringing the total cost to:

$$(1.5)(\$7.20\text{E}+4) = \$1.08\text{E}+5.$$

Table 3 summarizes the cost analysis, including that for NRC costs.

TABLE 3. Safety Issue Costs

1. Title and Identification Number of Safety Issue:

Hydrogen Storage Facility Separation (167)

2. Affected Plants (N):

18 LWRs

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

~23 yr

Industry Costs (Steps 4 Through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance [$\Delta(FA)$]:

$$(\$1.65E+9)(6.25E-5/ry) = \$1.03E+5/ry$$

5. Total Industry Cost Savings Due to Accident Avoidance (ΔH):

Best Estimate	=	$(\$1.03E+5/ry)(18 \text{ reactors})(23 \text{ yr}) = \$4.27E+7$
Lower Bound	=	0
Upper Bound	=	$\$2.56E+8$

6-7. Per-Plant Industry Resources and Cost (I) to Implement Resolution:

18 wall panels, twelve 30 ft x 10 ft x 6 in and six 20 ft x 10 ft x 6 in, covering a footprint 50 ft x 30 ft

$$\begin{aligned} \text{Wall Area (18-in thick)} &= \\ (2)(50 \text{ ft} + 30 \text{ ft})(10 \text{ ft})(3 \text{ panels}) &= 4.80E+3 \text{ ft}^2 \end{aligned}$$

$$\begin{aligned} \text{Per-plant cost at } \$15/\text{ft}^2 \text{ with 50\% markup} &= \\ (\$15/\text{ft}^2)(1.5)(4.80E+3 \text{ ft}^2) &= \$1.08E+5 \end{aligned}$$

8. Total Industry Cost to Implement Resolution (NI):

$$(\$1.08E+5/LWR)(18 \text{ LWRs}) = \$1.94E+6$$

9-11. Industry Labor and Cost to Operate and Maintain Resolution:

Any industry operation and maintenance activities associated with this resolution are assumed to be performed as part of activities already in place, such as standard inspection and reporting procedures. No industry labor or cost is foreseen for operation and maintenance.

12. Total Industry Cost (S_I):

Best Estimate	=	\$1.94E+6
Lower Bound	=	\$0.97E+6
Upper Bound	=	\$2.92E+6

NRC Costs (Steps 13 Through 21)

13-14. NRC Resources and Total Cost to Develop Resolution (C_D):

A level of effort analogous to that required to produce the technical reports by Ballif, et al., (1990) and EG&G Idaho (1993) is assumed necessary, with a total cost = \$1.0E+5.

15-16. Per-Plant NRC Labor and Cost (C) to Support SIR Implementation

Sciaccia (1989) listed the cost for a "typical uncomplicated technical specification change" to be \$1.1E+4 (1988 dollars). Assuming this applies to NRC support to assure utility compliance with the resolution, the following per-plant cost is estimated (with ~4%/yr inflation since 1988):

$$C = (\$1.1E+4)(1.04)^5 = \$1.34E+4$$

17. Total NRC Cost to Support SIR Implementation (NC):

$$NC = (\$1.34E+4/LWR)(18 \text{ LWRs}) = \$2.41E+5$$

18-20. NRC Labor and Cost to Review SIR Operation and Maintenance:

Any industry operation and maintenance activities associated with this resolution are assumed to be performed as part of activities already in place, such as standard inspection and reporting procedures. No industry labor or cost is foreseen for operation and maintenance. Thus, no NRC labor or cost is foreseen for review.

21. Total NRC Cost (S_N):

Best Estimate	=	\$3.41E+5
Lower Bound	=	\$2.10E+5
Upper Bound	=	\$4.71E+5

APPENDIX 1

OVERVIEW OF THE TORNADO RISK ASSESSMENT FROM THE OCONEE-3 IPE

Duke Power (1990) considered two categories of tornado events: (1) tornadoes whose winds impact on Oconee 3; and (2) tornadoes passing within 2,000 ft of Oconee 3, for the purpose of analyzing tornado-generated missiles. The latter category was subsequently dismissed when analysis showed the probability of core-melt due to tornado-generated missiles to be 100-1,000 times lower than that due to tornado wind loadings. Only the first category is addressed here.

Duke Power assumed that a tornado would render unavailable all offsite AC power sources, except for one underground path. Tornadoes of intensity F-1 or less (i.e., with wind speeds < 113 mph) were assumed not to cause sufficient wind damage to generate core-melt with a frequency similar to higher F-scale tornadoes (F-2 and above). Oconee 3 had been designed to withstand wind loadings of F-1 tornadoes.

The Turbine Building was assumed susceptible to wall damage from F-2 and stronger tornadoes. Wall damage could fail the 4160V (4KV) AC switchgear that powers safety equipment and/or the Upper Surge Tank (UST), the prime suction source for the emergency feedwater pumps. The Auxiliary Building was assumed susceptible to wall damage from F-4 and stronger tornadoes (wind speeds > 206 mph), particularly the exterior walls of the West and East Penetration Rooms (WPR and EPR). Damage to the WPR wall could fail piping and electrical penetrations, including those from the Standby Shutdown Facility (SSF). This could lead to Reactor Coolant Pump (RCP) seal LOCAs, loss of the SSF backup for RCP seal cooling, and loss of feedwater from the SSF. Damage to the EPR wall would cause similar failures, although the likelihood of piping failures there was judged to be about ten times less due to tornado shielding by the Reactor Building.⁴

Table 1-1 lists the cut set elements and their associated values for all tornado-induced accident sequences. All are initiated by tornadoes falling within categories F-2 through F-5, as indicated by the terms "T(F2)" through "T(F5)." Each leads to a Plant Damage State (PDS), designated by the term following the dash in each sequence identifier (e.g., "4F"). The frequency (per reactor-year [ry]) of each initiating event (I.E.) is listed next, followed by the conditional failure probabilities for the UST, 4KV-AC switchgear, WPR walls, and EPR walls. The column "CUT SET SUM" lists the sum of the products of the probabilities for all remaining terms (i.e., those independent of the tornado effects) in the cut sets for each sequence. The last column lists each sequence frequency (1/ry), calculated as the product of the preceding six values. At the bottom of this column is the total core-damage frequency, the sum of the accident sequence frequencies (9.74E-6/ry).

Note the following two items regarding Table 1-1. First, not all core-damage sequences result in offsite release, namely those with PDSs 4I, 7I, 8L, 9I, and 12C. Thus, the total release category frequency (calculated in Table 1-3 to be 8.20E-6/ry) is less than the total core-damage frequency. Second, one or

⁴

Other exterior components, such as the Borated Water Storage Tank, are also susceptible to failure from tornadoes. However, they did not appear in the listed cut sets for core damage, so have not been discussed here.

more of the cut set elements either do not appear or remain unaffected by tornadoes in each sequence. These are indicated by italics to distinguish them from tornado-affected elements. This distinction will be important when these cut sets are redefined in Appendix 2 to reflect sequences initiated by hydrogen storage tank explosions.

Table 1-2 lists the conditional probabilities of offsite release via the 17 possible release categories (RCs) associated with the PDSs in Table 1-1. Duke Power designated these RCs as 4.01, 4.03, etc, which has been abbreviated to "RC41," "RC43," etc. When multiplied by the sequence frequency, each of these probabilities yields the sequence frequency per RC (1/ry). These are listed in Table 1-3 and summed below each column to yield the total frequency per RC ("RC FREQ. [1/ry]").

Associated with each RC is a whole-body person-rem (P-R) equivalent dose, as follows:

RC41	=	3.54E+5
RC43	=	5.34E+5
RC44	=	1.73E+6
RC46	=	1.52E+6
RC48	=	2.80E+6
RC51	=	1.25E+7
RC52	=	1.26E+7
RC62	=	1.40E+6
RC64	=	8.73E+6
RC66	=	1.42E+6
RC68	=	8.75E+6
RC72	=	4.27E+5
RC74	=	4.27E+5
RC81	=	3.40E+4
RC91	=	2.31E+2
RC92	=	4.90E+3
RC94	=	5.05E+3

The total risk per RC is the product of the RC frequency and its associated dose. This is listed for each RC beneath the RC frequency ("RC [P-R/ry]"). Finally, all RC frequencies and all RC risks are summed to yield the total RC frequency (8.20E-6/ry) and total RC risk (9.11 person-rem/ry).

TABLE 1-1: Cut Set Elements and Frequencies for Tornado Accident Sequences

	A	B	C	D	E	F	G	H
1								
2	SEQUENCE	I.E. (1/ry)	UST	4KV	WPR	EPR	CUT SET SUM	SEQ. FR. (1/ry)
3								
4	T(F2)BQRU-4F	3.70E-05	0.5	0.5	1	1	0.01	9.25E-08
5	T(F2)QsU-7F	3.70E-05	0.5	0.5	1	1	0.1891	1.75E-06
6	T(F2)BU-12F	3.70E-05	0.5	0.5	1	1	0.0081	7.49E-08
7	T(F3)BQRU-4C	2.00E-05	0.5	0.5	1	1	0.1	5.00E-07
8	T(F3)BQRU-4F	2.00E-05	0.5	0.5	1	1	0.1	5.00E-07
9	T(F3)QsU-7F	2.00E-05	0.5	0.5	1	1	0.1891	9.46E-07
10	T(F3)BU-12C	2.00E-05	0.5	0.5	1	1	0.009	4.50E-08
11	T(F3)BU-12F	2.00E-05	0.5	0.5	1	1	0.0081	4.05E-08
12	T(F4)BQRU-4I	1.80E-05	0.5	0.5	1	1	0.1	4.50E-07
13	T(F4)BQRU-4L	1.80E-05	0.5	0.5	1	1	0.1	4.50E-07
14	T(F4)BQRU-4L	1.80E-05	1	0.5	1	0.1	0.2	1.80E-07
15	T(F4)QsU-7I	1.80E-05	0.5	0.5	1	0.9	0.181	7.33E-07
16	T(F4)QsU-7L	1.80E-05	0.5	0.5	1	0.9	0.181	7.33E-07
17	T(F4)QsX-8L	1.80E-05	0.5	0.5	1	0.9	0.0181	7.33E-08
18	T(F4)BQsU-9C	1.80E-05	0.5	0.5	1	1	0.09	4.05E-07
19	T(F4)BQsU-9L	1.80E-05	1	0.5	1	0.1	1.8	1.62E-06
20	T(F4)BQsU-9L	1.80E-05	0.5	0.5	1	1	0.09	4.05E-07
21	T(F5)BQRU-4I	2.70E-06	0.5	0.5	1	1	0.1	6.75E-08
22	T(F5)BQRU-4L	2.70E-06	0.5	0.5	1	1	0.1	6.75E-08
23	T(F5)BQRU-4L	2.70E-06	1	0.5	1	0.1	0.2	2.70E-08
24	T(F5)QsU-7I	2.70E-06	0.5	0.5	1	0.9	0.181	1.10E-07
25	T(F5)QsU-7L	2.70E-06	0.5	0.5	1	0.9	0.181	1.10E-07
26	T(F5)BQsU-9I	2.70E-06	0.5	0.5	1	1	0.09	6.08E-08
27	T(F5)BQsU-9L	2.70E-06	1	0.5	1	0.1	1.8	2.43E-07
28	T(F5)BQsU-9L	2.70E-06	0.5	0.5	1	1	0.09	6.08E-08
29								
30	SEQUENCE	I.E. (1/ry)	UST	4KV	WPR	EPR	CUT SET SUM	SEQ. FR. (1/ry)
31								
32							C.D. FR. (1/ry) -->	9.68E-06
33								
34								
35								
36								

**TABLE 1-2: Conditional Probabilities per Release Category for Tornado
Accident Sequences**

	I	J	K	L	M	N
1						
2	SEQUENCE	RC41 PROB.	RC43 PROB.	RC44 PROB.	RC46 PROB.	RC48 PROB.
3						
4	T(F2)BQRU-4F	0	0	0	0	0
5	T(F2)QsU-7F	0	0	0	0	0
6	T(F2)BU-12F	0	0	0	0	0
7	T(F3)BQRU-4C	0	0	0	0	0
8	T(F3)BQRU-4F	0	0	0	0	0
9	T(F3)QsU-7F	0	0	0	0	0
10	T(F3)BU-12C	0	0	0	0	0
11	T(F3)BU-12F	0	0	0	0	0
12	T(F4)BQRU-4I	0	0	0	0	0
13	T(F4)BQRU-4L	0	0	0	2.45E-02	9.75E-01
14	T(F4)BQRU-4L	0	0	0	2.45E-02	9.75E-01
15	T(F4)QsU-7I	0	0	0	0	0
16	T(F4)QsU-7L	4.41E-03	8.96E-01	9.95E-02	0	0
17	T(F4)QsX-8L	0	0	0	0	0
18	T(F4)BQsU-9C	0	0	0	0	0
19	T(F4)BQsU-9L	0	0	0	4.90E-02	9.50E-01
20	T(F4)BQsU-9L	0	0	0	4.90E-02	9.50E-01
21	T(F5)BQRU-4I	0	0	0	0	0
22	T(F5)BQRU-4L	0	0	0	2.45E-02	9.75E-01
23	T(F5)BQRU-4L	0	0	0	2.45E-02	9.75E-01
24	T(F5)QsU-7I	0	0	0	0	0
25	T(F5)QsU-7L	4.41E-03	8.96E-01	9.95E-02	0	0
26	T(F5)BQsU-9I	0	0	0	0	0
27	T(F5)BQsU-9L	0	0	0	4.90E-02	9.50E-01
28	T(F5)BQsU-9L	0	0	0	4.90E-02	9.50E-01
29						
30	SEQUENCE	RC41 PROB.	RC43 PROB.	RC44 PROB.	RC46 PROB.	RC48 PROB.
31						
32						
33						
34						
35						
36						

**TABLE 1-2: Conditional Probabilities per Release Category for Tornado
Accident Sequences (Continued)**

	O	P	Q	R	S	T	U
1							
2	SEQUENCE	RC51 PR.	RC52 PR.	RC62 PR.	RC64 PR.	RC66 PR.	RC68 PR.
3							
4	T(F2)BQRU-4F	2.90E-05	3.34E-04	1.24E-04	1.21E-04	9.44E-04	4.83E-04
5	T(F2)QsU-7F	5.45E-05	2.35E-05	2.47E-04	2.42E-04	9.38E-04	9.17E-04
6	T(F2)BU-12F	5.45E-04	7.31E-05	2.47E-03	2.42E-03	7.32E-04	7.17E-04
7	T(F3)BQRU-4C	3.86E-04	5.41E-04	0	0	0	0
8	T(F3)BQRU-4F	2.90E-05	3.34E-04	1.24E-04	1.21E-04	9.44E-04	4.83E-04
9	T(F3)QsU-7F	5.45E-05	2.35E-05	2.47E-04	2.42E-04	9.38E-04	9.17E-04
10	T(F3)BU-12C	0	0	0	0	0	0
11	T(F3)BU-12F	5.45E-04	7.31E-05	2.47E-03	2.42E-03	7.32E-04	7.17E-04
12	T(F4)BQRU-4I	0	0	0	0	0	0
13	T(F4)BQRU-4L	2.90E-04	3.57E-04	0	0	0	0
14	T(F4)BQRU-4L	2.90E-04	3.57E-04	0	0	0	0
15	T(F4)QsU-7I	0	0	0	0	0	0
16	T(F4)QsU-7L	5.45E-05	2.35E-05	0	1.13E-03	0	0
17	T(F4)QsX-8L	0	0	0	0	0	0
18	T(F4)BQsU-9C	6.73E-04	5.51E-04	0	0	0	0
19	T(F4)BQsU-9L	5.45E-04	7.31E-05	0	0	0	0
20	T(F4)BQsU-9L	5.45E-04	7.31E-05	0	0	0	0
21	T(F5)BQRU-4I	0	0	0	0	0	0
22	T(F5)BQRU-4L	2.90E-04	3.57E-04	0	0	0	0
23	T(F5)BQRU-4L	2.90E-04	3.57E-04	0	0	0	0
24	T(F5)QsU-7I	0	0	0	0	0	0
25	T(F5)QsU-7L	5.45E-05	2.35E-05	0	1.13E-03	0	0
26	T(F5)BQsU-9I	0	0	0	0	0	0
27	T(F5)BQsU-9L	5.45E-04	7.31E-05	0	0	0	0
28	T(F5)BQsU-9L	5.45E-04	7.31E-05	0	0	0	0
29							
30	SEQUENCE	RC51 PR.	RC52 PR.	RC62 PR.	RC64 PR.	RC66 PR.	RC68 PR.
31							
32							
33							
34							
35							
36							

**TABLE 1-2: Conditional Probabilities per Release Category for Tornado
Accident Sequences (Continued)**

	V	W	X	Y	Z	AA	AB
1							
2	SEQUENCE	RC72 PR.	RC74 PR.	RC81 PR.	RC91 PR.	RC92 PR.	RC94 PR.
3							
4	T(F2)BQrU-4F	2.20E-03	1.28E-02	2.14E-01	0	0	7.69E-01
5	T(F2)QsU-7F	4.41E-03	1.67E-02	2.47E-01	0	0	7.29E-01
6	T(F2)BU-12F	4.41E-02	1.30E-02	2.07E-01	0	0	7.29E-01
7	T(F3)BQrU-4C	0	3.51E-06	8.78E-01	1.21E-03	1.20E-01	8.68E-06
8	T(F3)BQrU-4F	2.20E-03	1.28E-02	2.14E-01	0	0	7.69E-01
9	T(F3)QsU-7F	4.41E-03	1.67E-02	2.47E-01	0	0	7.29E-01
10	T(F3)BU-12C	0	0	0	0	0	0
11	T(F3)BU-12F	4.41E-02	1.30E-02	2.07E-01	0	0	7.29E-01
12	T(F4)BQrU-4I	0	0	0	0	0	0
13	T(F4)BQrU-4L	0	0	0	0	0	0
14	T(F4)BQrU-4L	0	0	0	0	0	0
15	T(F4)QsU-7I	0	0	0	0	0	0
16	T(F4)QsU-7L	0	0	0	0	0	0
17	T(F4)QsX-8L	0	0	0	0	0	0
18	T(F4)BQsU-9C	0	3.43E-06	8.56E-01	1.43E-03	1.41E-01	8.46E-06
19	T(F4)BQsU-9L	0	0	0	0	0	0
20	T(F4)BQsU-9L	0	0	0	0	0	0
21	T(F5)BQrU-4I	0	0	0	0	0	0
22	T(F5)BQrU-4L	0	0	0	0	0	0
23	T(F5)BQrU-4L	0	0	0	0	0	0
24	T(F5)QsU-7I	0	0	0	0	0	0
25	T(F5)QsU-7L	0	0	0	0	0	0
26	T(F5)BQsU-9I	0	0	0	0	0	0
27	T(F5)BQsU-9L	0	0	0	0	0	0
28	T(F5)BQsU-9L	0	0	0	0	0	0
29							
30	SEQUENCE	RC72 PR.	RC74 PR.	RC81 PR.	RC91 PR.	RC92 PR.	RC94 PR.
31							
32							
33							
34							
35							
36							

**TABLE 1-3: Frequencies and Risks per Release Category for Tornado
Accident Sequences**

	AC	AD	AE	AF	AG	AH
1						
2	SEQUENCE	RC41 (1/ry)	RC43 (1/ry)	RC44 (1/ry)	RC46 (1/ry)	RC48 (1/ry)
3						
4	T(F2)BQRU-4F	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
5	T(F2)QsU-7F	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
6	T(F2)BU-12F	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7	T(F3)BQRU-4C	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8	T(F3)BQRU-4F	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
9	T(F3)QsU-7F	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
10	T(F3)BU-12C	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
11	T(F3)BU-12F	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
12	T(F4)BQRU-4I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
13	T(F4)BQRU-4L	0.00E+00	0.00E+00	0.00E+00	1.10E-08	4.39E-07
14	T(F4)BQRU-4L	0.00E+00	0.00E+00	0.00E+00	4.41E-09	1.76E-07
15	T(F4)QsU-7I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
16	T(F4)QsU-7L	3.23E-09	6.57E-07	7.29E-08	0.00E+00	0.00E+00
17	T(F4)QsX-8L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	T(F4)BQsU-9C	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
19	T(F4)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	7.94E-08	1.54E-06
20	T(F4)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	1.98E-08	3.85E-07
21	T(F5)BQRU-4I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
22	T(F5)BQRU-4L	0.00E+00	0.00E+00	0.00E+00	1.65E-09	6.58E-08
23	T(F5)BQRU-4L	0.00E+00	0.00E+00	0.00E+00	6.62E-10	2.63E-08
24	T(F5)QsU-7I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
25	T(F5)QsU-7L	4.85E-10	9.85E-08	1.09E-08	0.00E+00	0.00E+00
26	T(F5)BQsU-9I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
27	T(F5)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	1.19E-08	2.31E-07
28	T(F5)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	2.98E-09	5.77E-08
29						
30	SEQUENCE	RC41 (1/ry)	RC43 (1/ry)	RC44 (1/ry)	RC46 (1/ry)	RC48 (1/ry)
31						
32	RC FREQ. (1/ry) -->	3.72E-09	7.55E-07	8.39E-08	1.29E-07	2.86E-06
33	RC (pers-rem/ry) -->	1.32E-03	4.03E-01	1.45E-01	1.96E-01	8.01E+00
34						
35						
36						

**TABLE 1-3: Frequencies and Risks per Release Category for Tornado
Accident Sequences (Continued)**

	AI	AJ	AK	AL	AM	AN	AO
1							
2	SEQUENCE	RC51 (1/ry)	RC52 (1/ry)	RC62 (1/ry)	RC64 (1/ry)	RC66 (1/ry)	RC68 (1/ry)
3							
4	T(F2)BQRU-4F	2.68E-12	3.09E-11	1.15E-11	1.12E-11	8.73E-11	4.47E-11
5	T(F2)QsU-7F	9.53E-11	4.11E-11	4.32E-10	4.23E-10	1.64E-09	1.60E-09
6	T(F2)BU-12F	4.08E-11	5.48E-12	1.85E-10	1.81E-10	5.48E-11	5.37E-11
7	T(F3)BQRU-4C	1.93E-10	2.71E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8	T(F3)BQRU-4F	1.45E-11	1.67E-10	6.20E-11	6.05E-11	4.72E-10	2.42E-10
9	T(F3)QsU-7F	5.15E-11	2.22E-11	2.34E-10	2.29E-10	8.87E-10	8.67E-10
10	T(F3)BU-12C	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
11	T(F3)BU-12F	2.21E-11	2.96E-12	1.00E-10	9.80E-11	2.96E-11	2.90E-11
12	T(F4)BQRU-4I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
13	T(F4)BQRU-4L	1.31E-10	1.61E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
14	T(F4)BQRU-4L	5.22E-11	6.43E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00
15	T(F4)QsU-7I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
16	T(F4)QsU-7L	4.00E-11	1.72E-11	0.00E+00	8.28E-10	0.00E+00	0.00E+00
17	T(F4)QsX-8L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	T(F4)BQsU-9C	2.73E-10	2.23E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
19	T(F4)BQsU-9L	8.83E-10	1.18E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
20	T(F4)BQsU-9L	2.21E-10	2.96E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00
21	T(F5)BQRU-4I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
22	T(F5)BQRU-4L	1.96E-11	2.41E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00
23	T(F5)BQRU-4L	7.83E-12	9.64E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
24	T(F5)QsU-7I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
25	T(F5)QsU-7L	5.99E-12	2.58E-12	0.00E+00	1.24E-10	0.00E+00	0.00E+00
26	T(F5)BQsU-9I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
27	T(F5)BQsU-9L	1.32E-10	1.78E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00
28	T(F5)BQsU-9L	3.31E-11	4.44E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
29							
30	SEQUENCE	RC51 (1/ry)	RC52 (1/ry)	RC62 (1/ry)	RC64 (1/ry)	RC66 (1/ry)	RC68 (1/ry)
31							
32	RC (1/ry) -->	2.18E-09	1.21E-09	1.02E-09	1.96E-09	3.17E-09	2.84E-09
33	RC (P-R/ry) -->	2.73E-02	1.52E-02	1.43E-03	1.71E-02	4.50E-03	2.48E-02
34							
35							
36							

**TABLE 1-3: Frequencies and Risks per Release Category for Tornado
Accident Sequences (Continued)**

	AP	AQ	AR	AS	AT	AU	AV
1							
2	SEQUENCE	RC72 (1/ry)	RC74 (1/ry)	RC81 (1/ry)	RC91 (1/ry)	RC92 (1/ry)	RC94 (1/ry)
3							
4	T(F2)BQRU-4F	2.04E-10	1.18E-09	1.98E-08	0.00E+00	0.00E+00	7.11E-08
5	T(F2)QsU-7F	7.71E-09	2.92E-08	4.32E-07	0.00E+00	0.00E+00	1.28E-06
6	T(F2)BU-12F	3.30E-09	9.74E-10	1.55E-08	0.00E+00	0.00E+00	5.46E-08
7	T(F3)BQRU-4C	0.00E+00	1.76E-12	4.39E-07	6.05E-10	6.00E-08	4.34E-12
8	T(F3)BQRU-4F	1.10E-09	6.40E-09	1.07E-07	0.00E+00	0.00E+00	3.85E-07
9	T(F3)QsU-7F	4.17E-09	1.58E-08	2.34E-07	0.00E+00	0.00E+00	6.89E-07
10	T(F3)BU-12C	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
11	T(F3)BU-12F	1.79E-09	5.27E-10	8.38E-09	0.00E+00	0.00E+00	2.95E-08
12	T(F4)BQRU-4I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
13	T(F4)BQRU-4L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
14	T(F4)BQRU-4L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
15	T(F4)QsU-7I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
16	T(F4)QsU-7L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
17	T(F4)QsX-8L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	T(F4)BQsU-9C	0.00E+00	1.39E-12	3.47E-07	5.79E-10	5.71E-08	3.43E-12
19	T(F4)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
20	T(F4)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
21	T(F5)BQRU-4I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
22	T(F5)BQRU-4L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
23	T(F5)BQRU-4L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
24	T(F5)QsU-7I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
25	T(F5)QsU-7L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
26	T(F5)BQsU-9I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
27	T(F5)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
28	T(F5)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
29							
30	SEQUENCE	RC72 (1/ry)	RC74 (1/ry)	RC81 (1/ry)	RC91 (1/ry)	RC92 (1/ry)	RC94 (1/ry)
31							
32	RC (1/ry) -->	1.83E-08	5.41E-08	1.60E-06	1.18E-09	1.17E-07	2.50E-06
33	RC (P-R/ry) -->	7.80E-03	2.31E-02	5.45E-02	2.74E-07	5.74E-04	1.26E-02
34							
35						TOT(/ry) -->	8.14E-06
36						T(PR/ry) -->	8.95E+00

APPENDIX 2

RISK ANALYSIS FOR HYDROGEN STORAGE TANK EXPLOSIONS

The best available data from Ballif, et al., (1990) indicated a frequency of 0.001/tank farm-year for a hydrogen storage tank explosion. They further stated:

Some reactors may have multiple tank farms, and in other cases multiple reactors may share a single tank farm. It is assumed these cases have a general cancelling effect and there is an average of one tank farm per reactor. This assumption is believed to be reasonable ...

Therefore, the frequency of a hydrogen storage tank explosion is assumed to be:

$$(0.001/\text{tank farm-year})(1 \text{ tank farm/reactor}) = 0.001/\text{ry.}$$

To translate this into an initiator frequency analogous to that of a tornado, it is necessary to associate the blast force from an explosion to the severity of a tornado. This information is not readily available, although Ballif, et al., reported that EPRI assumed the blast pressure to range from ~1.5 to ~4.5 psi, depending on the material and its equivalent mass of TNT explosive (EPRI 1987). This analysis assumes that the blast force is most likely to correspond to that from an F-2 tornado. An initiator frequency of 0.001/ry is assigned to T(F2) to reflect this, decreasing the frequency by a factor of 10 for each increasing category of tornado severity (assumed to correspond to increasing blast forces), i.e., 1E-4/ry for T(F3), 1E-5/ry for T(F4), and 1E-6/ry for T(F5). These are listed in Table 2-1.

Unlike tornadoes, hydrogen tank farm explosions would not exert pressure on all site buildings at one time. Thus, multiple building wall failures are not expected as in the tornado accident sequences. To reflect this limitation, the cut sets of the tornado accident sequences have been reviewed and found to nearly all contain conditional failure of the Turbine Building wall. Associated with this are failures of the UST and/or 4KV-AC switchgear. Failures of the walls of the WPR and/or EPR are contained in less of the cut sets of the tornado accident sequences.

A hydrogen tank farm explosion is assumed to fail only the Turbine Building wall, resulting in failures of the UST and/or 4KV-AC switchgear. No failure of the WPR or EPR walls is assumed. This is reflected in Table 2-1 by reassigning all non-italicized conditional probabilities of the WPR or EPR walls to be zero (see Appendix 1 and Table 1-1). This eliminates most of the T(F4) and T(F5) sequences. However, the higher initiator frequencies for T(F2) and T(F3) result in a total core-damage frequency from hydrogen tank explosion of 6.25E-5/ry, ~64% higher than Duke Power's assessment for tornadoes.

Using the same RC conditional probabilities and equivalent doses as for tornadoes (see Table 1-2 and Appendix 1), a total RC frequency of 6.20E-5/ry (again, less than the total core-damage frequency since not all accident sequences lead to offsite release) and a total RC risk of 2.94 person-rem/ry are

obtained (see Table 2-2).⁵ Observe that, despite the higher core-damage and RC frequencies, the risk is less than that for tornadoes since the RCs associated with the PDSs of the T(F2) and T(F3) sequences are less consequential than those associated with T(F4) and T(F5).

⁵ Table 2-2 is analogous to Table 1-3, rather than Table 1-2. Table 1-2 was not reproduced as Table 2-2 since the RC conditional probabilities remained the same.

**TABLE 2-1: Cut Set Elements and Frequencies for Hydrogen Storage Tank
Explosion Accident Sequences**

	A	B	C	D	E	F	G	H
1								
2	SEQUENCE	I.E. (1/ry)	UST	4KV	WPR	EPR	<i>CUT SET SUM</i>	SEQ. FR. (1/ry)
3								
4	T(F2)BQRU-4F	1.00E-03	0.5	0.5	1	1	0.01	2.50E-06
5	T(F2)QsU-7F	1.00E-03	0.5	0.5	1	1	0.1891	4.73E-05
6	T(F2)BU-12F	1.00E-03	0.5	0.5	1	1	0.0081	2.03E-06
7	T(F3)BQRU-4C	1.00E-04	0.5	0.5	1	1	0.1	2.50E-06
8	T(F3)BQRU-4F	1.00E-04	0.5	0.5	1	1	0.1	2.50E-06
9	T(F3)QsU-7F	1.00E-04	0.5	0.5	1	1	0.1891	4.73E-06
10	T(F3)BU-12C	1.00E-04	0.5	0.5	1	1	0.009	2.25E-07
11	T(F3)BU-12F	1.00E-04	0.5	0.5	1	1	0.0081	2.03E-07
12	T(F4)BQRU-4I	1.00E-05	0.5	0.5	1	1	0.1	2.50E-07
13	T(F4)BQRU-4L	1.00E-05	0.5	0.5	1	1	0.1	2.50E-07
14	T(F4)BQRU-4L	1.00E-05	1	0.5	0	0	0.2	0.00E+00
15	T(F4)QsU-7I	1.00E-05	0.5	0.5	0	0	0.181	0.00E+00
16	T(F4)QsU-7L	1.00E-05	0.5	0.5	0	0	0.181	0.00E+00
17	T(F4)QsX-8L	1.00E-05	0.5	0.5	0	0	0.0181	0.00E+00
18	T(F4)BQsU-9C	1.00E-05	0.5	0.5	0	1	0.09	0.00E+00
19	T(F4)BQsU-9L	1.00E-05	1	0.5	0	0	1.8	0.00E+00
20	T(F4)BQsU-9L	1.00E-05	0.5	0.5	0	1	0.09	0.00E+00
21	T(F5)BQRU-4I	1.00E-06	0.5	0.5	1	1	0.1	2.50E-08
22	T(F5)BQRU-4L	1.00E-06	0.5	0.5	1	1	0.1	2.50E-08
23	T(F5)BQRU-4L	1.00E-06	1	0.5	0	0	0.2	0.00E+00
24	T(F5)QsU-7I	1.00E-06	0.5	0.5	0	0	0.181	0.00E+00
25	T(F5)QsU-7L	1.00E-06	0.5	0.5	0	0	0.181	0.00E+00
26	T(F5)BQsU-9I	1.00E-06	0.5	0.5	0	1	0.09	0.00E+00
27	T(F5)BQsU-9L	1.00E-06	1	0.5	0	0	1.8	0.00E+00
28	T(F5)BQsU-9L	1.00E-06	0.5	0.5	0	1	0.09	0.00E+00
29								
30	SEQUENCE	I.E. (1/ry)	UST	4KV	WPR	EPR	<i>CUT SET SUM</i>	SEQ. FR. (1/ry)
31								
32							C.D. FR. (1/ry) -->	6.25E-05
33								
34								
35								
36								

**TABLE 2-2: Frequencies and Risks per Release Category for Hydrogen Storage
Tank Explosion Accident Sequences**

	AC	AD	AE	AF	AG	AH
1						
2	SEQUENCE	RC41 (1/ry)	RC43 (1/ry)	RC44 (1/ry)	RC46 (1/ry)	RC48 (1/ry)
3						
4	T(F2)BQrU-4F	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
5	T(F2)QsU-7F	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
6	T(F2)BU-12F	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7	T(F3)BQrU-4C	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8	T(F3)BQrU-4F	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
9	T(F3)QsU-7F	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
10	T(F3)BU-12C	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
11	T(F3)BU-12F	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
12	T(F4)BQrU-4I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
13	T(F4)BQrU-4L	0.00E+00	0.00E+00	0.00E+00	6.13E-09	2.44E-07
14	T(F4)BQrU-4L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
15	T(F4)QsU-7I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
16	T(F4)QsU-7L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
17	T(F4)QsX-8L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	T(F4)BQsU-9C	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
19	T(F4)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
20	T(F4)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
21	T(F5)BQrU-4I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
22	T(F5)BQrU-4L	0.00E+00	0.00E+00	0.00E+00	6.13E-10	2.44E-08
23	T(F5)BQrU-4L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
24	T(F5)QsU-7I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
25	T(F5)QsU-7L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
26	T(F5)BQsU-9I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
27	T(F5)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
28	T(F5)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
29						
30	SEQUENCE	RC41 (1/ry)	RC43 (1/ry)	RC44 (1/ry)	RC46 (1/ry)	RC48 (1/ry)
31						
32	RC FREQ. (1/ry) -->	0.00E+00	0.00E+00	0.00E+00	6.74E-09	2.68E-07
33	RC (pers-rem/ry) -->	0.00E+00	0.00E+00	0.00E+00	1.02E-02	7.51E-01
34						
35						
36						

**TABLE 2-2: Frequencies and Risks per Release Category for Hydrogen Storage
Tank Explosion Accident Sequences (Continued)**

	AI	AJ	AK	AL	AM	AN	AO
1							
2	SEQUENCE	RC51 (1/ry)	RC52 (1/ry)	RC62 (1/ry)	RC64 (1/ry)	RC66 (1/ry)	RC68 (1/ry)
3							
4	T(F2)BQrU-4F	7.25E-11	8.35E-10	3.10E-10	3.03E-10	2.36E-09	1.21E-09
5	T(F2)QsU-7F	2.58E-09	1.11E-09	1.17E-08	1.14E-08	4.43E-08	4.34E-08
6	T(F2)BU-12F	1.10E-09	1.48E-10	5.00E-09	4.90E-09	1.48E-09	1.45E-09
7	T(F3)BQrU-4C	9.65E-10	1.35E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8	T(F3)BQrU-4F	7.25E-11	8.35E-10	3.10E-10	3.03E-10	2.36E-09	1.21E-09
9	T(F3)QsU-7F	2.58E-10	1.11E-10	1.17E-09	1.14E-09	4.43E-09	4.34E-09
10	T(F3)BU-12C	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
11	T(F3)BU-12F	1.10E-10	1.48E-11	5.00E-10	4.90E-10	1.48E-10	1.45E-10
12	T(F4)BQrU-4I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
13	T(F4)BQrU-4L	7.25E-11	8.93E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00
14	T(F4)BQrU-4L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
15	T(F4)QsU-7I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
16	T(F4)QsU-7L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
17	T(F4)QsX-8L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	T(F4)BQsU-9C	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
19	T(F4)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
20	T(F4)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
21	T(F5)BQrU-4I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
22	T(F5)BQrU-4L	7.25E-12	8.93E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
23	T(F5)BQrU-4L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
24	T(F5)QsU-7I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
25	T(F5)QsU-7L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
26	T(F5)BQsU-9I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
27	T(F5)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
28	T(F5)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
29							
30	SEQUENCE	RC51 (1/ry)	RC52 (1/ry)	RC62 (1/ry)	RC64 (1/ry)	RC66 (1/ry)	RC68 (1/ry)
31							
32	RC (1/ry) -->	5.24E-09	4.51E-09	1.90E-08	1.86E-08	5.51E-08	5.17E-08
33	RC (P-R/ry) -->	6.55E-02	5.68E-02	2.66E-02	1.62E-01	7.83E-02	4.52E-01
34							
35							
36							

**TABLE 2-2: Frequencies and Risks per Release Category for Hydrogen Storage
Tank Explosion Accident Sequences (Continued)**

	AP	AQ	AR	AS	AT	AU	AV
1							
2	SEQUENCE	RC72 (1/ry)	RC74 (1/ry)	RC81 (1/ry)	RC91 (1/ry)	RC92 (1/ry)	RC94 (1/ry)
3							
4	T(F2)BQRU-4F	5.50E-09	3.20E-08	5.35E-07	0.00E+00	0.00E+00	1.92E-06
5	T(F2)QsU-7F	2.08E-07	7.89E-07	1.17E-05	0.00E+00	0.00E+00	3.45E-05
6	T(F2)BU-12F	8.93E-08	2.63E-08	4.19E-07	0.00E+00	0.00E+00	1.48E-06
7	T(F3)BQRU-4C	0.00E+00	8.78E-12	2.20E-06	3.03E-09	3.00E-07	2.17E-11
8	T(F3)BQRU-4F	5.50E-09	3.20E-08	5.35E-07	0.00E+00	0.00E+00	1.92E-06
9	T(F3)QsU-7F	2.08E-08	7.89E-08	1.17E-06	0.00E+00	0.00E+00	3.45E-06
10	T(F3)BU-12C	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
11	T(F3)BU-12F	8.93E-09	2.63E-09	4.19E-08	0.00E+00	0.00E+00	1.48E-07
12	T(F4)BQRU-4I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
13	T(F4)BQRU-4L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
14	T(F4)BQRU-4L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
15	T(F4)QsU-7I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
16	T(F4)QsU-7L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
17	T(F4)QsX-8L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	T(F4)BQsU-9C	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
19	T(F4)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
20	T(F4)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
21	T(F5)BQRU-4I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
22	T(F5)BQRU-4L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
23	T(F5)BQRU-4L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
24	T(F5)QsU-7I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
25	T(F5)QsU-7L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
26	T(F5)BQsU-9I	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
27	T(F5)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
28	T(F5)BQsU-9L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
29							
30	SEQUENCE	RC72 (1/ry)	RC74 (1/ry)	RC81 (1/ry)	RC91 (1/ry)	RC92 (1/ry)	RC94 (1/ry)
31							
32	RC (1/ry) -->	3.39E-07	9.61E-07	1.66E-05	3.03E-09	3.00E-07	4.34E-05
33	RC (P-R/ry) -->	1.45E-01	4.11E-01	5.63E-01	6.99E-07	1.47E-03	2.19E-01
34							
35						TOT(/ry) -->	6.20E-05
36						T(PR/ry) -->	2.94E+00

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10. SUPPLEMENTARY NOTES

R. Emrit, NRC Project Manager

11. ABSTRACT (200 words or less)

This is the sixth in a series of reports to document the use of a methodology developed by Pacific Northwest Laboratories to calculate, for prioritization purposes, the risk, dose, and cost impacts of implementing resolutions to reactor safety issues (NUREG/CR-2800, Andrews, et al., 1983). This report contains the results of issue-specific analyses for 34 generic issues. The results are referenced, as one consideration in setting priorities for reactor safety issues, in NUREG-0933, A Prioritization of Generic Safety Issues.

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