PNL-4297 Supp 4

NUREG/CR-2800
PNL-4297
Supplement 4

Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development

Prepared by A. S. Tabatabai, B. A. Fecht, T. B. Powers, W. E. Bickford,
W. B. Andrews, R. H. V. Gallucci, S. H. Bian, P. M. Daling,
E. J. Eschbach, C. H. Allen, G. J. Konzek, M. A. McLean, A. M. Schreiber

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory Commission

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Prepared by
A. S. Tabatabai, B. A. Fecht, T. B. Powers, W. E. Bickford,
W. B. Andrews, R. H. V. Gallucci, S. H. Bian, P. M. Daling,
E. J. Eschbach, C. H. Allen, G. J. Konzek, M. A. McLean, A. M. Schreiber

Pacific Northwest Laboratory
Richland, WA 99352

Prepared for
Division of Safety Review and Oversight
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
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ABSTRACT

This is the fifth in a series of reports to document the use of a methodology developed by the Pacific Northwest Laboratory to calculate, for prioritization purposes, the risk, dose and cost impacts of implementing resolutions to reactor safety issues (NUREG/CR-2800, Andrews et al. 1983). This report contains results of issue-specific analyses for 23 issues. Each issue was considered within the constraints of available information as of winter 1986, and two staff-weeks of labor. The results are referenced, as one consideration in setting priorities for reactor safety issues, in NUREG-0933, A Prioritization of Generic Safety Issues.

PREFACE

This report was prepared by the Pacific Northwest Laboratory (PNL) to communicate results of the Prioritization of Safety Issues (PSI) Project. An objective of the project is to develop a methodology to quantify risk, dose and cost impacts of 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 fifth 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 Prioritizataion of the U.S. Nuclear Regulatory Commission Human Factors Program, summarizes the results of risk and cost analyses conducted by the Pacific Northwest Laboratory in support of efforts for the Human Factors Program Plan.

The following listing identifies issues that were documented in the initial NUREG/CR-2800 report and in the three 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

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B-6	Loads, Load Combinations, Stress Limits
B-10	Behavior of BWR Mark III Containments
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I.C.1(4)	Confirmatory Analysis of Selected Transients
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II.C.2	Continuation of Interim Reliability Evaluation Program
II.C.3	Systems Interatction
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 or Operator Fitness
I.A.3.4	Licensing of Additional Operations Personnel

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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
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II.D.2	Research on Relief and Safety Valve Test Requirements
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II.E.6	In-Site Testing of Valves
II.E.6	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

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
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C. H. Allen	53
W. B. Andrews	59
W. E. Bickford	59, 70, 74 Review Team
S. H. Bian	B.59, B.65
P. M. Daling	60
E. J. Eschbach	49
B. A. Fecht	61, 63, 69, 71, 105, B.22, B-50, B.59 Review Team
R. H. V. Gallucci	51, 57, 58
G. J. Konzek	57
M. A. McLean	85
T. B. Powers	43 Review Team
A. M. Schreiber	45
A. S. Tabatabai	94, 99 Review Team
J. W. Thielman	Editor
R. Gibby	Word Processor
M. R. Guttenberg	Word Processor
K. J. Morgan	Word Processor
M. H. White	Word Processor

A. S. Tabatabai
Project Manager

CONTENTS

CONTENTS (cont'd.)

1.0 INTRODUCTION

This report documents the use of a methodology developed by the Pacific Northwest Laboratory[a] to provide the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Reactor Regulation (NRR) with information to use in prioritizing 23 safety issues related to nuclear power plants. Estimates in this report, along with other subjective factors, were used by the NRC to rank safety issues for further investigation or possible implementation. The safety issue ranking decisions made by NRC are documented in NUREG-0933 (NRC 1986).

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.

(a) Operated by Battelle Memorial Institute.

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

1.1 PUBLIC RISK REDUCTION

The public risk reduction term is defined as the product of the number of plants affected by the 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:

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

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

where $\underline{N}$ = number of reactors affected by the SIR
$\bar{T}$ = average remaining operating life of reactors affected (years)
$\Delta W = \Delta(FR)$ = change, due to the SIR, in the product of estimated time frequency of accidents in $(\text{reactor-years})^{-1}$ and public consequences per accident in man-rem for an average plant.

1.2 OCCUPATIONAL DOSE

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

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

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

where $\underline{N}$ = number of reactors affected by the SIR
$\bar{T}$ = average remaining operating life of reactors affected (years)
D_o = annual incremental dose increase due to O/M of the SIR (man-rem/reactor-year)
D = incremental dose increase due to implementation of the SIR (man-rem/reactor).

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

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

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

where N = number of reactors affected by the SIR
$\bar{T}$ = average remaining operating life of reactors affected (years)
$\Delta(FD_R)$ = change, due to the SIR, in the product of estimated time frequency of accidents in (reactor-years)$^{-1}$ and occupational dose due to cleanup and repair of the reactor following an accident (man-rem).

1.3 COSTS

Costs incurred for implementing the SIR include:

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

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

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

1.3.1 NRC Costs

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

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

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

where N = number of plants affected by the SIR

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

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

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

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

1.3.2 Industry Costs

Industry costs are defined as follows:

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

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

where N = number of reactors affected

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

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

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

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

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

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

where N = number of reactors affected

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

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

REFERENCES (for SECTION 1.0 INTRODUCTION)

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-1800, PNL-4297, Pacific Northwest Laboratory, Richland, Washington.

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

2.0 ISSUE ANALYSES

Twenty-three 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: 43, Contamination of Instrument Air Lines

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

The principal concern of this issue is that material contamination of instrument air lines can result and has resulted in reactor transients and unit scrams. The proposed resolution is to issue an IE circular that suggests corrective actions where necessary.

AFFECTED PLANTS			
	BWR:	Operating = 24	Planned = 0
	PWR:	Operating = 47	Planned = 0

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 30

OCCUPATIONAL DOSES:

SIR Implementation =	0
SIR Operation/Maintenance =	200
Total of Above =	200
Accident Avoidance =	1.8E-01

COST RESULTS ($$10^6$)

INDUSTRY COSTS:

SIR Implementation =	16
SIR Operation/Maintenance =	6.1
Total of Above =	22
Accident Avoidance =	1.5E-02

NRC COSTS:

SIR Development =	0
SIR Implementation Support =	0.97
SIR Operation/Maintenance Review =	0.25
Total of Above =	1.2

CONTAMINATION OF INSTRUMENT AIR LINES

ISSUE 43

1.0 SAFETY ISSUE DESCRIPTION

The principal concern of this issue is that material contamination of instrument air lines can result and has resulted in reactor transients and unit scrams. Because of the nonsafety classification for the instrument air system (IAS), only limited attention has been given to this subject in the past. More recently, both the Nuclear Regulatory Commission (NRC) and the Advisory Committee on Reactor Safeguards (ACRS) have directed that attention be given to this subject and to the role these systems play in causing reactor transients. One ACRS recommendation involved a systematic reevaluation of the common-cause failure potential of compressed-air systems used for control or service in both safety and nonsafety applications. Among the matters to be considered in such a review should be the effect of moisture and corrosion products and a total loss of air supply. Also of concern is any interconnection of compressed-air supplies to both safety and nonsafety devices and to other fluid systems. Consideration should be given to the adequacy of separation rules for air systems.

Compressed air may be contaminated from several sources, including 1) the ambient air, 2) the compressor itself, 3) drying equipment, and 4) corrosion products in the piping systems. Thus, compressed air must be cleaned and dried for many applications. Service air for many applications such as tools, cylinders, brakes, and various machinery can carry dirt, water, and sludge into the equipment, causing corrosion and impeding free movement of moving parts. Instrument air must be of a higher grade to prevent clogging and corrosion inside tubing, instruments, and valves.

The trend is to use higher-quality air, with many decisions on auxiliary equipment being made on the basis of preventing trouble. The potential for damage and loss is real, and the cost of the auxiliary equipment is low in comparison with compressor capacity cost or reactor unit downtime. The potential for cost reductions in maintenance due to use of higher quality air is recognized. However, this potential is not estimated in this issue analysis because its realization is uncertain.

A compressed-air system is provided for normal nuclear steam supply instrumentation and valve operators, both of which are required for plant control. The objective of the compressed-air system is to ensure the availability of required air of suitable quality and pressure for instruments, controls, maintenance, and general power plant uses and operations.

The compressed-air system is generally divided into two subsystems, the service air system (SAS) and the instrument air system (IAS). The compressed-gas system (air and nitrogen) is not classified as safety grade except for those portions of the distribution system that penetrate the containment. In some cases, a separate and independent system called the containment instrument air system (CIAS) is located entirely within the containment structure to preclude any pressurization of the containment structure.

The SAS is designed to back up the IAS during abnormal unit operations. The IAS is designed so that the instrument air shall be available under all normal and abnormal operating conditions. All essential systems requiring air during or after an accident are self-supporting, and after an accident the air system is reestablished.

Operation of the IAS is not required to initiate operation of engineered safeguards equipment. However, scenarios can be developed where, after the storage accumulators are exhausted, failure of the IAS can be shown to influence performance of equipment in other service groups which, after their subsequent failure, can then adversely affect the performance of yet other equipment in engineered safeguards systems. The probability of such a common-cause failure happening is very low.

All pneumatically operated valves are designed to assume their safety-related positions upon loss of a supply of compressed air. Even so, in the event of loss of normal power, individual air accumulators serve as a "reliable" source of compressed air for the main steam isolation valves, main steam relief valve, feedwater control valves, and containment air locks. If a compressed-air system fails, accumulator air is trapped by a check valve. Should an accumulator failure occur, the associated control valves will assume their safety-related positions.

The availability of the IAS distribution system is improved by the use of the SAS distribution system as a backup supply. When IAS header pressure is low, the SAS is manually diverted by remote control to the IAS distribution system. This is a contingency situation: Even though it is recognized that "dirty" air is contaminating the system, the immediate need for the supply of air outweighs the prior consideration that that supply be clean.

PROPOSED RESOLUTION

For purposes of this analysis, this proposed safety issue resolution (SIR) is assumed to involve the following actions for holders of operating licences and for holders of construction permits:[a]

Actions for Holders of Operating Licenses

1. Examine the existing design and establish the significance of the safety role played by the compressed-air systems. The examination should identify 1) the safety related components and systems that require air to operate, and 2) the non-safety-related air-operated components and systems which may fail in such a manner, as a result of a malfunction to the air supply systems, that will cause a challenge to the plant protection systems.

(a) From a proposed IE circular entitled "Contamination of Air Serving Safety Related Equipment" written by Jose A. Calvo.

2. Determine whether redundant safety-related components are being supplied or are capable of being supplied from a common compressed-air source. If this is the case, evaluate the safety consequences of a single failure resulting from air contamination that leads to a common-cause failure of sufficient redundant components that will cause the loss of the associated systems safety function when required. If the consequences are found unacceptable, corrective actions should be taken to preclude such a contamination of air from resulting in loss of the safety function.

3. If the recommended action No. 1 above has identified safety-related air-operated components or nonsafety air-operated components which upon their failure challenge the plant protection systems, the following recommended action will then apply:

   To assure that the air supply equipment (compressors and associated controls and backup air supplies) and the equipment provided to maintain the quality of air supplied (filters and dryers) are functioning within the design requirements, the air quality should be tested at suitable locations, every refueling outage or every 18 months, whichever is less. The air quality test results should be consistent with satisfying the requirements set forth in ANSI MC11.1-1976 (ISA-S7.3) - "Quality Standard for Instrument Air," or the test results justified on some other defined basis.

4. If the test results indicate that the required air quality has not been met or that it has been determined by other means such as surveillance tests of air-operated safety-related equipment that the air supply has been contaminated, appropriate corrective actions should be taken. These should include blowdown, cleanup and test of the entire air system and the safety and non-safety components (identified in action No. 1 above) which could have been affected by the contamination of the air.

Actions for Holders of Construction Permits

Evaluate the design of compressed-air systems serving safety-related equipment and consider design changes to preclude an air contamination event from leading to a common-cause failure of redundant safety-related equipment.

AFFECTED PLANTS

The resolution of this safety issue is assumed to affect all operating 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 and summarized in Tables 1 and 2, respectively. The analysis is conducted for a representative LWR, rather than for a representative PWR and BWR, a consequence of the data base employed for this issue analysis (see Attachment 1 following Table 1).

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Contamination of Instrument Air Lines (43)

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

	N	$\bar{T}$
Backfit LWRs	71	26.9 yr

3. Plants Selected for Analysis:

   A hypothetical LWR is assumed to be representative of all LWRs.[a]

4-7. Steps Related to Affected Parameters, Accident Sequences, and Release Categories and Their Base-Case Values:

   The base-case, affected core-melt frequency is estimated directly in the next step. Thus, these steps are omitted.

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

   $\bar{F}$ = 9.4E-09/py[a]

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

   W = .031 man-rem/py[a]

10. Steps Related to Affected Parameters, Accident Sequences, and Release Categories and Their Base-Case

   The adjusted-case, affected core-melt frequency is estimated directly in the next step. Thus, these steps are omitted.

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

   $\bar{F}$* = 4.7E-09/py[a]

(a) See Attachment 1.

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

W^* = .0155 man-rem/py[a]

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

$\Delta\bar{F}$ = 4.7E-09/py

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

ΔW = .0155 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
30	1780	0

(a) See Attachment 1.

ATTACHMENT 1 (To Table 1)

The Oak Ridge precursor study (Minarick and Kukielka 1982) is used to calculate the base-case, affected core-melt frequency for a hypothetical LWR. The Oak Ridge precursor study is based on analyses of Licensee Event Reports (LERs) from 1969-1979. The calculation of the base-case, affected core-melt frequency uses LER data, which are assumed to be representative of all LWRs.

Three LERs were considered in the Oak Ridge precursor study that involved contaminated instrument air lines. Two of the three LERs involved the freezing of moisture contamination in air lines. These two LERs are considered to be assessed in Safety Issue 45, "Inoperability of Instrumentation Due to Extreme Cold Weather," and are therefore omitted here. Thus, the one other LER [Nuclear Safety Information Center (NSIC) assession number 60227] is used to calculate the base-case, affected core-melt frequency for the hypothetical LWR.

LER number 60227 involved the failure of several main steam isolation valves (MSIVs) which failed to operate properly due to contamination film on the MSIV pilot valves. Considering this MSIV failure in sequence with a main steamline break initiating event and other system failures eventually leading to core damage, the probability of the MSIV failure leading to core damage was calculated to be 1.35E-06 in the precursor study. Dividing 1.35E-06 by 432 reactor years, the number of reactor years covered in the study, results in the frequency of core damage (3.13E-09/py) due to MSIV failure from contamination.

In this analysis, the core-melt frequency is adjusted upward by a factor of three to account for the two events which were omitted, but are cases of compressed air contamination. Therefore, the base-case, affected core-melt frequency for a hypothetical LWR is assumed to be 9.40E-09/py.

The issue resolution is assumed to reduce the base-case, affected core-melt frequency by 50%. Thus, the adjusted case, affected core-melt frequency is 4.7E-09/py.

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

$$W_O = \bar{F}_O R_O$$

where W_O = overall risk

$\bar{F}_O$ = overall core-melt frequency

R_O = average dose factor.

Denoting the number of plants as N and their average remaining lives as $\bar{T}$, the average dose factor for an LWR can be estimated as follows:

$$(R_o)_{LWR} = \frac{(N\bar{T}W_o)_{PWR} + (N\bar{T}\bar{W}_o)_{BWR}}{(NTF_o)^{PWR} + (NTF_o)^{BWR}}$$

Based on Appendices A-D of PNL-4297 (Andrews et al. 1983).

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

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

$\bar{T}$ = 28.8 yr (PWR) and 27.4 yr (BWR)

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

$\bar{F}_o$ = 8.2E-05/py (PWR) and 3.7E-05/py (BWR)

Thus, for this issue,

W = (9.4E-09/py)(3.3E+06 man-rem)

= .031 man-rem/py

W^* = (4.7E-09/py)(3.3E+06 man-rem)

= .0155 man-rem/py

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Contamination of Instrument Air Lines (43)

2. Affected Plants (N):

   All 71 Operating LWRs

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

   $\bar{T}$ = 26.9 yr

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

   $\Delta(\bar{F}D_R)$ = (19,900 man-rem)(4.7E-09/py) = 1.0E-4 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.8E-01	2.1	0

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

The occupational dose increase for SIR implementation is zero because the implementation of this issue resolution will consist of the issuance of an IE circular for all plants and addition of dryer subsystem for 10% of the plants. The installation of the dryer subsystem will not involve any radiation zone work (D = 0).

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

It is assumed that 75% of the labor associated with operation/maintenance (see Step 9 of Table 3) will involve work in radiation zones.

Thus,

Affected Plants	Labor in Radiation Zone (man-wk/py)
60 plants only performing annual air quality tests	(1)(0.75) = 0.75
7 plants taking appropriate corrective action to improve substandard air quality as well as performing annual air quality tests	(5)(0.75) = 3.75

TABLE 2. (cont'd.)

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

A 2.5 mR/hr radiation field is assumed.

Affected Plants	Occupational Dose (man-rem/py)
60	(0.0025 R/hr)(0.75 man-wk/py) (40 man-hr/man-wk) = 0.075 man-rem/py
7	(0.0025 R/hr)(3.75 man-wk/py) (40 man-hr/man-wk) = 0.375 man-rem/py

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

$N\bar{T}D_O$ = [(60 LWRs)(0.075 man-rem/py) + (7 LWRs)(0.375 man-rem/py)](28.3 yr)
= 2.0E+02 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
200	600	67

3.0 SAFETY ISSUE COSTS

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

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Contamination of Instrument Air Lines (43)

2. Affected Plants (N):

All 71 Operating LWRs

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

$\bar{T}$ = 26.9 yr

TABLE 3. (cont'd.)

Industry Costs (Steps 4 through 12):

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

$$\Delta(\bar{F}A) = (\$1.65E+09)(4.7E-09/py) = \$7.8E-02/py$$

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

Best Estimate	Upper Bound	Lower Bound
\$1.5E+04	\$1.8E+05	0

6. Per-Plant Industry Resources for SIR Implementation:

Cost is estimated directly in the next step.

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

A. For All LWRs with Operating Licenses:

Based on recommended action No. 1 and No. 2 of the SIR, each of the 71 operating LWRs will perform a design review related to 1) the significance of the safety role played by the compressed-air systems and 2) whether redundant safety-related components are being supplied or are capable of being supplied from a common compressed-air source. This design review is estimated to cost \$100,000/plant based upon an assumed one man-year of effort.

Based on recommended action No. 2 of the SIR, it is assumed that 50% of the 71 operating LWRs (36) will need to provide corrective action because safety consequences of a single failure resulting from air contamination are considered unacceptable. The air contamination is assumed to lead to a common-cause failure of sufficient redundant components to cause the loss of the associated system's safety function when required. This corrective action is assumed to be the installation of an auxiliary air purification system to be used to purify the service air when it is called upon to replace the instrument air system. The cost to install an auxiliary air purification system is estimated to be \$250,000/plant.

In summary:

TABLE 3. (cont'd.)

Affected Plants	I ($/plant)
35 operating LWRs only performing design reviews	1.0E+05
36 operating LWRs adding auxiliary air purification systems as well as performing design reviews	2.5E+05 + 1.0E+05 = 3.5E+05

B. For all LWRs with Construction Permits:

The cost of design evaluations of compressed-air systems is part of the original design cost.

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

NI= (35 LWRs)($1.0E+05/plant) + (36 LWRs)($3.5E+05/plant) = $1.6E+07

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

Based on recommended action No. 1 under actions for holders of operating licenses, it is assumed that 50% of all 134 LWRs will perform an annual air quality test. Each air quality test is assumed to require 1 man-wk/py. It is also assumed that 10% of all air quality tests will indicate the need to take appropriate corrective actions such as blowdown, cleanup, and testing. The appropriate corrective action is assumed to require 4 man-wk/py.

Affected Plants	Labor (man-wk/py)
60 plants only performing annual air quality tests	1
7 plants taking appropriate corrective action to improve substandard air quality as well as performing annual air quality tests	1 + 4 = 5

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

Affected Plants	I_0 ($/py)
60	2270
7	11,400

TABLE 3. (cont'd.)

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

$$N\bar{T}I_O = [(60\ \text{plants})(\$2270/\text{py}) + (7\ \text{plants})(\$11{,}400/\text{py})](28.3\ \text{yr}) = \$6.1\text{E}{+}06$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$2.2E+07	\$3.1E+07	\$1.3E+07

NRC Costs (Steps 13 through 21)

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

The resolution to this issue in the form of an IE Bulletin is expected to have been finalized in 1983. Therefore, there are no NRC costs for development after 1983 ($C_D = 0$).

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

To support SIR implementation, the following amounts of NRC labor are assumed:

a. Monitor and review design reviews
4 man-wk/plant (71 LWRs)

b. Monitor and review air purification system installation
4 man-wk/plant (36 LWRs)

Thus,

Affected Plants	Labor (man-wk/plant)
35 LWRs (a only)	4
36 LWRs (a & b)	4 + 4 = 8

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

Affected Plants	C (\$/plant)
35 LWRs (a only)	9080
36 LWRs (a & b)	18,200

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

$$NC = (35\ \text{LWRs})(\$9080/\text{plant}) + (36\ \text{LWRs})(\$18{,}200/\text{plant}) = \$9.7\text{E}{+}05$$

TABLE 3. (cont'd.)

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

To review SIR operation/maintenance, the following amounts of NRC labor are assumed:

a. Review air quality tests (for 67 LWRs performing air quality tests) = 2 man-hr/py (or 0.05 man-wk/py)

b. Review of air quality tests corrective actions - (for 7 LWRs taking corrective actions) = 4 man-hr/py (or 0.10 man-wk/py)

Thus,

Affected Plants	(man-wk/plant)
60 LWRs (a only)	0.05
7 LWRs (a & b)	0.10 + 0.05 = 0.15

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

Affected Plants	C_O (\$/py)
60 LWRs (a only)	110
7 LWRs (a & b)	340

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

$$N\bar{T}Co = [(60\ LWRs)(\$110/py) + (7\ LWRs)(\$340/py)](28.3\ yr)$$
$$\simeq \$2.5E+05$$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$1.2E+06	\$1.7E+06	\$7.0E+05

REFERENCES (For Issue 43)

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.

Minarick, J. W., and C. A. Kukielka. 1982. Precursors to Potential Severe Core Damage Accidents: 1969-1979. A Status Report. NUREG/CR-2497, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 45, Inoperability of Instruments Due to Extreme Cold

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This issue is concerned with the freezing of instruments due to cold weather. The proposed resolution is the implementation of I & E Inspection Manual Amendment, Procedure No. 71714, and the timely issuance and implementation of Draft Regulatory Guide Instrument Sensing Lines (Task IC 126-5). These are intended to ensure that licensees take measures to prevent instrument freezing.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	16
OCCUPATIONAL DOSES:	
SIR Implementation =	1.8
SIR Operation/Maintenance =	284
Total of Above =	290
Accident Avoidance =	0.099

COST RESULTS($$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	0.23
SIR Operation/Maintenance =	13
Total of Above =	13
Accident Avoidance =	0.0081
NRC COSTS:	
SIR Development =	0
SIR Implementation Support =	0.0014
SIR Operation/Maintenance Review =	0.43
Total of Above =	0.43

INOPERABILITY OF INSTRUMENTS DUE TO EXTREME COLD

ISSUE 45

1.0 SAFETY ISSUE DESCRIPTION

This issue is concerned with the freezing of instruments due to cold weather. Past occurrences of instrument freeze-up can be grouped into three categories:

- refueling water or borated water storage tank level instrumentation
- main steamline pressure and flow instrumentation sensing lines
- radiological effluent sampling lines.

Many of the occurrences were directly related to inadequacies associated with the heat tracing provided for these sensing and sampling lines. Some of the commonly reported causes of line freeze-up are the absence of heat tracing or adequate insulation, de-energized heat trace circuits, improper thermostat settings or sensor location for the heat tracing, and space heater failures.

The issuance of IE Bulletin 79-24, "Frozen Lines," dated September 27, 1979, which addressed this subject, apparently has not resulted in a decrease in the incidence of frozen lines. The Office of Inspection and Enforcement (I&E) issued an amendment to the Inspection Manual on January 1, 1982, (Procedure No. 71714). The objective of this procedure is to "ascertain whether the licensee has maintained effective implementation of the program of protective measures for extreme cold weather committed to in response to IE Bulletin 79-24." In addition, a Regulatory Guide (Task IC 126-5), titled "Instrument Sensing Lines," is to be issued. Among other considerations, the Regulatory Guide covers the issue of prevention of frozen instrument lines.

ISSUE RESOLUTION

For purposes of this analysis, this proposed safety issue resolution (SIR) is assumed to involve the following actions: 1) Implementation of I&E Inspection Manual Amendment, Procedure No. 71714, for existing plants, and 2) the timely issuance and implementation of the aforementioned Regulatory Guide (Task IC 126-5) for plants under construction.

AFFECTED PLANTS

This issue affects all 90 PWRs and all 44 BWRs, including those completed and those under construction.

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 PWR and BWR, a consequence of the data base employed for this issue analysis.[a]

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Inoperability of Instruments Due to Extreme Cold (45)

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

   All 134 plants are assumed to be affected ($\bar{T}$ = 28.3 yr).

3. Plants Selected for Analysis:

   A hypothetical LWR is assumed to be representative of all LWRs.[a]

4-7. Steps Related to Affected Parameters, Accident Sequences and Release Categories and Their Base-Case Values:

   The base-case, affected core-melt frequency is estimated directly in Step 8. Thus, these steps are omitted.

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

   $\bar{F}$ = 2.6E-09/py[a]

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

   W = 8.6E-03 man-rem/py[a]

10-12. Steps Related to Affected Parameters, Accident Sequences and Release Categories and Their Adjusted-Case Values:

   The adjusted-case, affected core-melt frequency is estimated directly in Step 13. Thus, these steps are omitted.

(a) See Attachment 1.

TABLE 1. (contd)

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

$\bar{F}^* = 1.3E-09/py$[a]

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

$W^* = 4.3E-03$ man-rem/py[a]

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

$\Delta\bar{F} = 1.3E-09/py$

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

$\Delta W = 4.3E-03$ man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
16	985	0

(a) See Attachment 1.

ATTACHMENT 1 (To Table 1)

The Oak Ridge precursor study (Minarick and Kukielka 1982) is used to calculate the base-case, affected core-melt frequency for a hypothetical LWR. This study is based on analysis of Licensee Event Reports (LERs) from 1969 to 1979. The calculation of the base-case, affected core-melt frequency uses LER data which are assumed to be representative of all LWRs.

Two LERs were considered in the precursor study that involved frozen instrument sensing lines. They are Nuclear Safety Information Center (NSIC) assession numbers 109295 and 122366 and are used to calculate the base-case, affected core-melt frequency for the hypothetical LWR. LER number 109295 involved the failure of both channels of a Borated Water Storage Tank (BWST) level indicator due to freezing of moisture in the level transmitter impulse lines. Considering this failure in sequence with a small LOCA and other system failures eventually leading to core damage, the probability of the freezing of BWST level indicator transmission lines leading to core damage was calculated to be 3.4E-05.

LER number 122366 involved the freezing of all four Steam Line Break Instrumentation and Control (SLBIC) sensing lines. Considering this failure in sequence with a steam line break and other system failures eventually leading to core damage, the probability of the freezing of SLBIC sensing lines leading to core damage was calculated to be 4.6E-08. The probability of frozen instrument sensing lines leading to core damage is the sum of these aforementioned probabilities, or 3.4E-05.

Dividing the probability of 3.4E-05 by the 432 reactor years covered in the study results in the frequency of core damage due to frozen instrument sensing lines. This quotient is 7.9E-08/py.

As reported in Nucleonics Week, October 2, 1982, an analysis by the Institute of Nuclear Power Operations (INPO) of the Oak Ridge precursor study claims that the chances of a severe nuclear accident were estimated 30 times too high. Furthermore, severe core damage (assumed to be analogous to that at TMI-2 in the precursor study) is presumably less severe than core-melt, the level of core damage normally considered in nuclear power plant risk studies. Based on these considerations, it is assumed for this study that the frequency of core damage as assessed using the precursor study should be divided by INPO's factor of 30 to result in the frequency of core-melt. Therefore, the base-case, affected core-melt frequency for a hypothetical LWR is assumed to be 2.6E-09/py.

The issue resolution is assumed to reduce the base-case, affected core-melt frequency by 50%. The adjusted-case, affected core-melt frequency is, then, 1.3E-09/py.

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

$$W_o = \bar{F}_o R_o$$

where

W_o = overall risk

$\bar{F}_o$ = overall core-melt frequency

R_o = average dose factor

Denoting the number of plants as N and their average remaining lives as $\bar{T}$, the average dose factor for an LWR can be estimated as follows:

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

Based on Appendices A-D of PNL-4297 (Andrews et al. 1983),

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

where

N = 90 (PWR) and 44 (BWR)

$\bar{T}$ = 28.8 yr (PWR) and 27.4 yr (BWR)

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

$\bar{F}_o$ = 8.2E-05/py (PWR) and 3.7E-05/py (BWR)

Thus, for this issue,

W = (2.6E-09/py)(3.3E+06 man-rem)
 = 8.6E-03 man-rem/py)

W* = (1.3E-09/py)(3.3E+06 man-rem)
 = 4.3E-03 man-rem/py

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Inoperability of Instruments Due to Extreme Cold (45)

2. Affected Plants (N):

   All 134 plants (71 operating and 63 planned) are assumed to be affected.

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

   $$\left.\begin{aligned} \text{Operating } \bar{T} &= 26.9 \text{ yr} \\ \text{Planned } \bar{T} &= 30 \text{ yr} \end{aligned}\right\} \quad \bar{T} = 28.3 \text{ yr for all plants}$$

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

   $\Delta(\bar{F}D_R)$ = (19,860 man-rem)(1.3E-09/py)= 2.6E-5 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
0.099	1.2	0

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

   For the 71 operating plants, it is assumed that one-third, or 24 plants, will require additional installation of freeze prevention devices and that one man-week (40 man-hours) per plant of labor will be required. With a 75% utilization factor, the labor becomes 30 man-hr/plant in radiation zones. For planned plants, no additional work in radiation zones is foreseen since installation of freeze protection devices is already required per IE Bulletin 79-24.

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

   It is assumed that environmental areas subject to freezing are outside reactor containment. The dose rate is assumed to be 2.5 mR/hr.

   D = (0.0025 R/hr)(30 man-hr/plant) = 0.075 man-rem/plant

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

   ND = (24 plants)(0.075 man-rem/plant) = 1.8 man-rem

TABLE 2. (contd)

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

It is assumed that for all LWRs, maintenance requires 8 man-hr/plant, 5 times a year, to inspect and verify that the freeze protection devices are operable, or 40 man-hr/py. Again, using a 75% utilization factor, this labor becomes 30 man-hr/py in radiation zones.

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

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

D_0 = (0.0025 R/hr)(30 man-hr/py) = 0.075 man-rem/py

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

$N\bar{T}D_0$ = (134 plants)(28.3 yr)(0.075 man-rem/py) = 284 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
290	860	95

3.0 SAFETY ISSUE COSTS

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

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Inoperability of Instruments Due to Extreme Cold (45)

2. Affected Plants (N):

All 134 plants (71 operating and 63 planned) are assumed to be affected.

TABLE 3. (contd)

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

$$\left.\begin{array}{r}\text{Operating } \bar{T} = 26.9 \text{ yr} \\ \text{Planned } \bar{T} = 30 \text{ yr}\end{array}\right\} \quad \bar{T} = 28.3 \text{ yr for all plants}$$

Industry Costs (Steps 4 through 12):

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

$\Delta\bar{F}A$ = (1.3E-09/py)($1.65E+09) = $2.1/py

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

Best Estimate	Upper Bound	Lower Bound
$8.1E+03	$9.8E+04	0

6. Per-Plant Industry Resources for SIR Implementation:

For the 24 operating plants (one-third of total, see Step 6 in Table 2), the assumed resources are as follows:

Labor = 40 man-hr/plant for installation (Step 6, Table 2) plus 40 man-hr/plant for planning and record-keeping.

Equipment = additional heat tracing, spare heaters, temperature indicators as required.

For operating plants, no additional labor, down-time or equipment is needed because the installation of freeze protection devices is already required per IE Bulletin 79-24.

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

Labor = (80 man-hr/plant)($2270/man-wk)(1 man-wk/40 man-hr)
= $4.54E+03/plant
Equipment = $5.00E+03/plant
Total = $9.54E+03/plant

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

NI = (24 plants)($9540/plant) = $2.29E+05

TABLE 3. (contd)

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

Labor = 40 man-hr/py for maintenance (Step 9, Table 2) plus 20 man-hr/py for planning and record-keeping.

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

I_O = (60 man-hr/py)($2270/man-wk)(1 man-wk/40 man-hr)
= $3.41E+03/py

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

$N\bar{T}I_O$ = (134 plants)(28.3 yr)($3410/py) = $1.29E+07

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$1.3E+07	$2.0E+07	$6.7E+06

NRC Costs (Steps 13 through 21):

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

The resolution to this issue involves the implementation of I&E Inspection Manual Amendment, Procedure No. 71714, and the issuance and implementation of Draft Regulatory Guide Instrument Sensing Lines (Task IC 126-5). The I&E Amendment was issued on 1/1/82 and the Regulatory Guide is to be issued in early 1983. Therefore, there are no additional NRC costs for development (C_D = 0).

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

For the existing plants which already have adequate freeze prevention devices, there is no NRC labor time for implementation. For the assumed 24 existing plants (Step 6, Table 2) which need to improve their freeze protection capability, one man-hour of NRC labor time is assumed to be required per plant for record-keeping. For the planned plants, no additional NRC labor time is foreseen because the installation of freeze protection devices is already required per IE Bulletin 79-24.

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

C = (1 man-hr)($2270/man-wk)(1 man-wk/40 man-hr) = $56.8/plant

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

$$NC = (24\ \text{plants})(\$56.8/\text{plant}) = \$1.36E+03$$

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

Per I&E Inspection Manual Amendment, Procedure No. 71714, the inspector is to verify that the licensee has maintained effective implementation of the program of protective measures for extreme cold weather. It is assumed that this verification will take place during routine inspections and that two additional man-hours will be necessary per plant-year for inspection of the licensee's records.

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

$$C_O = (2\ \text{man-hr/py})(\$2270/\text{man-wk})(1\ \text{man-wk}/40\ \text{man-hr}) = \$114/\text{py}$$

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

$$N\bar{T}C_O = (134\ \text{plants})(28.3\ \text{yr})(\$114/\text{py}) = \$4.30E+05$$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$4.3E+05	\$6.5E+05	\$2.2E+05

REFERENCES (For Issue 45)

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

Minerick, J., and C. Kukielka. 1982. Precursors to Potential Severe Core Damage Accidents, 1969-1979: A Status Report. NUREG/CR-2497, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 49, Interlocks and LCOs (Limiting Conditions of Operations) for Class 1E Tie Breakers

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Several operating plants have only one tie breaker between redundant Class 1E buses. If this tie breaker was in the closed position during a need for emergency power and subsequently failed to open, then the emergency diesel generators would be prevented from supplying power to these buses. The proposed resolution is to require, in the form of an IE circular, that all such bus tie breakers be physically disengaged except during shutdown maintenance activities, and that QA procedures be provided which double check that the tie breakers are disengaged and red tagged prior to each startup.

AFFECTED PLANTS	PWR:	Operating = 47	Planned = 0
	BWR:	Operating = 24	Planned = 0

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	16
OCCUPATIONAL DOSES:	
SIR Implementation =	1.1
SIR Operation/Maintenance =	18
Total of Above =	19
Accident Avoidance =	8.4

COST RESULTS ($1E+06)

INDUSTRY COSTS:	
SIR Implementation =	0.17
SIR Operation/Maintenance =	0.54
Total of Above =	0.71
Accident Avoidance =	0.69
NRC COSTS:	
SIR Development =	0.0091
SIR Implementation Support =	0.032
SIR Operation/Maintenance Review =	0
Total of Above =	0.041

INTERLOCKS AND LCOs

(LIMITING CONDITIONS OF OPERATIONS)

FOR CLASS 1E TIE BREAKERS

ISSUE 49

1.0 SAFETY ISSUE DESCRIPTION

This safety issue is concerned with facilities whose AC electrical distribution system employs only one tie breaker between 4160-V Class 1E redundant buses. If this tie breaker is left closed during normal operation and then fails to open on emergency power demand, the closed tie breaker will prevent power from being supplied to the 4160-V Class 1E redundant buses.

Concern for this issue arose when the Wisconsin Electric Power Company reported to the NRC on June 27, 1980, that Point Beach 2 had operated at power for a period of about five weeks with one of the 4160-V Class 1E redundant buses being supplied by the offsite power source via the other 4160-V redundant bus tie breaker. This occurred because of a personnel error in failing to trip the tie breaker and to close the respective offsite source breaker.

GDC-17 requires that the onsite source and distribution systems have sufficient independence and redundancy to perform their safety function assuming a single failure. Operating the plant for five weeks in the reported configuration violates the independency requirement of being able to accommodate a single failure.

The following are the design features of the 4160-V Class 1E bus feeders and tie breaker at Point Beach 2:

(a) The tie breaker opens upon undervoltage on either of the two 4160-V Class 1E buses.

(b) Neither one of the two diesel generator breakers can be closed if the tie breaker is closed.

(c) The bus tie breaker cannot be closed if both offsite source breakers are closed.

(d) Neither one of the two offsite source breakers can be closed if the tie breaker is closed.

With these features of breaker operation, there are at least the following problems having potential for impairing plant safety:

1. A failure of the tie breaker to open on loss of voltage would prevent both emergency diesel generators from automatically supplying power to their respective buses (single failure).

2. The tie breaker is capable of being closed when the offsite source breaker is closed on one bus and the respective diesel generator breaker is closed on the other bus. (Paralleling two divisions, one with offsite and the other with emergency sources.)

3. The tie breaker is capable of being closed when both 4160-V Class 1E buses are being supplied by their respective diesel generators. (Paralleling redundant emergency sources.) This is contrary to the requirement of Regulatory Guide 1.6 which states, "If means exist for manually connecting redundant load groups together, at least one interlock should be provided to prevent an operator error that would parallel their standby power sources."

PROPOSED RESOLUTION

For purposes of this analysis, the proposed resolution to this safety issue is to require that the NRC finalize and issue the proposed OIE (Office of Inspection and Enforcement) circular[a] requiring the following action for holders of an operating license:

All holders of operating licenses should review the design and operational features of all Class 1E bus tie breakers.

If only one tie breaker exists between redundant Class 1E buses, then the licensee should promptly take, as a minimum, the following actions, via procedural requirements (these are taken as the proposed resolution for affected plants):

1. Use a bus tie breaker only during shutdown when it is absolutely necessary.

2. Physically disengage each tie breaker and rack out (withdraw) following each usage.

3. "Red tag" the tie breaker enclosure for the breaker to be kept open.

4. Incorporate QA procedures to reconfirm that all tie breakers are racked out and "red tagged" prior to each plant startup.

AFFECTED PLANTS

The present licensing practice as stated in Section 8.3.1, III.2.C of the Standard Review Plan requires two physically separated tie breakers in series between redundant Class 1E buses. In addition, the standard technical specifications for new plants require tie breakers between redundant buses to be open as a condition of operability of the redundant Class 1E electrical distribution system. Thus, only operating LWRs are affected.

(a) The proposed OIE circular is an enclosure to the November 24, 1980 memorandum from D. Eisenhart to F. Schroeder entitled "Tie Breakers Between Redundant Class 1E Buses."

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.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Interlocks and LCOs (Limiting Conditions of Operations) for Class 1E Tie Breakers (49)

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

   Only operating LWRs will be affected by this issue.

	N	$\bar{T}$(yr)
PWR	47	27.7
BWR	24	25.2

3. Plants Selected for Analysis:

   The ORNL Precursor Study[(a)]

4-7. Steps Related to Affected Parameters, Accident Sequences, Release Categories and Their Base-Case Values:

   A discussion of these areas is found in Attachment 1.

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

   $\bar{F}_{PWR}$ = 3.2E-07/py $\bar{F}_{BWR}$ = 1.4E-09/py

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

   W_{PWR} = (3.2E-07)(2.5E+06) = 8.0E-01 man-rem/py
   W_{BWR} = (1.4E-09)(6.8E+06) = 9.5E-03 man-rem/py

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

   Resolution of this issue is expected to eliminate this problem. Thus, the adjusted-case values would be zero.

(a) See Attachment 1.

TABLE 1. (cont'd.)

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

$\Delta\bar{F}_{PWR}$ = 3.2E-07/py $\Delta\bar{F}_{BWR}$ = 1.4E-09/py

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

ΔW_{PWR} = 8.0E-01 man-rem/py ΔW_{BWR} = 9.5E-03 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
16	485	0

ATTACHMENT 1 (To Table 1)

The Oak Ridge precursor study is used to calculate the base-case, affected core-melt frequency for a hypothetical PWR and BWR (Minarick and Kukielka 1982). The Oak Ridge precursor study is based on analyses of Licensee Event Reports (LERs) from 1969-1979. The calculation of the base-case affected core-melt frequency uses failure probabilities from LERs. Standardized event trees (see Figures 1 and 2) from the Oak Ridge study are used to obtain the accident sequences affected by this issue. The failure probabilities from the LERs were calculated based on an estimate of the total number of test demands and the number of additional non-test demands to which the function would be expected to respond. The calculations of the base-case, affected core-melt frequency used the standardized event trees and failure probabilities from the LERs. It is assumed that the calculations are representative of all PWRs and BWRs.

The event trees in Figures 1 and 2 are used to determine the base-case affected core-melt frequency. The events in Figures 1 and 2 have been designated with the letters I, A, B, C, D, E, F, and G.

The probability of the emergency diesel generators failing on demand to supply power Class 1E bases is conditional on the tie breaker between these buses being in the failed position during a loss of offsite power.

For PWRs and BWRs the frequency of a Loss of Offsite Power (LOOP) occurring is x - 0.2/py. The Mean Time To Repair (MTTR) for the failed single tie breaker is assumed to be one hundred hours.

$$\text{MTTR} = 100 \text{ hour} = \frac{100 \text{ hr}}{(8760 \text{ hr/yr})} = 1.1\text{E-02 yr}$$

The likelihood of a loss of offsite power is then[(a)]:

$$I = x(\text{MTTR}) = (0.2/\text{yr})(1.1\text{E-02 yr}) = 2.28\text{E-03}$$

Only PWRs have been identified as having one tie breaker between buses in their electrical distribution system. Furthermore, failure of a single tie breaker has only been documented once. Thus, it is assumed that, in the 343 PWR plant-years recorded to date, this incident has occurred just once, giving an occurrence (failure) frequency of 1/(343/py) = 2.92E-03/py. For BWRs, the zero failure approximation based on the chi-square distribution is used (Green and Bourne 1972). With a 50 percent confidence limit for zero failures and the 605 BWR plant-years recorded to date, this incident is assumed to have an occurrence (failure) frequency of (1.39/2)/(235 py) = 2.96E-03/py.

(a) Analysis using the precursor study methodology requires an estimate of the likelihood of a sequence's initiating event occurring while one of the sequence's conditional failures is in effect. Thus, the MTTR of the tie breaker (taken as one hundred hours) is used to determine the likelihood of the initiating event (LOOP) occurring while the breaker is in its failed state.

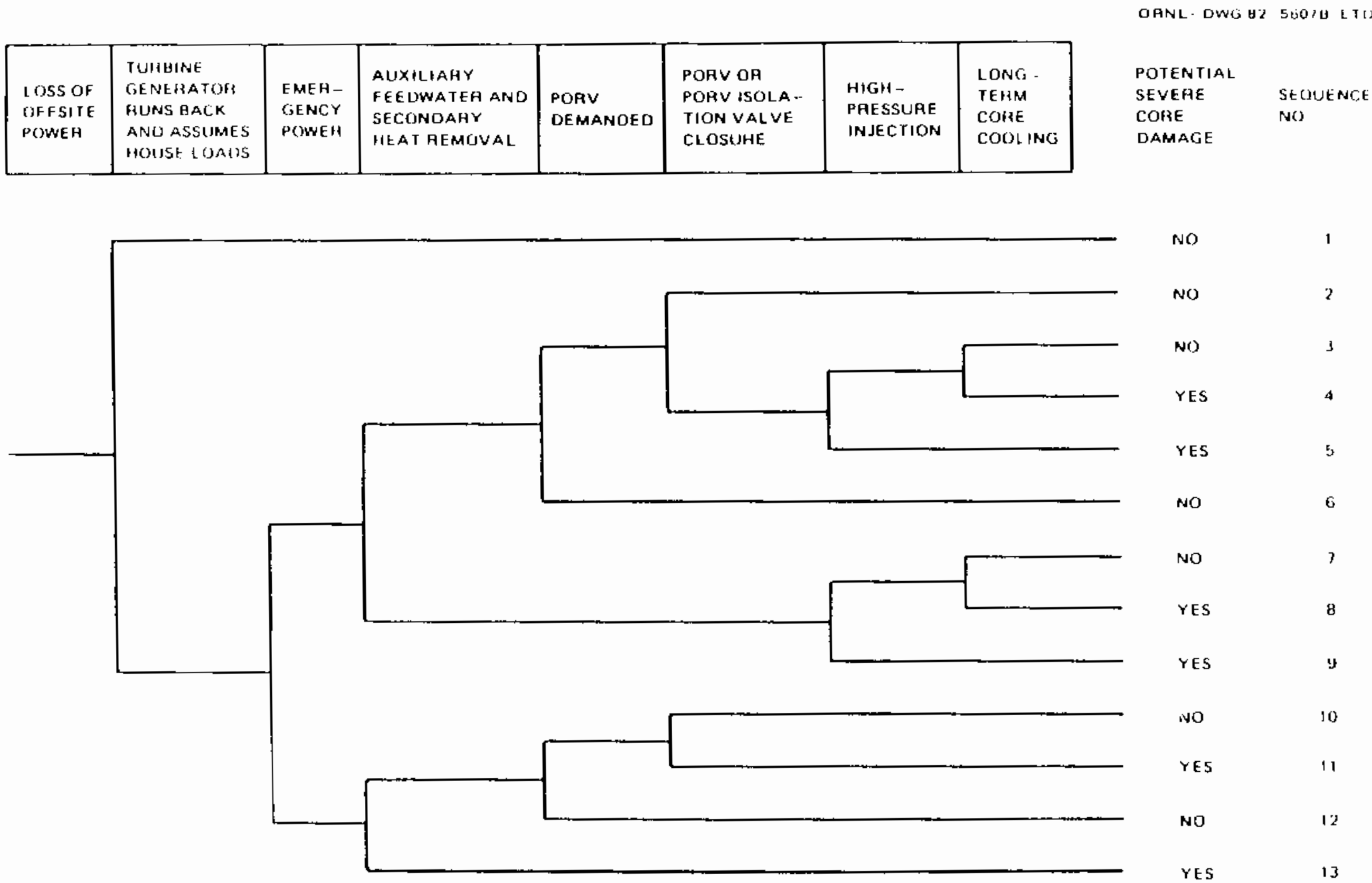

FIGURE 1. Standard event for PWR loss of offsite power (Minarick and Kukielka 1982, Figure A.3)

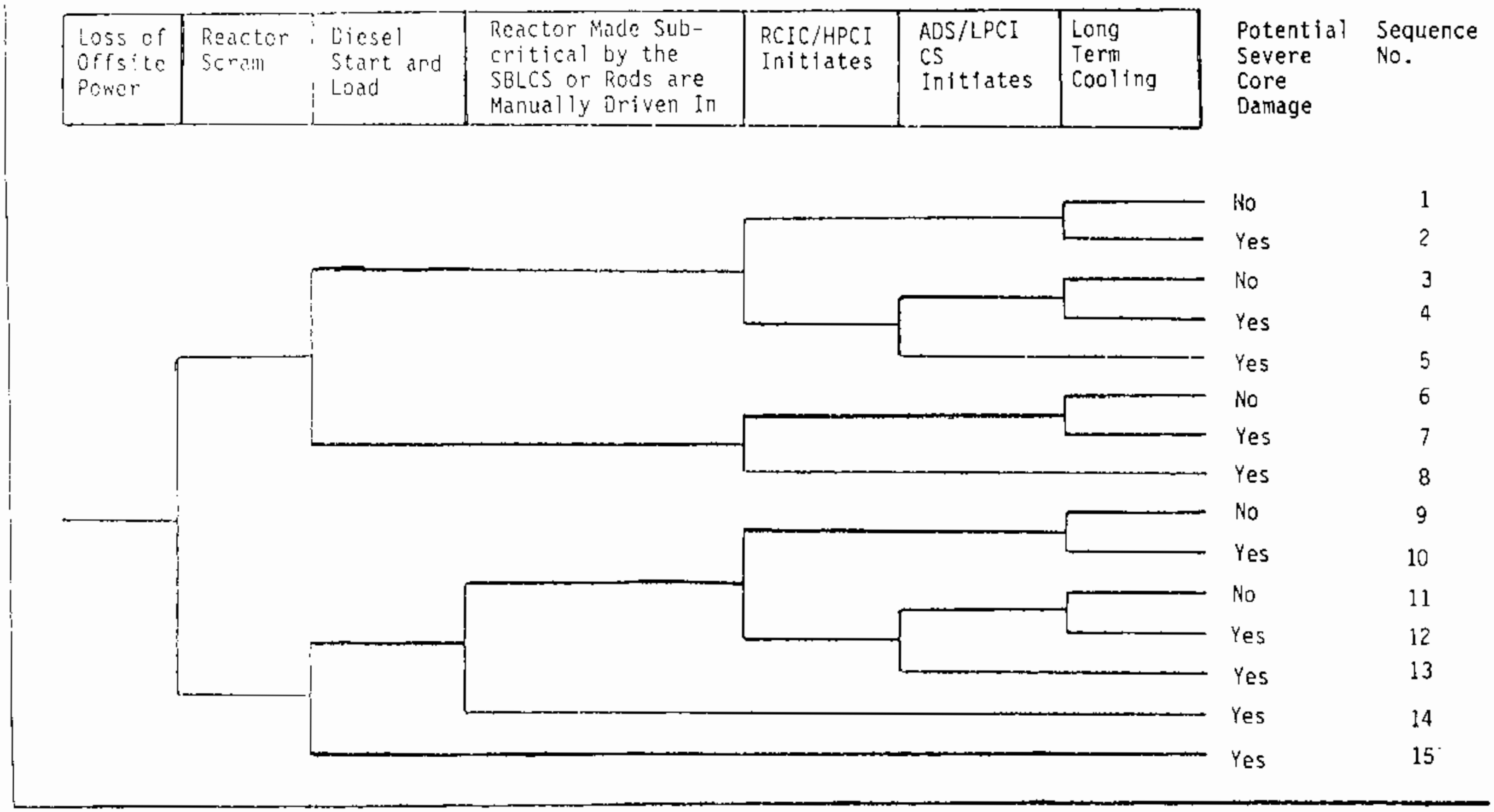

FIGURE 2. Standard event tree for BWR loss of offsite power. (Minarick and Kukielka 1982, Figure A.1)

Based on engineering judgment, an accounted failure on demand appearing capable of short-term rectification at an accessible failure location was assumed to have a nonrectification probability of 0.5 in the precursor study. This is judged applicable to failure of the single Class 1E tie breaker. Thus, the probability of failure for emergency power at a PWR ($\bar{B}$) due to the single tie breaker failure is taken as 0.5. Likewise, the failure probability for the diesel generators at a BWR to start and load ($\bar{B}$) due to the single tie breaker failure is taken as 0.5 also.

Since failure of the single Class 1E tie breaker is manifested as a loss of emergency AC power (B), only those sequences containing (B) in Figures 1 and 2 can result in severe core damage due to this failure. For the hypothetical BWR and PWR, these sequences are as follows:

PWR[a]	BWR[a]
---	--
IABC	IBF
	$I\bar{B}\bar{D}$
	$I\bar{A}\bar{B}$

The probability of severe core damage due to the tie breaker failure is the sum of the above sequence probabilities (using the values listed in Figures 1 and 2 for A thru $\bar{F}$ and I = 2.28E-03).

$$P\ (\text{severe core damage}) = \begin{array}{ll} 1.1\text{E-03} & (\text{PWR}) \\ 4.6\text{E-06} & (\text{BWR}) \end{array}$$

The frequency of severe core damage is the produce of the probability of severe core damage and the occurrence frequency of the tie breaker failure (2.92E-03/py for the PWR and 2.96E-03/py for the BWR):

$$F\ (\text{severe core damage}) = \begin{array}{ll} (1.1\text{E-03})(2.92\text{E-03/py}) = 3.2\text{E-06/py} & (\text{PWR}) \\ (4.6\text{E-06})(2.96\text{E-03/py}) = 1.4\text{E-08/py} & (\text{BWR}) \end{array}$$

An analysis by the Institute of Nuclear Power Operations (INPO) on the Oak Ridge Precursor study claims that the chances of a severe nuclear accident were estimated conservatively to as much as 30 times too high as reported in INPO 82-025, September 1982. In addition, to this point in the analysis no credit has been taken for the recognition and correction of the tie breaker configuration within a safe period of time i.e., no core damage once a LOOP has begun. Hence, it can be argued that the frequency of core damage as assessed using the precursor study is highly conservative. It has been assumed for the purposes of this study that the core damage frequency as assessed using the precursor study should be divided by 10. Note that division by 10 corresponds to a probability of 0.1 for recognition and correction within a safe period after initiation of a LOOP; and is 1/3 of the INPO factor.

(a) Only failure events are given in each sequence.

ATTACHMENT 1. (cont'd.)

$$\bar{F}_{PWR} = (3.2E-06/py)(.1) = 3.2E-07/py$$

$$\bar{F}_{BWR} = (1.4E-08/py)(.1) = 1.4E-09/py$$

Dose factors for the PWR and BWR core-melt releases were developed using the release category weighted average values from Oconee and Grand Gulf RSSMAP studies with meteorology similar to that of the Braidwood site. Results are as follows:

Oconee: 2.5E+06 man-rem/core-melt
Grand Gulf: 6.8E+06 man-rem/core-melt

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

$$W_o = \bar{F}_o R_o$$

where W_o = overall risk

$\bar{F}_o$ = overall core-melt frequency

R_o = average dose factor

Based on Appendices A-D of PNL-4297 (Andrews et al. 1983),

$$(R_o)_{PWR} = \frac{W_{oPWR}}{\bar{F}_{oPWR}} = \frac{207 \text{ man-rem/py}}{8.2E-05/py} = 2.5E+06 \text{ man-rem}$$

$$(R_o)_{BWR} = \frac{W_{oBWR}}{\bar{F}_{oBWR}} = \frac{250 \text{ man-rem/py}}{3.7E-05/py} = 6.8E+06 \text{ man-rem}$$

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

$\bar{F}_o$ = 8.2E-05/py (PWR) and 3.7E-05/py (BWR).

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Interlocks and LCOs (Limiting Conditions of Operations) for Class 1E Tie Breakers (49)

2. Affected Plants (N):

   Only some operating LWRs will be affected by this issue.

	N
PWRs:	47
BWRs:	24

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

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

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

   PWR: (19,900 man-rem) (3.2E-07/py) = 6.4E-03 man-rem/py

   BWR: (19,900 man-rem) (1.4E-09/py) = 2.8E-05 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
8.3	50	0

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

   SIR implementation is assumed to require one man-day/plant to "red tag" the appropriate tie breakers.[a] With a 75% utilization factor for radiation zone work, this labor estimate becomes 0.75 man-day/plant for radiation zone work (at both PWRs and BWRs).

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

   The tie breakers are located outside of containment; thus, a 2.5 mR/hr radiation field is assumed.

$$D = (0.0025\ R/hr)(0.75\ man\text{-}day/plant)(8\ man\text{-}hr/man\text{-}day)$$
$$= 0.015\ man\text{-}rem/plant$$

(a) See Attachment 2 to the Safety Issue Cost Work Sheet.

TABLE 2. (cont'd.)

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

$$ND = (24 + 47)(0.015 \text{ man-rem/plant}) = 1.1 \text{ man-rem}$$

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

SIR operation/maintenance is assumed to require 5 man-hr/py for racking out and red tagging tie breakers.[a] With a 75% utilization factor for radiation zone work, this labor estimate becomes 3.75 man-hr/py for radiation zone work (at both PWRs and BWRs).

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

As in Step 7, a 2.5 mR/hr radiation field is assumed.

$$D_o = (0.0025 \text{ R/hr})(3.75 \text{ man-hr/py}) = 0.0094 \text{ man-rem/py}$$

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

$$N\bar{T}D_o = [(24)(25.2 \text{ yr}) + (47)(27.7 \text{ yr})](.0094 \text{ man-rem/py}) = 18 \text{ man-rem}$$

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
19	57	6.3

3.0 SAFETY ISSUE COSTS

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

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Interlocks and LCOs (Limiting Conditions of Operations) for Class 1E Tie Breakers (49)

(a) See Attachment 2 to the Safety Issue Cost Work Sheet.

TABLE 3. (cont'd.)

2. Affected Plants (N):

	N
PWRs	47
BWRs	24
All	71

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

	$\bar{T}$ (yr)
PWRs	27.7
BWRs	25.2

Industry Costs (Steps 4 through 12)

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

PWR: (\$1.65E+09)(3.2E-07/py) = \$528/py

BWR: (\$1.65E+09)(1.4E-09/py) = \$2.31/py

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

Best Estimate	Upper Bound	Lower Bound
\$6.9E+05	\$4.1E+06	0

6. Per-Plant Industry Resources for SIR Implementation:

	Labor (man-hr/plant)[a]
71 operating LWRs performing design review and corrective action ("affected" LWRs)	42

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

I (affected LWRs) = (42 man-hr/plant)(1 man-wk/40 man-hr)
(\$2270/man-wk) = \$2380/plant

(a) See Attachment 2.

TABLE 3. (cont'd.)

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

$NI = (71)(\$2380/\text{plant}) = \$1.7E+05$

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

Maintenance and QA = 5 man-hr/py (at each of 71 affected LWRs)[a]

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

$I_o = (5\ \text{man-hr/py})(1\ \text{man-wk}/40\ \text{man-hr})(\$2270/\text{man-wk}) = \$284/\text{py}$

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

$N\bar{T}I_o = [(24)(25.2\ \text{yr}) + (47)(27.7\ \text{yr})](\$284/\text{py}) = \$5.4E+05$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$7.1E+05	\$9.9E+05	\$4.3E+05

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

It is assumed that finalizing the proposed OIE circular (see "Issue Resolution" for Issue 49 in Section 1.0) and distributing it to the licensees will require four man-weeks of NRC staff labor.

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

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

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

It is assumed that NRC staff will expend 0.5 man-day/plant to review the response from each of the 71 operating LWRs addressing the OIE circular.

	NRC Support Labor (man-day/plant)
71 affected LWRs	1.0

(a) See Attachment 2.

TABLE 3. (cont'd.)

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

C (affected LWRs) = (1.0 man-day/plant)(1 man-wk/5 man-days)
($2270/man-wk) = $454/plant

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

NC = (71)($454/plant) = $3.2E+04

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

Monitoring of utility compliance with the corrected procedures should be included as part of the annual NRC inspections. No additional effort beyond that currently expended by the NRC is foreseen.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$4.1E+04	$5.8E+04	$2.4E+04

ATTACHMENT 2 (To Table 2)

Industry resources for SIR implementation, operation, and maintenance are estimated here.

SIR IMPLEMENTATION

Based on the recommended action that "all holders of operating licenses should review the design and operational features of all Class 1E bus tie breakers," it is assumed that 71 of the operating LWRs will conduct such a review to identify whether it is one of the affected plants. This is estimated to require two man-hours/plant. All LWRs must respond to the NRC that they have taken action to the circular and report their findings. The estimated time for the licensees to write this response is 0.5 man-day/plant.

Four actions (see "Issue Resolution" for Issue 49 in Section 1.0) remain for plants that are identified as being affected by this issue.

Action 1 is not a major change in operating procedures. Four man-hours/plant will be required to write up the work procedures to implement it as a plant requirement.

Action 2 requires that a precautionary measure be taken during routine maintenance. It is assumed that four man-hours/plant would be required to implement this measure as a procedural change.

Action 3 requires red tagging the affected tie-breakers. This is assumed to require four man-hours/plant in order to complete the necessary paperwork (costs for red tags are negligible). One man-day/plant will also be necessary to red tag the appropriate tie-breakers.

Action 4 requires that additional QA procedures be incorporated. Two man-days/plant will be required.

No additional equipment or down-time is needed to implement the SIR. Thus, the labor required for SIR implementation at the operating LWRs is as outlined below.

Action	Labor (man-hr/plant) 71 "Affected" Plants
Design Review	6
#1	4
#2	4
#3	12
#4	16
	42

SIR OPERATION/MAINTENANCE

It is assumed that maintenance is performed and startup occurs once a year, during the annual outage for refueling. SIR operation/maintenance is needed only at the 32 affected LWRs.

For Action 2, one man-hour per plant-year is required, during every routine maintenance, to ensure that each tie breaker is racked out and red tagged.

For Action 4, four man-hours per plant-year are required to perform the additional QA procedures to confirm that all tie breakers are racked out and red tagged prior to each plant startup. Thus, the labor required for SIR operation and maintenance is five man-hours per plant-year.

REFERENCES (for Issue 49)

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.

Green, A., and A. Bourne. 1972. Reliability Technology. Wiley Interscience, New York.

Institute of Nuclear Power Operations (INPO). 1982. Review of NRC Report: Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report. NUREG/CR-2497, INPO 82-025, Atlanta, Georgia.

Minarick, J., and C. Kukielka. 1982. Precursors to Potential Severe Core Damage Accidents, 1969-1979: A Status Report. NUREG/CR-2497, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 51, Improved Reliability of Open Service Water Systems

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Fouling of safety-related service water systems (SWS) by either mud silt, corrosion products, or aquatic bivalves has led to plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation. The resolution involves recommending procedures to prevent and control SWS fouling.

AFFECTED PLANTS			
	BWR:	Operating = 17	Planned = 21
	PWR:	Operating = 31	Planned = 39

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 4.5E+04

OCCUPATIONAL DOSES:

SIR Implementation =	48
SIR Operation/Maintenance =	-8700
Total of Above =	-8600
Accident Avoidance =	240

COST RESULTS ($$10^6$)

INDUSTRY COSTS:

SIR Implementation =	77
SIR Operation/Maintenance =	-6400
Total of Above =	-6300
Accident Avoidance =	19

NRC COSTS:

SIR Development =	0.27
SIR Implementation Support =	0.49
SIR Operation/Maintenance Review =	0.25
Total of Above =	1.0

IMPROVED RELIABILITY OF OPEN SERVICE WATER SYSTEMS

ISSUE 51

1.0 SAFETY ISSUE DESCRIPTION

"Service water system fouling has been a subject of concern by the Offices of Nuclear Reactor Regulation (NRR), Inspection and Enforcement (IE), and Analysis and Evaluation of Operational Data (AEOD) during the past 18 months. A recent series of significant fouling by aquatic bivalves at operating plants has focused attention on this continuing industry problem," according to Eisenhut (1982).

The following examples illustrate the seriousness of the problem and describe bivalve fouling within service water systems (SWSs) of several affected plants.

1. Arkansas Nuclear One, Unit 2, was shut down when there was a failure to meet technical specification surveillance require ments for minimum service water flow rate through a containment fan cooler unit (CCU). Extensive plugging by corbicula was found. Additional biofouling was found in the supply piping to the CCUs and in the cooler inlet water boxes. Since strainers were intact, it was determined that clams were growing in the system (Camp Dresser and McKee 1982).

2. Inspection of the service water system at Pilgrim Nuclear Power station revealed an accumulation of mytilus in the reactor building closed cooling water heat exchanger (HX). A high differential pressure across the baffle plate due to the presence of these mussels had apparently deformed the baffle plates, allowing service water to bypass the HX tubes (see Michelson 1982).

3. Redundant residual heat removal (RHR) HXs at Brunswick Unit 1 were declared inoperable due to displaced divider plates resulting from a differential pressure buildup. This was a consequence of an unobserved accumulation of oysters and oyster shells (Eisenhut 1982).

4. San Onofre 1 reported problems in a component cooling water HX due to a buildup of barnacles during extended plant shutdown. The buildup was a consequence of suspending normal biofouling prevention methods (Eisenhut 1982).

PROPOSED RESOLUTION

As a consequence of such problems and due to the concern for operability of associated safety-related equipment, a Licensee Event Report (LER) review has been conducted, IE Bulletin 81-03 has been issued, and the results analyzed.

Technical support has been obtained to investigate biofouling problems and potential solutions.

A review of an Oak Ridge National Laboratory (ORNL) study, "Evaluation of Events involving Service Water Systems in Nuclear Plants," which used data from LERs between January 1979 and June 1981, revealed no generic design deficiencies among plants affected by biofouling. Subsequently, concentration has been shifted to the basic concern of fouling. "The AEOD report on the events of Arkansas and Brunswick concludes that improvements of surveillance and preventive maintenance programs at sites where bivalves are known to exist could significantly improve the reliability of the service water systems" (Eisenhut 1982). The Operating Reactors Assessment 8ranch (ORAB) has made several recommendations for improving SWS reliability.

Some of the significant findings and recommendations include:

1. Service water system fouling has been and will continue to pose a potential common mode failure for the redundant service water system trains. Further separation of trains or separation of safety versus nonsafety-related piping systems is not recommended due to the fact that the sources of fouling emanate from a common ultimate heat sink. Control strategies must be developed to mitigate this potential.

2. Fouling due to mud and silt buildups are a more serious concern than bivalves at many plants.

3. Improvements in service water system reliability can be accomplished at most plants through modifications in surveillance programs, increased monitoring of key parameters and improved preventive maintenance programs. The usual methods used to identify system fouling--namely normal maintenance, inservice testing, and testing required by the plant's technical specifications--have proved to be ineffective at many plants.

4. Plants should impose additional technical specifications on flow parameters of safety-related components.

5. The recommendations of the enclosed proposal [Enclosure 1[(a)]] can only be considered as general guidelines. The plant improvements that we envision will not be likely until utilities recognize the serious potential of service water system fouling and then take the appropriate steps (Eisenhut 1982).

(a) Enclosure 1 incorporates recommended procedures to prevent bivalve buildup. Methods include measuring flow rates in safety-related equipment and heat transfer coefficients at HXs, flushing techniques, and control strategies such as chlorination, visual inspection, and heat treatments.

The assumed safety issue resolution (SIR) involves upgrading prevention and detection techniques for fouling problems in SWSs in affected plants. This includes installation of equipment (e.g., strainers, chlorination units, and monitoring equipment), training of maintenance personnel, and implementing cleaning and flushing procedures.

AFFECTED PLANTS

Some 70 planned and operating PWRs and 38 planned or operating BWRs are assumed to be affected.

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.

PUBLIC RISK REDUCTION

It is assumed that public risk reduction will result from upgrading biofouling surveillance and prevention methods in affected plants (those that detect the existence of bivalves in the plant vicinity). The degree of upgrading required is dependent on existing methods in use. Because insuffi-ient information is available regarding the habitat requirement of these organisms and the raw cooling water system environment, it is difficult to predict the extent of flow blockage at specific plants. Therefore, flow blockage is based on data taken from responses to IE Bulletin 81-03 and information available within the NRC (Masnik 1982).

In applying the methodology (Andrews et al. 1983), it is assumed that fouling conditions anywhere within the SWS would increase the probability of failure of the entire system, the probability being based on the historical rate of shutdown at operating plants due to biofouling (see Attachment 1 to Public Risk Reduction Work Sheet). This systematic approach does not directly address the problems of fouling locations (i.e., problems seem to more realistically occur at HX divider plates, etc.).

OCCUPATIONAL DOSE

Additional radiation exposure will be accumulated by personnel during implementation of the SIR (i.e., installation of new surveillance and prevention instrumentation and equipment). Very small doses are expected for most of the SWS. Higher doses can be expected at RHR HX and other points serviced by the SWS. A dose reduction is expected over the lifetime of the affected plants through application of biofouling prevention methods.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Improved Reliability of Open Service Water Systems (51)

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

The resolution is intended for implementation at all operating and planned plants where existing or potential biofouling problems exist. The affected plants are those that have experienced the presence of Asiatic clams, blue mussels, American oysters or barnacles in the water bodies on which the plants are located. Most of the biofouling incidents have occurred at plants that presently have some form of prevention and detection methods (occurrences, in part, due to lax implementation or ineffective techniques). Therefore, all plants where existing or potential biofouling problems exist are considered affected under the assumption that SWS unavailability is equally probable in plants with or without biofouling detection and prevention techniques (see Hayes 1983).

		N	$\bar{T}$ (yr)
PWR:	Operating	31	27.7
	Planned	39	30.0
	Total	70	29.0
BWR:	Operating	17	25.2
	Planned	21	30.0
	Total	38	27.9

3. Plants Selected for Analysis:

Arkansas Nuclear One 1 - representative PWR

Grand Gulf 1 - representative BWR

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

4. Parameters Affected by SIR:

Based on the redefinition of ANO-1 parameters related to a potential loss of flow in open SWS, a common-cause parameter Z is defined as the

TABLE 1. (cont'd.)

affected parameter for this SIR. Z represents loss of SWS flow due to biofouling.[a]

5. Base-Case Values for Affected Parameters:

$$Z = 0.00445$$ [a]

6. Affected Accident Sequences and Base-Case Frequencies:

Sequence		Frequency (1/py)
B(1.2)D1 -	α (PWR-1)	2.87E-11
	γ (PWR-2)	1.44E-07
	β (PWR-5)	2.01E-09
	ε (PWR-7)	1.44E-07
B(1.2)D1C -	α (PWR-1)	3.36E-11
	γ (PWR-2)	1.68E-07
	β (PWR-4)	2.35E-09
	ε (PWR-6)	1.68E-07
B(4)H1 -	α (PWR-1)	1.69E-08
	γ (PWR-2)	8.45E-07
	β (PWR-5)	1.18E-08
	ε (PWR-7)	8.45E-07
T(DO1)LD_1YC -	α (PWR-1)	5.05E-11
	δ (PWR-2)	1.01E-07
	β (PWR-4)	3.54E-09
	ε (PWR-6)	4.04E-07
B(1.66)H_1 -	α (PWR-1)	1.38E-10
	γ (PWR-2)	6.90E-07
	β (PWR-5)	9.66E-09
	ε (PWR-7)	6.90E-07

(Note: Containment failure mode likelihoods come from Table 8.2 of Kolb et al. 1982).

7. Affected Release Categories and Base-Case Frequencies:

Category	Frequency (1/py)
PWR-1	1.72E-08
PWR-2	1.95E-06
PWR-4	5.89E-09

(a) See Attachment 1.

TABLE 1. (cont'd.)

Category	Frequency (1/py)
PWR-5	2.35E-08
PWR-6	5.72E-07
PWR-7	1.68E-06

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

$$\bar{F}_{PWR} = 4.2E\text{-}06/py$$

$$\bar{F}_{BWR} = 3.1E\text{-}06/py$$

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

$$W_{PWR} = 9.58 \text{ man-rem/py}$$

$$W_{BWR}^{(a)} = 24.0 \text{ man-rem/py}$$

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 biofouling failure of open SWS. Thus, the adjusted-case value of Z is effectively zero ($Z^* \ll Z$). Consequently, the adjusted-case, affected accident sequence frequencies, core-melt frequency, and public risk are effectively zero.[a]

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

$$\Delta\bar{F}_{PWR} = 4.2E\text{-}06/py$$

$$\Delta\bar{F}_{BWR}^{(a)} = 3.1E\text{-}06/py$$

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

$$\Delta W_{PWR} = 9.58 \text{ man-rem/py}$$

$$\Delta W_{BWR}^{(a)} = 2.40 \text{ man-rem/py}$$

(a) See Attachment 1.

TABLE 1. (cont'd.)

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
4.5E+04	1.3E+06	0

ATTACHMENT 1 (to Table 1)

All parameters related to potential loss of flow in an open SWS are redefined to include a common-cause failure Z which represents loss of SWS flow due to biofouling. For the representative plant (Arkansas Nuclear One 1 PWR [Kolb et al. 1982]), the redefined parameters are as follows:

$$\text{LF-SWS-S1} = (\text{LF-SWS-S1})_o + Z$$

$$\text{LF-SWS-S2} = (\text{LF-SWS-S2})_o + Z$$

$$\text{LF-SWS-S5} = (\text{LF-SWS-S5})_o + Z$$

$$\text{LF-SWS-S14} = (\text{LF-SWS-S14})_o + Z$$

$$\text{LF-SWS-S82} = (\text{LF-SWS-S82})_o + Z$$

$$\text{LF-SWS-S83} = (\text{LF-SWS-S83})_o + Z$$

$$\text{LF-SWS-VCH4A} = (\text{LF-SWS-VCH4A})_o + Z$$

$$\text{LF-SWS-VCH4B} = (\text{LF-SWS-VCH4B})_o + Z$$

where the subscripted terms represent the original parameters. The addition of common-cause factor Z is necessary because, as originally defined, these parameters did not include the biofouling failure possibility.

All minimal cut sets containing the identified parameters are modified by replacing each term by the above redefinition. In effect, this adds a new minimal cut set for each replacement. As an example:

Original Cut Set (ANO-1)

$$\text{B(1.2)} \cdot \text{LF-HP1-H14} \cdot \text{LF-SWS-S2}$$

Substitution

$$\text{B(1.2)} \cdot \text{LF-HP1-H14} \cdot [(\text{LF-SWS-S2})o + Z] =$$
$$[\text{B(1.2)} \cdot \text{LF-HP1-H14} \cdot (\text{LF-SWS-S2})o]$$
$$+ [\text{B(1.2)} \cdot \text{LF-HP1-H14} \cdot Z]$$

New Cut Set

$$\text{B(1.2)} \cdot \text{LF-HP1-H14} \cdot Z$$

Note: Since $(\text{LF-SWS-S2})_o$ is the original value for LF-SWS-S2, the cut set $\text{B(1.2)} \cdot \text{LF-HP1-H14} \cdot (\text{LF-SWS-S2})_o$ is the original cut set.

As a result of the aforementioned redefinition, the affected minimal cut sets and accident sequences are as follows:

Affected Sequence	Affected Minimal Cut Sets[(a)]
ANO-1:	
$B(1.2)D_1$	B(1.2) · LF-HPI-H14 · Z · [0.23]
$B(1.2)D_1C$	B(1.2) · LPI1407A-VCC-LF · Z · [0.23]
	B(1.2) · LPI1408B-VCC-LF · Z · [0.23]
$B(4)H_1$	B(4) · Z · [1]
$T_1(D01)LD_1YC$	T(D01) · Z · LF-EFS-E4 · [0.25]
	T(D01) · Z · LF-EFC-ACBD4 · [0.05]
	T(D01) · Z · LF-EFS-E29 · [0.22]
	T(D01) · Z · LF-EFC-BB7B1CM · [0.05]
	T(D01) · Z · LF-EFC-D1D2CM · [0.05]
	T(D01) · Z · LF-EFC-VCD2 · [0.05]
$B(1.66)H_1$	B(1.66) · Z · [1]

To estimate the failure probability for Z, it is necessary to consider available historical data. These data indicate seven observed failures of both redundant service water trains at Arkansas Nuclear One (2), Brunswick (2), Pilgrim, Browns Ferry, and Millstone. An additional 19 biofouling occurrences have been recorded in these systems, but not in both trains simultaneously (Hayes 1983). Given the number of affected plants and their average remaining lives from Step 2 of Table 1, a conservative estimate of the SWS failure rate due to biofouling can be derived by assuming that the above failures represent SWS failures. Z is estimated as follows.

(a) The multiplicative factors in brackets [] given with each ANO-1 cut set are the probability of nonrecovery for the recoverable minimal cut set elements, if any, in sequences containing the A term. These probabilities of nonrecovery are found in Appendix C, section C.2 of original ANO-1 assessment (Kolb et al. 1982).

The average number of operating years per plant for the 71 operating plants (Table C.2 of Andrews et al. 1983), is 580/71 = 8.17 yr.

The failure probability due to biofouling based on historical data, the average number of operating years per plant and the 48 affected operating plants is

$$\lambda = 7 \text{ failures}/[(8.17 \text{ yr})(48 \text{ plants})] = 0.0178/\text{py}.$$

To convert this failure rate into unavailability (failure probability) for parameter Z, estimates of SWS downtime due to biofouling-related failures are needed. A mean value of three months (based on ANO, Brunswick, and Pilgrim occurrences) is the estimated downtime due to fouled SWS (Hayes 1983). Denoting this downtime as t, one obtains the following estimate for the unavailability of SWS due to biofouling:

$$Z = \lambda t = (0.0178/\text{py})(3 \text{ mo})/(12 \text{ mo/yr}) = 0.00445.$$

This is taken as the base-case value for the parameter Z. Substituting the above values for Z in the affected minimal cut sets and using original values for the remainder of the cut set elements yield the following base-case frequencies for the affected accident sequences.

Affected Sequence	Base-Case Frequency(1/py)
ANO-1:	
$B(1.2)D_1$	2.87E-07
$B(1.2)D_1C$	3.36E-07
$B(4)H_1$	1.69E-06
$T_1(DO1)LD_1YC$	5.05E-07
$B(1.66)H_1$	1.38E-06
	4.2E-06

Thus, the base-case affected core-melt frequency is 4.2E-06. This value is for PWRs only but is assumed to apply to both operating and planned affected PWRs. BWR values are scaled from this, as discussed at the end of this attachment.

Issue resolution is assumed to effectively eliminate the problem of bio fouling failure of open SWS. Thus, the adjusted-case value for parameter Z is effectively zero; i.e., $Z^* \ll Z$. The affected accident sequence frequencies, core-melt frequency, and public risk are effectively zero as a consequence.

The reliability studies for Arkansas Nuclear One (1) (Kolb et al. 1982) and Grand Gulf 1 (Hatch et al. 1981) give total core-melt frequencies (F_o) of 5.0E-05/py and 3.7E-05/py, respectively, for these plants. Using the original release category frequencies and the public dose factors (Appendix D of Andrews et al. 1983), one obtains total public risks (W_o) of 100 man-rem/py and 250 man-rem/py, respectively, for Arkansas Nuclear One 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 Arkansas Nuclear One to Grand Gulf, the following are assumed:

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

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

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

$$\bar{F}_{BWR} = 0.74\ \bar{F}_{PWR}$$

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

$$W_{BWR} = 2.5\ W_{PWR}$$

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

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Improved Reliability of Open Service Water Systems (51)

2. Affected Plants (N):

		N
PWR:	Operating	31
	Planned	39
	Total	70
BWR:	Operating	17
	Planned	21
	Total	38

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

		$\bar{T}$ (yr)
PWR	31 Operating	27.7
	39 Planned	30.0
	All 70	29.0
BWR	17 Operating	25.2
	21 Planned	30.0
	All 38	27.9

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

$$\Delta(\bar{F}D_R)_{PWR} = (19{,}900 \text{ man-rem})(4.2E\text{-}06/\text{py}) = 8.36E\text{-}02 \text{ man-rem/py}$$

$$\Delta(\bar{F}D_R)_{BWR} = (19{,}900 \text{ man-rem})(3.1E\text{-}06/\text{py}) = 6.17E\text{-}02 \text{ man-rem/py}$$

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.4E+02	1.4E+03	0

TABLE 2. (cont'd.)

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

It is assumed that installation of monitoring equipment would take approximately 10 man-wk/plant. This assumes that a majority of the maintenance force is assigned to the task.

(10 man-wk/plant)(40 man-hr/man-wk) = 400 man-hr/plant

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

The average assumed radiation exposure is 2.5 mR/hr for installation of equipment in the intake systems, SWS piping, and associated individual components (e.g., high-pressure coolant injection system [HPCIS], RHR HX).

D = (400 man-hr/plant)(0.0025 R/hr) = 1 man-rem/plant

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

(48 plants)(1 man-rem/plant) = 48 man-rem

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

A. Unresolved case: It is assumed that without detection devices, a 2-month shutdown requiring cleanup and replacement of equipment would occur every 3 years. It is estimated that 4000 man-hr would be used in this effort. The effort would include such tasks as performing chemical cleaning tasks, mechanical cleaning, replacement of equipment that cannot be cleaned, and diving support to clean intake structures.

B. Resolved case: It is assumed that equipment installed requires monitoring (5 man-hr/mo) and that cleaning, backflushing, and some replacement of equipment are still necessary during scheduled outages. This latter time is estimated at 150 man-hr/py.

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

Assuming an average radiation exposure at 2.5 mR/hr:

A. Unresolved case:
(4000 man-hr/3 py)(0.0025 R/hr) = 3.3 man-rem/py

B. Resolved case:
[(5 man-hr/mo)(12 mo/py) + 150 man-hr/py](0.0025 R/hr)
= 0.53 man-rem/py

TABLE 2. (cont'd.)

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

Occupational dose increase for SIR operation and maintenance:

D_O = 0.53 - 3.3 = -2.8 man-rem/py

(Negative sign indicates reduction.)

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

$N\bar{T}D_O$ = [70(29.0 yr) + 38(27.9 yr)](-2.8 man-rem/py) = -8650 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
-8.6E+03	-2.9E+03	-2.6E+04

3.0 SAFETY ISSUE COSTS

Best estimates are used for labor time and equipment costs needed for the issue resolution. In most cases, cost estimates have been obtained from utilities that have implemented or are currently implementing prevention and detection devices for biofouling. Results of industry and NRC cost estimates are presented in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Improved Reliability of Open Service Water Systems (51)

2. Affected Plants (N):

		N
PWR:	Operating	31
	Planned	39
	Total	70

TABLE 3. (cont'd.)

2. Affected Plants (N): (cont'd.)

BWR:	Operating	17
	Planned	21
	Total	38

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

		$\bar{T}$ (yr)
PWR:	31 Operating	27.7
	39 Planned	30.0
	All 70	29.0
BWR:	17 Operating	25.2
	21 Planned	30.0
	All 38	27.9

Industry Costs (Steps 4 through 12)

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

$$\Delta(\bar{F}A)_{PWR} = (\$1.65E+09)(4.2E-06/py) = \$6.93E+03/py$$

$$\Delta(\bar{F}A)_{BWR} = (\$1.65E+09)(3.1E-06/py) = \$5.12E+03/py$$

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

Best Estimate	Upper Bound	Lower Bound
\$1.9E+07	\$1.1E+08	0

6-7. Steps Related to Per-Plant Industry Cost for SIR Implementation (I):

SIR implementation will most likely be different for every plant. Because of the unique problems found at each plant, technical support is most likely to be required (e.g., biologists, chemists and engineers). Installation of equipment will also vary with type and costs. The amounts below were obtained from conversations with utility staff and are believed representative of implementation costs.

TABLE 3. (cont'd.)

6-7. Steps Related to Per-Plant Industry Cost for SIR Implementation (I): (cont'd.)

Plants with fouling detection and prevention equipment and methods have been grouped with those that currently have no techniques. Because equipment or methods seem to be ineffective in these equipped plants, it is assumed that costs for equipment already present can be transferred to personnel training or upgrading of equipment. Labor hours are included in figures obtained.

• technical support	= $100K
• installation of strainers for individual components (e.g., HPCIS)	= $500K
• chlorination units, including analyzers and retrofitting equipment	= $ 50K
• monitoring equipment (e.g., flow rate)	= $ 50K
• mechanical cleaning access ways for coolers where flushing is ineffective	= $ 8K
• labor for implementation not included above (2.5 man-wk)($2270/man-wk)	= $ 6K
I	= $714K/plant

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

NI = ($7.14E+05/plant)(108) = $7.71E+07

9-10. Steps Related to Per-Plant Industry Cost for SIR Operation and Maintenance:

A. Unresolved Case

It is assumed that, on the average, one calendar-month of lost revenue would result every 5 years. This is time beyond normal outages. Maintenance required and equipment costs are projected as examples (5 year estimates):

• replacement power (30 days)($3.0E+05/day)	= $9.0M
• chemical cleaning of system excluding coolers	= $0.70M
• replacement piping and installation of critical control units	= $0.95M

TABLE 3. (cont'd.)

9-10. Steps Related to Per-Plant Industry Cost for SIR Operation and Maintenance: (cont'd.)

- divers and support team to clean intake structures = \$0.030M
- labor not included above (12.5 man-wk)(\$2270/man-wk) = \$0.028M

\$10.71M/5 py,

or \$2.14M/py

B. Resolved Case

- cleaning cost (\$200K/5 py) =\$40K
- equipment (includes chorine at 50 tons/yr) =\$20K
- divers and support team (\$15K/5 py) = \$3.0K
- labor not included above [(3.75 man-wk)(\$2270/man-wk) from Step 9 of Table 2] = \$8.5K

\$71.5K/py

I_o = \$7.15E+4 - \$2.14E+6 = -\$2.07E+6/py (savings)

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

$(N\bar{T}I_o)$ = (-\$2.07E+06/py)[70(29.0 yr) + 38(27.9 yr)] = -\$6.39E+09

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
-\$6.3E+09	-3.1E+09	-9.5E+09

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

The NRC cost to review and develop the generic SIR is estimated at 10 man-wk.

TABLE 3. (cont'd.)

Technical support with the objective being to "provide the NRC with recommendations for plant surveillance and testing procedures for early detection of fouling of service water systems by freshwater and marine organisms" has been undertaken by PNL (Hayes 1982). Total estimated project costs for future work are as follows:

FY83 $191K

FY84 $ 60K

Total $251K

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

$$C_D = (10 \text{ man-wk})(\$2270/\text{man-wk}) + \$2.51E+05 = \$2.74E+05$$

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

It is anticipated that an average of 2 man-wk for affected plants will be required to account for differences among plant designs and applicable surveillance methods.

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

$$C = (2 \text{ man-wk/plant})(\$2270/\text{man-wk}) = \$4.54E+03/\text{plant}$$

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

$$NC = (108)(\$4.54E+03/\text{plant}) = \$4.90E+05$$

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

It is anticipated that a monitoring program will be employed to track the results of implementing new surveillance or prevention methods for biofouling in affected plants. In addition, the changing status of affected versus nonaffected plants will continue to be assessed. It is estimated that this program will require approximately 0.2 man-wk/py for each affected plant over the next 5 years.

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

$$C_O = (0.2 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$454/\text{py}$$

TABLE 3. (cont'd.)

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

The effective average plant life for NRC review of SIR operation and maintenance is only 5 yr (see Step 18). Thus,

$$\bar{NTC}_O = (108)(5\ yr)(\$454/py) = \$2.45E+05$$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$1.0E+06	\$1.3E+06	\$6.9E+05

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 53, BWR Flow Blockage

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Worst-case analysis of flow blockage to a single BWR fuel assembly shows only local fuel and clad damage potential, with no possiblity of a core melt. A resolution has been proposed that involves design changes in fuel assembly lower tie plates to maintain flow to an otherwise blocked channel.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 0

OCCUPATIONAL DOSES:

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

COST RESULTS ($1E+6)

INDUSTRY COSTS:

SIR Implementation =	1.5
SIR Operation/Maintenance =	68
Total of Above =	69
Accident Avoidance =	0

NRC COSTS:

SIR Development =	0.027
SIR Implementation Support =	0.10
SIR Operation/Maintenance Review =	0
Total of Above =	0.13

BWR FLOW BLOCKAGE

ISSUE 53

1.0 SAFETY ISSUE DESCRIPTION

NEDO-10174 Rev. 1, "Consequences of a Postulated Flow Blockage Incident In a Boiling Water Reactor," was prepared by General Electric Co. in response to an Advisory Committee on Reactor Safeguards (ACRS) concern on the potential of melting a portion of BWR fuel assembly due to inlet orifice flow blockage. Rev. 1 was prepared when applicability of NEDO-10174 to the 8x8 fuel design was questioned.

The main conclusions reached in NEDO-10174 Rev. 1 are as follows:

1. The only mechanism capable of causing a major flow blockage is that induced by a foreign object.

2. Fragmentation, crudding or fuel swelling cannot cause major flow blockages.

3. A fuel assembly is capable of withstanding very severe blockages before losing adequate cooling.

4. For orifice blockage greater than 98%, fuel and cladding melt are expected to occur. However, this will not result in failure propagation to adjacent assemblies, local high-pressure production or offsite doses in excess of a small fraction of 10 CFR 100 guidelines. For this worst-case event, no action is required of the reactor ECCS. However, the reactor must be scrammed by the main steam line radiation monitor.

5. For orifice blockages between 95% and 98%, clad melting is expected, but fuel melting is not calculated to occur. For this case, the consequences are less severe than in #4 above.

6. For orifice blockages between 79% and 95%, boiling transition and attendant cladding heatup are calculated to occur. No clad nor fuel melting is calculated. However, cladding failure is not precluded. The off-gas radiation monitor will provide an alarm to the reactor generator if fission product releases are significant.

7. For orifice blockages less than 79%, nucleate boiling is maintained. Therefore, the fuel and cladding are unaffected.

This report has not been reviewed nor the conclusions confirmed by experience or tests. However, for this analysis it is assumed that the report conclusions are valid and, in the worst cases (79% or greater blockage), that the only release category involved would be BWR-5 (a non-core melt category). Since no core melt will result from these worst-case flow blockages, no public risk reduction analysis is performed for this issue.

SAFETY ISSUE RESOLUTION

For the safety issue resolution (SIR), it is assumed that a fuel element design change is implemented that consists of holes added to the lower tie date of the fuel assemblies. The holes are closed with an internal flapper valve that is flow- and pressure-operated (no springs). Under normal reactor operation, the holes would remain closed. If a high degree of blockage occurred during operation, the holes would open, allowing core bypass coolant to flow into the blocked channel.

AFFECTED PLANTS

All 44 BWRs are assumed to be affected by issue resolution.

2.0 SAFETY ISSUE RISK AND DOSE

As discussed previously, no reduction in public risk is estimated for this issue since it does not deal with core melt. Furthermore, the SIR is assumed to involve design changes on new fuel assemblies only, not on existing ones. Since all such changes would be implemented prior to insertion of the assemblies into the core, no occupational dose should result from SIR implementation. Similarly, no maintenance or inspection of these new fuel elements will presumably be conducted once they have been placed inside the core. Thus, no occupational dose should result from SIR operation/maintenance. Since no occupational dose is perceived to result from SIR, the Occupational Dose Work Sheet is omitted from this report.

3.0 SAFETY ISSUE COSTS

The industry and NRC costs due to SIR are estimated this section. Analysis results are summarized in Table 1.

TABLE 1. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   BWR Flow Blockage (53)

TABLE 1. (cont'd.)

2. Affected Plants (N):

All BWRs

	N
Operating	24
Planned	20
All BWRS	44

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

	$\bar{T}$ (yr)
24 Operating BWRs	25.2
20 Planned BWRs	30
44 Total	27.4

Industry Costs (Steps 4 through 12)

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

Since this issue does not affect core-melt frequency, these steps are omitted.

6. Per-Plant Industry Resources for SIR Implementation:

Costs are estimated directly in Step 8.

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

Costs are estimated directly in Step 8.

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

The industry cost associated with SIR is assumed to be twofold:

1. Costs generic to BWRs as a whole for design change studies, tests, etc.

2. Costs specific to each BWR for fabricating modified fuel assemblies.

The latter are considered to be SIR operation/maintenance costs and are presented in Step 10. The former are viewed as SIR implementation costs for BWRs and are assumed to amount to $1.5E+06.

TABLE 1. (cont'd.)

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

Cost is estimated directly in the next step.

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

Each BWR must pay an incremental cost on each new fuel assembly for the additional holes and flapper valves in the lower tie plate as assumed for SIR. On the average, a plant replaces 1/3 of its core during annual refueling outages. Assuming 600 and 750 fuel assemblies per core for operating and planned BWRs, respectively (based on BWR design information), one obtains the following refueling rates:

Operating BWR:	200 fuel assemblies/py
Planned BWR:	250 fuel assemblies/py

Only the incremental cost due to SIR-related design changes in the assemblies is credited here. Assuming that this amounts to \$250/fuel assembly, the operation/maintenance costs (interpreted as refueling costs) are as follows:

I_O (operating BWR) = (200)(\$250) = \$5.0E+04/py
I_O (planned BWR) = (250)(\$250) = \$6.25E+04/py

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

$N\bar{T}I_O$ = 24(25.2)(\$5.0E+04) + 20(30)(\$6.25E+04) = \$6.77E+07

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$6.9E+07	\$1.0E+08	\$3.5E+07

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC Staff Labor = 12 man-wk

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

C_D = \$2.7E+04

TABLE 1. (cont'd.)

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

NRC Staff Labor = 1 man-wk/plant

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

C = $2270/plant

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

NC = (44)($2270) = $1.0E+05

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

No additional NRC monitoring is foreseen due to SIR.

Thus, C_o = 0.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$1.3E+05	$1.8E+05	$7.5E+04

REFERENCES (for Issue 53)

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

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 57, Effects of Fire Protection System Actuation on Safety-Related Equipment/Systems

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Recent operating reactor events show that safety-related equipment subjected to water spray from fire protection systems can be rendered inoperable. A two-part resolution is proposed: 1) issuance of a IE Information Notice to all licensees (requiring no action) and 2) implementation of administrative changes and hardware modifications based upon plant-specific Fire Protection Program reevaluations. Results given below are for part two.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	1.3E+03
OCCUPATIONAL DOSES:	
SIR Implementation =	710
SIR Operation/Maintenance =	560
Total of Above =	1.3E+03
Accident-Avoidance =	7.9

COST RESULTS ($$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	120
SIR Operation/Maintenance =	17
Total of Above =	130
Accident-Avoidance =	0.66
NRC COSTS:	
SIR Development =	0.27
SIR Implementation Support =	7.2
SIR Operation/Maintenance Review =	4.2
Total of Above =	12

EFFECTS OF FIRE PROTECTION SYSTEM ACTUATION

ON SAFETY-RELATED EQUIPMENT/SYSTEMS

SAFETY ISSUE 57

1.0 SAFETY ISSUE DESCRIPTION

This issue is concerned with recent fire protection system (FPS) actuations at operating nuclear plants. As discussed at the Operating Reactor Event meeting held on January 7, 1982 (Lainas 1982), it was concluded that safety-related equipment subjected to water spray from an FPS could be rendered inoperable. The events also indicated that spurious actuation of an FPS can be initiated by operator error, steam, high humidity or maintenance activities in the vicinity of the FPS detectors. Other events also exemplify that interactions of the FPS with other systems (e.g., ventilation and diesel fuel oil) have not been adequately considered. At the meeting, the NRC Division of Engineering/Inspection and Enforcement (DE/IE) was assigned the responsibility to review recent FPS actuations and consider development of an Information Notice. Furthermore, the Division of Engineering/Office of Nuclear Reactor Regulation (DE/NRR) was to review the events and consider the need for modifications to requirements[a] or review procedures for Fire Protection Systems (FPSs). The DE study is ongoing and will consider the Office for Analysis and Evaluation of Operational Data (AEOD) concern for all types of FPSs (e.g., water, halides, carbon dioxide and other chemicals).

In summary, the NRC should have confidence that all safety-related and essential support equipment located in areas where fire protection spray systems are provided will perform the intended function during and following the actuation of the FPSs.

PROPOSED RESOLUTION

For purposes of this analysis, the proposed resolution to this safety issue is assumed to consist of two parts.

Based upon a review of NRC guidelines regarding system interactions between fire protection features and safety systems as well as recent Licensee Event Reports (LERs) and NRC review experience regarding such interactions, the NRC found that, if their guidelines are properly implemented, such interactions should be minimized. However, recent LERs indicate that the guidelines have not been properly implemented at some plants (Vollmer 1982). Therefore, part one of this resolution is assumed to consist of issuance of an IE Information Notice. The purpose of the Information Notice is to inform

(a) In the context of the resolution postulated for this issue, it is assumed that requirements refer to any actions that impose burdens on licensees. Furthermore, the term "licensee" used here is assumed to apply to applicants for CPs and OLs.

licensees and applicants with a construction permit of potential interactions for FPSs with other plant systems which can result in reducing the availability of safety-related systems. Currently, a draft of the IE Information Notice is being circulated within NRC divisional components for review.

Potential interactions between FPSs and other systems that affect the operation of safety-related systems need to be thoroughly understood. Safety-related equipment, not damaged by a fire itself, should be designed and qualified to perform its intended function during and following an FPS actuation.

NRC regulations and guidelines recognize that fire protection features can have adverse effects on structures, systems, and components important to safety. If these guidelines are implemented properly, such adverse effects as well as adverse interactions between fire protection features and safety systems should be precluded by design. As mentioned previously, recent LERs indicate that such interactions are occurring and that guidelines are not being implemented properly.

Therefore, it is assumed for part two of this resolution that the multiple concerns associated with the adequacy of design and qualifications of safety-related equipment and systems located in areas where fire protection is provided should be reevaluated by all licensees. The methodology postulated to accomplish this plant-specific task is presented in Attachment 1, together with the chronology of anticipated events, responsible entities, activities, end-products/results, and time and cost estimates, associated with each activity. The latter two components are used subsequently in Tables 2 and 3 of this issue for those steps where that information is applicable. It should be recognized, however, that at present the NRC does not plan to require licensees to perform an extensive reevaluation of LWR Fire Protection Programs as outlined herein.

For purposes of this analysis, however, it is assumed that the plant-specific SIR implementation (i.e., procedural changes and hardware fixes delineated by the reevaluation) would decrease the unavailabilities of the representative equipment associated with the common-cause failures described in Section 2.0 by a factor of four.

AFFECTED PLANTS

All plants are assumed to be affected for part two of this resolution, with SIR implementation occurring in 1986. Three age-groups of plants are identified below.

Age Group	Plants	N	$\overline{T}$(yr)
1	Backfit PWRs	47	27.7
	Backfit BWRs	24	25.2
2	Forward-fit PWRs commencing operation by 1986	24	27.7
	Forward-fit BWRs commencing operation by 1986	13	27.5
3	Forward-fit PWRs commencing operation after 1985	19	30
	Forward-fit BWRs commencing operation after 1985	7	30

ATTACHMENT 1 (For Section 1.0)

Step	Responsible Entity	Postulated Activity	Postulated End-Product and/or Results	Estimated Time/Cost
1	NRC	Continuing AEOD trend analysis (generic)	• Study with resultant NUREG issued based on trend analysis[a] • Issue IE Bulletin directing a Fire Protection all licensees	≥ 1 year (starting Jan. 1983) / $200,000 (NUREG) $50,000 (Continuing trend analysis) 0.17 year / $16,700
2	Industry	Plant-specific study done in response to IE Bulletin with guidance from NUREG	• Plant-specific reevaluation of the likelihood of future inadvertent FPS actuations and potential consequences; desirable changes identified and initial cost estimates completed. • Reevaluation submitted to NRC for review, comment, and approval.	≥ 1 year per plant (all of 1984) / $200,000 (review) per plant $100,000 (report) per plant
3	NRC	Review and approval of licensee's reevaluation	• Licensee notified to proceed after 4-month per plant review period	3 to 6 man-months per plant;[b] 4 man-months per plant average is assumed. / $33,300 per plant
4	Industry	Implement approved changes (starting in Jan. 1986)	• Procedural and hardware fixes completed in 1986[c] • Notify NRC upon completion	≤ 1 year / Procedural changes (including license fees) @ $125,800 per plant[d] Hardware fixes (e.g., spray shields) @ $150,000 per plant Labor @ $300,000 per plant —

ATTACHMENT 1 (cont'd.)

Step	Responsible Entity	Postulated Activity	Postulated End-Product and/or Results	Estimated Time/Cost
5	NRC	Onsite inspection	• NRC inspection (approval is assumed)	~ 4 man-month per plant[e] / $33,300 per plant
6	Industry	SIR operation and maintenance	• As needs dictate.	2 man-wk per plant-year / $4540 per plant-year
7	NRC	Annual inspection of SIR operation and maintenance	• As needs dictate.	0.5 man-wk per plant-year / $1140 per plant-year

(a) For purposes of this issue, it is assumed that no new NRC regulations are called for as a result of the continuing AEOD trend analysis. The development of additional data and subsequent analysis are documented formally in a NUREG report, which is assumed to provide the technical basis for an IE Bulletin. The bulletin is the formal regulatory mechanism directing each licensee to perform a reevaluation of his fire protection program.

(b) Personal communication with R. Ferguson, NRC, November 29, 1982.

(c) The review and modification of procedures are aimed at decreasing the frequency of inadvertent actuation of the FPS. The review portion is directed to the review of design of the FPS relating to systems interaction, with subsequent procedural fixes aimed at improvements in administration (i.e., those measures identified to mitigate potential for inadvertent actuation of the FPS). No occupational dose is associated with procedural modifications. On the other hand, hardware fixes will incur some occupational dose. Hardware fixes are intended to minimize the challenge to the systems and to minimize the likelihood that the plant operators will be faced with bringing the plant to a shutdown condition with potentially malfunctioning equipment.

(d) Includes Class V license amendment fee of $25,800 per 10 CFR 170.22.

(e) For purposes of this analysis, it is assumed that: 1) either the NRC onsite inspector or the regional inspector has been following some of this work as part of his regular duties (no cost assigned), and 2) the onsite inspection itself is conducted by a team of 4 inspectors. The 4 inspectors are assumed to spend 2 weeks on preparations for each plant inspection, 1 week onsite, and 1 week on followup duties, for a total of 4 man-months per plant inspection.

2.0 SAFETY ISSUE RISK AND DOSE

There is no risk reduction associated with part one of the proposed resolution to this safety issue. The estimates of public risk reduction and occupational dose associated with part two are summarized in Tables 1 and 2, respectively. These estimates are based upon implementation of hardware and procedural fixes postulated for the SIR occurring in 1986 for all LWRs.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue: Effects of Fire Protection System Actuation on Safety-Related Equipment/Systems (No. 57)

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

All plants are assumed to be affected, with SIR implementation occurring in 1986. Three age-groups of plants are identified.

Age Group	Plants	N	$\overline{T}$(yr)
1	Backfit PWRs	47	27.7
	Backfit BWRs	24	25.2
2	Forward-fit PWRs commencing operation by 1986	24	27.7
	Forward-fit BWRs commencing operation by 1986	13	27.5
3	Forward-fit PWRs commencing operation after 1985	19	30
	Forward-fit BWRs commencing operation after 1985	7	30

For all 90 PWRs and 44 BWRs, the average remaining lives for this SIR are 28.2 yr and 26.6 yr, respectively.

TABLE 1. (cont'd.)

3. Plants Selected for Analysis:

Oconee 3 - representative PWR
Grand Gulf 1 - representative BWR

(The analysis is conducted for Oconee 3, and the results are scaled for Grand Gulf 1, as discussed at the end of Attachment 2).

4. Parameters Affected by SIR:

Two new parameters are defined for this SIR:[a]

ZD = common-cause failure of onsite emergency AC power (via the diesel generators [DGs]) due to inadvertent FPS actuation

ZH = common-cause failure of high pressure coolant injection (HPCI) due to inadvertent FPS actuation.

5. Base-Case Values for Affected Parameters:

$$\left.\begin{matrix} ZD \\ ZH \end{matrix}\right\} = 2.2E\text{-}5^{(a)}$$

6. Affected Accident Sequences and Base-Case Frequencies:

$$T_2MLU - \begin{cases} \gamma\ (PWR\text{-}3) & = 8.8E\text{-}10/py^* \\ \beta\ (PWR\text{-}5) & = 1.3E\text{-}11/py^* \\ \epsilon\ (PWR\text{-}7) & = 8.8E\text{-}10/py^* \end{cases}$$

$$T_1MLU \begin{cases} \gamma\ (PWR\text{-}3) & = 1.5E\text{-}09/py \\ \beta\ (PWR\text{-}5) & = 2.1E\text{-}11/py \\ \epsilon\ (PWR\text{-}7) & = 1.5E\text{-}09/py \end{cases}$$

$$T_1(B_3)MLU - \begin{cases} \gamma\ (PWR\text{-}3) & = 4.4E\text{-}08/py \\ \beta\ (PWR\text{-}5) & = 6.4E\text{-}10/py \\ \epsilon\ (PWR\text{-}7) & = 4.4E\text{-}08/py \end{cases}$$

$$T_2KMU - \begin{cases} \gamma\ (PWR\text{-}3) & = 8.6E\text{-}10/py \\ \beta\ (PWR\text{-}5) & = 1.3E\text{-}11/py \\ \epsilon\ (PWR\text{-}7) & = 8.6E\text{-}10/py \end{cases}$$

$$S_2D - \begin{cases} a\ (PWR\text{-}1) & = 8.8E\text{-}11/py \\ \gamma\ (PWR\text{-}3) & = 1.8E\text{-}09/py \\ \beta\ (PWR\text{-}5) & = 6.4E\text{-}11/py \\ \epsilon\ (PWR\text{-}7) & = 7.0E\text{-}09/py \end{cases}$$

(a) See Attachment 2.

TABLE 1. (cont'd.)

$$S_3D - \begin{cases} \gamma \text{ (PWR-3)} = 1.4E\text{-}08/py \\ \beta \text{ (PWR-5)} = 2.1E\text{-}10/py \\ \epsilon \text{ (PWR-7)} = 1.4E\text{-}08/py \end{cases}$$

$$T_2MQD - \begin{cases} \gamma \text{ (PWR-3)} = 1.7E\text{-}8/py \\ \beta \text{ (PWR-5)} = 2.4E\text{-}10/py \\ \epsilon \text{ (PWR-7)} = 1.7E\text{-}8/py \end{cases}$$

*Non-dominant minimal cut sets are also assumed to be affected in direct proportion to the effect on dominant minimal cut sets.

7. Affected Release Categories and Base-Case Frequencies:

PWR-1 = 8.8E-11/py
PWR-3 = 8.0E-08/py
PWR-5 = 1.2E-09/py
PWR-7 = 8.5E-08/py

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

$\bar{F}_{PWR}$ = 1.7E-07/py $\bar{F}_{BWR}$ = 7.5E-08/py[(a)]

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

W_{PWR} = 0.43 man-rem/py W_{BWR} = 0.52 man-rem/py[(a)]

10. Adjusted-Case Values for Affected Parameters:

The unavailabilities of the affected parameters ZD and ZH are assumed to decrease by a factor of four due to SIR.

$$\left.\begin{matrix} ZD \\ ZH \end{matrix}\right\} = 5.5E\text{-}06$$

11. Affected Accident Sequences and Adjusted-Case Frequencies:

$$T_2MLU - \begin{cases} \gamma = 2.2E\text{-}10/py^* \\ \beta = 3.2E\text{-}12/py^* \\ \epsilon = 2.2E\text{-}10/py^* \end{cases}$$

$$T_1MLU - \begin{cases} \gamma = 3.7E\text{-}10/py \\ \beta = 5.4E\text{-}12/py \\ \epsilon = 3.7E\text{-}10/py \end{cases}$$

(a) See Attachment 2.

TABLE 1. (cont'd.)

$$T_1(B_3)MLU - \begin{cases} \gamma = 1.1E\text{-}08/py \\ \beta = 1.6E\text{-}10/py \\ \epsilon = 1.1E\text{-}08/py \end{cases}$$

$$T_2KMU - \begin{cases} \gamma = 2.1E\text{-}10/py \\ \beta = 3.1E\text{-}12/py \\ \epsilon = 2.1E\text{-}10/py \end{cases}$$

$$S_2D - \begin{cases} \alpha = 2.2E\text{-}11/py \\ \gamma = 4.4E\text{-}10/py \\ \beta = 1.6E\text{-}11/py \\ \epsilon = 1.8E\text{-}09/py \end{cases}$$

$$S_3D - \begin{cases} \gamma = 3.6E\text{-}09/py \\ \beta = 5.2E\text{-}11/py \\ \epsilon = 3.6E\text{-}09/py \end{cases}$$

$$T_2MQD - \begin{cases} \gamma = 4.1E\text{-}09/py \\ \beta = 6.0E\text{-}11/py \\ \epsilon = 4.1E\text{-}09/py \end{cases}$$

*Non-dominant contribution included as in Step 6.

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 2.2E-11/py
PWR-3 = 2.0E-08/py
PWR-5 = 3.0E-10/py
PWR-7 = 2.1E-08/py

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

$\bar{F}^*_{PWR}$ = 4.2E-08/py

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

W^*_{PWR} = 0.11 man-rem/py

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

$(\Delta\bar{F})_{PWR}$ = 1.3E-07/py $(\Delta\bar{F})_{BWR}$ = 5.7E-08/py[a]

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

$(\Delta W)_{PWR}$ = 0.32 man-rem/py $(\Delta W)_{BWR}$ = 0.39 man-rem/py[a]

(a) See Attachment 2.

TABLE 1. (cont'd.)

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.3E+03	5.1E+04	0

ATTACHMENT 2 (To Table 1)

Over the reporting period reviewed (May 1981 through January 1982), two LERs were found in which inadvertent actuation of an FPS potentially caused failure of a safety system typically represented in a plant's dominant accident sequences (with respect to core-melt):

1. Water contaminating the diesel generator (DG) fuel tanks at Surry 2 on May 28, 1981.

2. Water contaminating the high pressure coolant injection (HPCI) lube oil system and causing electrical grounds in the HPCI 125-v DC control system, thereby resulting in the HPCI system being declared inoperative (incident at Dresden 2 on December 24, 1981).

Both events typify common-cause failures, the first for onsite AC emergency power (via the DGs) and the second for the HPCI system. For the purpose of estimating the public risk reduction associated with resolution of issue 57, these failures are assumed representative of all those associated with inadvertent FPS actuation. Analysis is performed only for these failures.

Given the seven-month reporting period that was reviewed and assuming that 70 plants were operational during that period, the frequency of each incident is estimated to be:

$$1/[(70 \text{ plants})(7/12 \text{ yr})] = 0.024/\text{py}$$

Assuming each incident resulted in a potential common-cause failure of the respective system (onsite emergency AC and HPCI), the above frequency is taken to represent that for common-cause failure of each respective system due to inadvertent actuation of the FPS.

A review of Table A.4 (Andrews 1982) for Oconee indicates that the following parameters relate to failures of onsite AC and HPCI:

Onsite AC	HPCI
(B_3)	A1
	B1
	C1
	HPMAN
	HPMAN1
	RCSRBCM

For this issue, (B_3) should be interpreted as if it referred to DGs, as was done in issue B-56, "Diesel Generator Reliability."

Defining the terms ZD and ZH to represent "common-cause failure of onsite emergency AC power (via the DGs) and HPCI, respectively, due to inadvertent actuation of the FPS," one redefines the previous list of onsite AC- and HPCI-related parameters as follows to include these common-cause failures:

$$(B_3) = (B_3)_o + ZD$$

$$A1 = A1_o + ZH$$

$$B1 = B1_o + ZH$$

$$C1 = C1_o + ZH$$

$$HPMAN = HPMAN_o + ZH$$

$$HPMAN1 = HPMAN1_o + ZH$$

$$RCSRBCM = RCSRBCM_o + ZH$$

where the terms with "o" subscripts represent the parameters as originally defined.

Substituting these redefined parameters into the list of dominant minimal cut sets (Table A.3 of PNL-4297) and utilizing Boolean algebra, one obtains the following list of affected accident sequences and cut sets:

Sequence	Cut Sets
T_2MLU	$T_2 \cdot M \cdot CONST1 \cdot PCSNR \cdot ZH$ $T_2 \cdot M \cdot F1 \cdot G1 \cdot PCSNR \cdot ZH$ $T_2 \cdot M \cdot F1 \cdot CH4 \cdot PCSNR \cdot ZH$
T_1MLU	$T_1 \cdot M \cdot CONST2 \cdot ZH$ $T_1 \cdot M \cdot F1 \cdot G1 \cdot ZH$ $T_1 \cdot M \cdot F1 \cdot CH4 \cdot ZH$
$T_1(B_3)MLU$	$T_1 \cdot ZD \cdot M \cdot HHMAN \cdot LOPNRE$ $T_1 \cdot ZD \cdot M \cdot HHMAN \cdot ZH$
T_2KMU	$T_2 \cdot K \cdot M \cdot ZH$
S_2D	$S_2 \cdot ZH$
S_3D	$S_3 \cdot ZH$
T_2MQD	$T_2 \cdot M \cdot \overline{P}_1 \cdot Q \cdot ZH$

To evaluate the frequencies of the affected cut sets and sequences, it is necessary to estimate the unavailabilities of ZD and ZH. Each has a frequency of 0.024/py. To convert this to an unavailability, the expected duration of the fault is needed. Based on the LER descriptions of the two incidents chosen as representative, it seems reasonable to assume that plant personnel respond immediately to an indication of FPS actuation (inadvertent or otherwise). Furthermore, since these incidents indicate that the water contamination was recognized immediately upon investigating the FPS actuation, it is assumed that such faults are immediately recognized. Such should be the case with a safety-conscious plant crew. Thus, the expected fault duration should be no longer than that required to clean up and replace the contaminated DG fuel or HPCI lube oil. This should be accomplished within one crew shift, or about eight hours. Assuming an eight-hour expected fault duration, one obtains the following estimates for the unavailabilities of ZD and ZH:

$$\left.\begin{matrix} ZD \\ ZH \end{matrix}\right\} = (0.024/py)(8\ ph)(1\ py/8760\ ph) = 2.2E-5$$

The results of the public risk reduction analysis for Oconee 3 as the representative PWR (see Table 1 for completion of the analysis) are scaled for Grand Gulf 1 as the representative BWR in the following manner.

The RSSMAP studies for Oconee 3 and Grand Gulf 1 give total core-melt frequencies (F_o) of 8.2E-05/py and 3.7E-05/py, respectively, for these plants (Andrews et al. 1982). Using the original release category frequencies and the public dose factors (Appendix D of PNL-4297), one obtains total public risks (W_o) of 207 man-rem/py and 250 man-rem/py, respectively, for Oconee and Grand Gulf. For the purposes of scaling the base-case, affected core-melt frequency (F) and public risk (W), and the reductions in the core-melt frequency (ΔF) and public risk (ΔW) from Oconee to Grand Gulf, the following are assumed:

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

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

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

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

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

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

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

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Effects of Fire Protection System Actuation on Safety-Related Equipment/Systems (No. 57)

2-3. Affected Plants (N) and Average Remaining Lives ($\bar{T}$):

All plants (134) are affected and are categorized in three age groups:

Age Group	Affected Plants	N	$\bar{T}$ (yr)
1	Backfit PWRs	47	27.7
1	Backfit BWRs	24	25.2
2	Forward-fit PWRs commencing operation by 1986	24	27.7
2	Forward-fit BWRs commencing operation by 1986	13	27.5
3	Forward-fit PWRs commencing operation after 1985	19	30
3	Forward-fit BWRs commencing operation after 1985	7	30

For all 90 PWRs and 44 BWRs, the average remaining lives for this SIR are 28.2 yr and 26.6 yr, respectively.

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

PWR: (19,900 man-rem)(1.3E-07/py) = 2.6E-03 man-rem/py
BWR: (19,900 man-rem)(5.7E-08/py) = 1.1E-03 man-rem/py

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

Best Estimate	Upper Bound	Lower Bound
7.9 man-rem	62 man-rem	0

TABLE 2. (cont'd.)

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

It is assumed that 3 man-years of labor/plant are required for hardware fixes, with 25% of that time spent in radiation zones averaging 0.005 R/hr. This applies only to plants in operation by 1986 since these fixes can be implemented prior to operation (and, hence, necessitate no radiation exposure) at plants becoming operational in 1986 or beyond.

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

$$D = (0.005\ \text{R/hr})(0.25)(3\ \text{man-yr/plant})(44\ \text{man-wk/man-yr})(40\ \text{man-hr/man-wk})$$

$$= 6.6\ \text{man-rem/plant}$$

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

Only 108 LWRs are assumed to be in operation by 1986 (see Step 2).

$$ND = (108\ \text{LWRs})(6.6\ \text{man-rem/plant})$$

$$= 713\ \text{man-rem}$$

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

It is assumed that one additional man-week per plant-year will be required for examination of equipment installed in the various plant locations as part of the routine maintenance program. This applies to all 134 LWRs. Assuming a 75% utilization factor for actual work in the radiation fields gives

$$(0.75)(1\ \text{man-wk/py})(40\ \text{man-hr/man-wk}) = 30\ \text{man-hr/py}$$

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

Again, a dose rate of 0.005 R/hr is assumed.

$$D_O = (0.005\ \text{R/hr})(30\ \text{man-hr/py}) = 0.15\ \text{man-rem/py}$$

11. Total Occupational Dose Increase for SIR Operation and Maintenance (NTD_O):

$$N\bar{T}D_O = [90(28.2) + 44(26.6)](0.15) = 556\ \text{man-rem}$$

TABLE 2. (cont'd.)

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.3E+03	3.8E+03	4.2E+02

3.0 SAFETY ISSUE COSTS

The industry and NRC costs are estimated in this section. The results are summarized in Table 3. The funding of SIR activities for part one of the resolution to this issue is assumed to be already absorbed within the operating budgets of the NRC. Therefore, there are no specific costs identified with part one.

For part two of the resolution to this issue, it is assumed that the AEOD trend analysis continues through 1983. It is assumed further that a NUREG report is subsequently issued that documents all incidents of the past few years associated with the effects of FPS actuation on safety-related equipment. It is postulated that, as a result of comprehensive analysis of these additional data, an IE Bulletin is issued. It is assumed that the Bulletin directs all licensees to perform plant-specific reevaluations of their Fire Protection Programs. The chronology of anticipated events from that point on is summarized in Attachment 1 to Table 1 and is not repeated here.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Effects of Fire Protection System Actuation on Safety-Related Equipment/Systems (No. 57).

2-3. Affected Plants (N) and Average Remaining Lives ($\bar{T}$):

All 134 plants are affected and are categorized in three age groups:

Age Group	Affected Plants	N	$\bar{T}$ (yr)
1	Backfit PWRs	47	27.7
	Backfit BWRs	24	25.2

TABLE 3. (cont'd.)

Age Group	Affected Plants	N	$\tilde{T}$ (yr)
2	Forward-fit PWRs commencing operation by 1986	24	27.7
2	Forward-fit BWRs commencing operation by 1986	13	27.5
3	Forward-fit PWRs commencing operation after 1985	19	30
3	Forward-fit BWRs commencing operation after 1985	7	30

For all 90 PWRs and 44 BWRs, the average remaining lives for this SIR are 28.2 yr and 26.6 yr, respectively.

Industry Costs (Steps 4 through 12)

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

PWR: $\Delta(\overline{FA})$ = (\$1.65E+09)(1.3E-07/py) = \$2.15E+02/py

BWR: $\Delta(\overline{FA})$ = (\$1.65E+09)(5.7E-08/py) = \$9.4E+01/py

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

Best Estimate	Upper Bound	Lower Bound
\$6.6E+05	\$5.1E+06	0

6. Per-Plant Industry Resources for SIR Implementation:

No additional down-time is foreseen. All other estimates are included directly in the next step.

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

	108 Plants Operating by 1986	Remaining 26 Plants Operating After 1985
FPS reevaluation and report	\$3.0E+05	\$3.0E+05

TABLE 3. (cont'd.)

	108 Plants Operating by 1986	Remaining 26 Plants Operating After 1985
Procedural changes (excluding license amendment)	\$1.0E+05	\$1.0E+05
Class V license amendment	\$2.58E+04	--
Hardware fixes	\$1.5E+05	\$1.5E+05
Labor	\$3.0E+05	\$3.0E+05
I =	\$8.76E+05/plant	\$8.50E+05/plant

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

NI = 108(\$8.76E+05) + 26(\$8.50E+05) = \$1.17E+08

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

2 man-wk/py

This applies to all LWRs.

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

I_0 = (2 man-wk/py)(\$2270/man-wk) = \$4.54E+03/py

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

$N\bar{T}I_0$ = [90(28.2) + 44(26.6)](\$4.54E+03) = \$1.68E+07

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$1.3E+08	\$1.9E+08	\$7.5E+07

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Estimates included directly in next step.

TABLE 3. (cont'd.)

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

Continuing Trend Analysis	=	\$5.0E+04
NUREG	=	\$2.0E+05
IE Bulletin	=	\$1.67E+04
CD	=	\$2.7E+05

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

Assumed to consist of two parts:

1. Review and approval of licensee's reevaluation = 4 man-months
2. Onsite inspection = 4 man-months

Total = 8 man-months/plant or 0.67 man-yr/plant

This is assumed to apply only to the 108 plants operational by 1986 since these activities will presumably be incorporated initially into the normal review and inspection accorded plants prior to operation (as for the remaining 26 plants becoming operational after 1985).

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

C = (0.67 man-yr/plant)(\$1.0E+05 man-yr) = \$6.7E+04

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

NC = (108 plants)(\$6.7E+04/plant) = \$7.24E+06

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

0.5 man-wk/py

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

C_o = (0.5 man-wk/py)(\$2270/man-wk) = \$1,140/py

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

$N\bar{T}C_o$ = [90(28.2) + 44(26.6)](\$1140) = \$4.23E+06

TABLE 3. (cont'd.)

21. Total NRC Cost ($N):

Best Estimate	Upper Bound	Lower Bound
$1.2E+07	$1.6E+07	$7.5E+06

REFERENCES (For Issue 57)

Andrews, W. B. et al. 1982. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. PNL-4297. Pacific Northwest Laboratory, Richland, Washington.

Lainas, G. 1982. "Summary of Operating Reactor Events Meeting on January 7, 1982." January 13, 1982 Memo to D. Eisenhut, United States Nuclear Regulatory Commission, Washington, D.C.

Vollmer, R. E. 1982. "Effects of Fire Suppression Systems Actuated by Fire Detectors on Safety Equipment." November 2, 1982 Memo to E. L. Jordan, United States Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 58, Containment Flooding

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

In October 1980, an open-loop cooling water system that supplied the fan coolers in the Indian Point 2 PWR leaked and flooded the containment. As a result, short-term action was taken at Indian Point 2 and other reactors with similar systems to prevent flooding incidents. Work on a long-term resolution of the problem is being considered. A possible solution, assumed for this analysis, is to use a sump pump flow monitoring system that is monitored in the control room.

AFFECTED PLANTS	PWR: Operating = 38	Planned = 35
	BWR: Operating = 14	Planned = 12

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 0

OCCUPATIONAL DOSES:

SIR Implementation =	2.3E+02
SIR Operation/Maintenance =	1.4E+03
Total of Above =	1.6E+03
Accident Avoidance =	0

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:

SIR Implementation =	4.2
SIR Operation/Maintenance =	19
Total of Above =	23
Accident Avoidance =	0

NRC COSTS:

SIR Development =	0.11
SIR Implementation Support =	0.23
SIR Operation/Maintenance Review =	3.2
Total of Above =	3.5

CONTAINMENT FLOODING

ISSUE 58

1.0 SAFETY ISSUE DESCRIPTION

The containment flooding issue was established as a result of a flooding event that occurred at the Indian Point 2 reactor in October 1980. A large quantity of water leaked from fan coolers, with an open water supply system,[a] onto the containment building floor and subsequently filled the sumps and the cavity beneath the reactor vessel. The cavity accumulated enough water to wet the bottom of the reactor vessel to a height of about nine feet.

There were multiple causes of the event, but the main cause was probably a lack of knowledge by the operators that the sump pumps were inoperable. There was no way to accurately monitor the sump flow rate, which had stopped.

The main concern at the time of the event writeup was the possibility of damage to the reactor pressure vessel from thermal stresses. Subsequent analyses indicated that cracking of the vessel should not occur. Another concern was that, due to the brackish water, chloride-induced cracking might occur in stainless steel conduits and instrument thimbles at the reactor vessel base. Detailed inspection showed that no damage occurred. There was also some concern about flooding safety-related electrical equipment, but no damaged equipment was found in the flooded region.

Other concerns recorded include the following:

- Leak openings in an open water supply system might cause a post-LOCA release path from containment.
- Flood water could cause boron dilution in the core cooling water following a LOCA and contribute to a recriticality event.
- Leakage from fan cooler systems could reduce the post-LOCA ability to cool the containment.

Of the several concerns raised, only the following are directly related to flood effects:

- thermal-induced cracking of the pressure vessel

(a) An open-loop system receives and discharges water without recycling or accountability. In this case, water was directly drawn and discharged back into the Hudson River.

- chloride-induced cracking of the pressure vessel
- failure of electrical equipment
- boron dilution following a LOCA.

The other two concerns, leak openings providing a post-LOCA release path from containment and leakage-induced inoperability of the containment fan coolers after a LOCA, do not require concurrent flooding to be potentially detrimental. Thus, for the purposes of this issue analysis, only the first four concerns, those directly related to flooding, are addressed.

As a result of the containment flooding incident, short- and long-term steps were established to resolve the issue. IE Bulletin 80-24 was issued on November 21, 1980, in response to the IP-2 incident. The bulletin required that all plants with open cooling water systems take a number of short-term actions to preclude IP-2 type events in the interim, before longer-term generic actions could be applied. These actions are still in place pending long-term resolution of the flooding issue. Both the short- and long-term resolutions are complex because of the many different designs involved and may need to be handled on a plant-by-plant basis. Long-term solutions would be aimed at improving the systems to detect, alarm, and prevent containment flooding.

There are potential flooding problems with closed water systems within containment that are supplied with automatic makeup water (e.g., pump seal, pump cooling, control rod injection). Also, open systems that are normally closed (e.g., fire protection, cleanup, post-LOCA) are potential flooding sources.

The long-term solution is likely to be plant specific. Some of the potential solutions that have been mentioned include:

- improved sump level indicators
- continuous sump(s) inventory system for control room monitoring
- pump totalizer indicators for control room
- improved alarm systems
- periodic physical surveillances by operators where practical
- television surveillance systems
- capacitance water-alarms
- periodic hydrostatic test checks for leaks to reduce risk of flooding.

Some plants already have a variety of these systems.

PROPOSED RESOLUTION

The proposed solution consists of installing a sump flow rate monitoring system with control room readout for surveillance of the flow volume and rate out of the sump. The system is assumed to be comprised of a continuous recorder which indicates the time in which the primary sump pumps are on and off. The reactor operator would check the recorder at least once per shift to assure that the pump flow cycle is repetitive.[a] Limit controls would be established for normal minimum and maximum pump cycle times. When the time limits are exceeded in either direction, this would be indicated by a blinking light to gain an operator's attention. The operator would be able to take appropriate action based upon whether the sump flow was decreasing or increasing.

AFFECTED PLANTS

The safety issue resolution (SIR) would apply to the plants with open water systems in containment. Plants whose closed water systems in containment are provided with automatic feed water makeup have not been included. Some of the plants already have these or better systems installed. The installation work would take place during a scheduled outage.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk associated with in-containment flooding is believed to be too low to merit the detailed analysis normally provided in estimating the risk reduction due to SIR. Instead, some arguments and scoping calculations are presented below to validate this contention. Thus, no Public Risk Reduction Work Sheet is prepared for this issue. The standard occupational dose analysis is performed, however, and the results are summarized in Table 1.

As stated in the IP risk study (PASNY 1982):

"The impact on plant risk due to concurrent service water system rupture and LOCA was investigated for possible increases in plant risk (due primarily to increased water in the containment sump and possible flooding of equipment necessary for the recirculation phase).

This event (October 17, 1980 flood at IP-2) is an insignificant contributor to plant risk and is not quantified for the following reasons:

(a) Normally, the pump(s) start after the sump water level rises to a certain point, then shut off when the sump is nearly empty.

1. The service water pipe rupture must either occur as a result of the LOCA or occur randomly after the start of the LOCA.

2. If the service water pipe were to rupture, the individual service water lines to the containment fan cooler units can be readily isolated from outside the containment building (two MOVs in series cooler outlet and fan cooler motor outlet service water lines for Unit 2, and similar manual valves for Unit 3).

3. Flow and temperature indication is available to allow rapid detection of a failed service water line." (PASNY 1982)

Thus, the IP-2 event was judged to be too insignificant to plant risk to merit quantification in the risk study performed for IP-2 and 3.

The problem of in-containment flooding was also addressed in the Zion risk study:

"By design, confirmed by the Three Mile Island and recent Indian Point Unit 2 experience, the containment buildings can tolerate a substantial water inventory without loss of safety functioning. The only substantial flooding sources in the containment are service water pipe [or Reactor Containment Fan Cooler (RCFC) coil] failure and inadvertent containment spray actuation. Post-TMI evaluations have shown that the Zion containment building can handle approximately two Refueling Water Storage Tank (RWST) volumes before beginning to approach problems with critical electrical equipment.

Given reactor vessel insulation in place, the reactor vessel would be protected from thermal shock by steam blanketing in the vessel/insulation annular spaces. Therefore, spray actuation which might wet portions of the vessel has a negligible probability of causing a core melt. As before, the probability of failing any of the service water pipes is 1.5×10^{-4} per year. There are several types of indicators that would alert the operator:

1. Main containment sump levels

2. Reactor cavity sump levels

3. Recirculation sump levels

4. Individual service waterline pressure indicators reading in control room.

We assess the probability of any of the first three indicators failing at 10^{-3} and the last one at 10^{-4}. We judge the probability of an operator ignoring all these, if functioning, as 10^{-3} for the first 2 hours and then 10^{-4} thereafter, considering that plant status is reviewed. Human error obviously dominates. Human error would terminate as operator examination of the containment occurred, as shift change occurred, and as review of radwaste logs took place. We judge that the leak could not last more than 8 hours without being detected.

The probabilities (frequencies) for an undetected leak therefore are:

- 1st 2 hours 1.5×10^{-7} [per year]
- 3 to 8 hours 1.5×10^{-8} [per year]

Even if we assume a large rupture of a fan cooler service waterline equivalent to half of full flow to one RCFC, it takes over 13 hours to exceed the two RWST criterion (sic). At that point, core melt conditions are not achieved. Some safety-related electrical equipment has been potentially endangered. As transmitters, etc., fail, the problem would be quickly identified and terminated by operator inspection of the in-containment transmitters and subsequent leak isolation.

We therefore judge that the probability of a core melt resulting from in-containment leakage and flooding is too small to visibly contribute to risk." (Commonwealth Edison Co. 1981)

Thus, as for IP-2 and 3, in-containment flooding was judged not to contribute significantly to the risk at the Zion PWRs.

Both thermal-induced and chloride-induced cracking of the pressure vessel in welds could initiate LOCAs if the cracking breaches the pressure boundary. However, the leak would have to be of sufficient magnitude and go undetected long enough to contact the vessel itself. The Zion calculations show this frequency to be on the order of 1E-07/py to 1E-08/py. Should this occur, the likelihood of severe crack growth would be very small (say <1E-03), bringing the frequency of a flood-induced breach of the pressure boundary to <1E-10/py.

Endangering of electrical equipment would require such a leak to go undetected for even longer periods of time since this equipment lies above the lower portion of the pressure vessel. The frequency of an undetected flood lasting over eight hours is <1E-08/py. Furthermore, the electrical equipment would have to fail in such a way as to cause inadvertent opening of isolation or relief valves to potentially induce a LOCA. Thus, the frequency of this type of LOCA should also be <1E-10/py, since the likelihood of electrical equipment failing in such a way as to induce a LOCA is small (say <1E-02).

Flood-induced boron dilution poses a potential problem only after a LOCA has occurred. Since the frequency of a flood-induced LOCA has been shown to be <1E-10/py only independent LOCAs considered for down dilution. The frequency of an independent LOCA was taken as 1E-03/py. The frequency of a flood of sufficient magnitude to dilute boron to levels where recriticality is possible is taken as 1E-07/py (two-hour duration at Zion). Assuming plant susceptibility to these concurrent failures lasts about one month from the LOCA occurrence, the frequency of LOCA coupled with flood-induced boron dilution is (1E-03/py)(1E-07/py)(1 mo/12 mo/py) ≈ 1E-11/py.

Based on existing studies and these simple scoping calculations, it seems reasonable to assume that the public risk associated with in-containment flooding is negligible. Therefore, no public risk reduction analysis was performed for this issue.

TABLE 1. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Containment Flooding (58)

2. Affected Plants (N):

Operating PWRs:	38	Operating BWRs:	14
Planned PWRs:	35	Planned BWRs:	12
	73		26

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

PWR:	Operating $\bar{T}$	= 27.7 years	28.8 years
	Planned $\bar{T}$	= 30 years	
BWR:	Operating $\bar{T}$	= 25.2 years	27.4 years
	Planned $\bar{T}$	= 30 years	

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

   Zero

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

   Zero

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

   It is assumed that each operating plant will require 160 man-hours inside containment and 210 man-hours outside containment for SIR implementation. This involves installation of containment sump flow rate monitors with control room readouts.

TABLE 1. (contd)

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

Dose rates of 25 mR/hr and 2.5 mR/hr are assumed for work inside and outside containment, respectively.

$$D = (160\ \text{man-hr/plant})(0.025\ \text{R/hr}) + (210\ \text{man-hr/plant})(0.0025\ \text{R/hr})$$
$$= 4.5\ \text{man-rem/plant}$$

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

$$ND = (52\ \text{operating plants})(4.5\ \text{man-rem/plant})$$
$$= 230\ \text{man-rem}$$

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

It is assumed that 20 man-hr/py will be required for operation checks and maintenance within containment. No additional operation and maintenance outside containment will involve work in radiation zones (primarily only in the control room).

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

Again, a 25 mR/hr radiation field is assumed.

$$D_0 = (20\ \text{man-hr/py})(0.025\ \text{R/hr})$$
$$= 0.50\ \text{man-rem/py}$$

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

$$N\bar{T}D_0 = [(73\ \text{PWRs})(28.8\ \text{yr}) + (26\ \text{BWRs})(27.4\ \text{yr})](0.50\ \text{man-rem/py})$$
$$= 1410\ \text{man-rem}$$

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1600	4900	550

3.0 SAFETY ISSUE COSTS

Results of the analysis of the costs associated with SIR are provided in Table 2.

TABLE 2. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Containment Flooding (58)

2. Affected Plants (N):

Operating PWRs:	38	Operating BWRs:	14
Planned PWRs:	35	Planned BWRs:	12
	73		26

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

PWR:	Operating $\bar{T}$ = 27.7 years	Planned $\bar{T}$ = 30 years	28.8 years
BWR:	Operating $\bar{T}$ = 25.2 years	Planned $\bar{T}$ = 30 years	27.4 years

Industry Costs (Steps 4 through 12):

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

   Zero

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

   Zero

6. Per-Plant Industry Resources for SIR Implementation:

   Labor (engineering, craft services, etc.) = 14 man-wk/plant

   Equipment (cost estimated directly in next step)

   Additional Down-time = none

   These apply to all affected plants.

TABLE 2. (contd)

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

Labor = (14 man-wk/plant)($2270/man-wk) = $3.2E+04/plant

Equipment = $1.0E+04/plant

I = $4.2E+04/plant

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

NI = (99 plants)($4.2E+04/plant)

= $4.16E+06

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

Operations = 1.5 man-wk/py

Equipment Maintenance = 1.5 man-wk/py

3 man-wk/py

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

I_o = (3 man-wk/py)($2270/man-wk)

= $6810/py

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

$N\bar{T}I_o$ = [(73 PWRs)(28.8 yr) + (26 BWRs)(27.4 yr)]($6810/py) = $1.92E+07

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$2.3E+07	$3.3E+07	$1.4E+07

NRC Costs (Steps 13 through 21):

13. NRC Resources for SIR Development:

NRC Staff Labor = 12 man-wk

Contractor Support (cost estimated directly in next step)

TABLE 2. (contd)

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

Labor = (12 man-wk)($2270/man-wk) = $2.7E+04

Contractor Support = $8.1E+04

C_D = $1.08E+05

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

1 man-wk/plant

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

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

= $2270/plant)

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

NC = (99 plants)($2270/plant)

= $2.25E+05

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

0.5 man-wk/py

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

C_o = (0.5 man-wk/py)($2270/man-wk)

= $1140/py

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

$N\bar{T}C_o$ = [(73 PWRs)(28.8 yr) + (26 BWRs)(27.4 yr)]($1140/py) = $3.21E+06

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$3.5E+06	$5.2E+06	$1.9E+06

REFERENCES (For Issue 58)

Commonwealth Edison Co. 1981. Zion Station Probabilistic Safety Study. Chicago, Illinois.

Power Authority of the State of New York (PASNY) and Consolidated Edison Co. of New York, Inc. 1982. Indian Point Probabilistic Safety Study. New York.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 59, Technical Specifications for Plant Shutdown when Safety Equipment is Inoperable

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Technical specifications for LWRs currently require plant shutdown when certain safety equipment is inoperable, primarily the emergency electrical or core cooling systems. However, situations can be postulated where the chance of core damage is increased on shutdown, due to induced transients which may then challenge the safety systems. Event trees for these scenarios would be better quantified using probabilistic analysis techniques, with subsequent changes in technical specifications recommended.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	947
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	38
Total of Above =	38
Accident Avoidance =	3.41

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	7.38
SIR Operation/Maintenance =	-171
Total of Above =	-164
Accident Avoidance =	0.52
NRC COSTS:	
SIR Development =	0.35
SIR Implementation Support =	0.016
SIR Operation/Maintenance Review =	0
Total of Above =	0.37

TECHNICAL SPECIFICATIONS FOR PLANT SHUTDOWN

WHEN SAFETY EQUIPMENT IS INOPERABLE

ISSUE 59

1.0 SAFETY ISSUE DESCRIPTION

This issue deals with a potential generic safety problem in PWRs and BWRs when technical specifications (TS) currently require plant shutdown when certain safety equipment is inoperable. The concern is that the TS do not fully consider the potential for placing the plant in a "less safe" condition as a result of shutting down an otherwise normally functioning unit. The potential does exist for transients during shutdown, where the inoperable system normally serves as a line of defense. The question is whether continued plant operation while undergoing repairs is a "safer" option than shutdown, and for what systems and under what conditions.

PROPOSED RESOLUTION

The safety equipment in question centers primarily on the emergency electrical system and subsystems of the emergency core cooling systems (ECCSs). Recent licensee event reports (LERs) have highlighted both of the above (Minners 1982); however, interest is focused primarily on the ECCSs (Baer 1982). A number of cases of single or multiple subsystem failures in BWR ECCS operation where continued plant operation should be considered were presented by Baer. These are suggestions for areas requiring further study. As a result, no cases have yet been fully quantified to determine whether the risk of plant operation can be reduced by modifying the TS. Because of the wide range of possible system failures and operating conditions, this quantification will be tedious, but probabilistic risk assessment (PRA) techniques are being promoted by the NRC. An initial examination indicates that the problem would center primarily on obtaining data for the conditional probability of inducing specified transients on plant scram or controlled shutdown.

At this time, little more can be done than a preliminary examination of several sequences to see whether this issue has merit for more vigorous study. The approach used here is to select specific failures in the ECCSs of a BWR plant, primarily as a result of previous NRC considerations for BWRs on this issue. For continued operation, the probability of a feedwater transient (loss of function) is examined over some postulated repair interval. For shutdown, a conditional probability on loss of feedwater given scram is estimated from what little data are available.

Another possible scenario for this issue involves the potential for losing offsite power due to a scram-induced transient when diesel generators are inoperable. This is based on the incident at Quad Cities (Minners 1982) where the diesel generators were out, and scram during a high grid load period could have induced a loss of offsite power. An initial examination of this scenario indicates that modifications to the TS are required, but the estimated change in core-melt frequency is less than that estimated for similar scenarios involving the ECCSs. Therefore, at this time the analysis will focus only on the influence of the ECCSs.

This safety issue resolution (SIR) could require modification of the TS to acknowledge when continued operation is preferable. Three alternatives are considered likely at this time:

1. Permitting continued operation at full power to minimize the likelihood of a transient that would cause the reactor to trip.

2. Requiring the plant to slowly reduce power to about 30-50% of full power and then allowing the plant to continue operation at this power level, essentially indefinitely, while the inoperable equipment is being repaired. Thus, if a reactor trip occurs before the equipment is repaired, the decay heat generation rate will be reduced.

3. Allowing the plant to slowly reduce power to about 30-50% of full power, and then requiring shutdown after some period of time (a few days, for example) if the inoperable equipment is not repaired. Shutdown would be required only if at least a minimum amount of equipment necessary to achieve hot, then cold, shutdown was available.

The two cases examined for this safety issue consider full-power operation versus scram to provide an estimate of the potential risk changes associated with each mode of operation. This estimate is presented in Attachment 1.

AFFECTED PLANTS

ECCS components particular to BWRs are utilized in this initial examination of Issue 59. However, it is assumed that the estimates derived are applicable to all LWRs in general.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational doses associated with the issue resolution are estimated in this section. Analysis results are summarized in Tables 1 and 2, respectively. The analysis is conducted for a hypothetical BWR rather than a representative PWR and BWR, as a consequence of the data base employed for this issue analysis (see Attachment 1).

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Technical Specifications for Plant Shutdown when Safety Equipment is Inoperable (59)

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

	N	$\bar{T}$
PWR	90	28.8
BWR	44	27.4

3. Plants Selected for Analysis:

A hypothetical BWR is assumed to be representative of all LWRs.[a]

4-7. Steps Related to Affected Parameters, Accident Sequences, Release Categories, and Their Base-Case Values:

The base-case, affected core-melt frequency is estimated directly in the next step. Thus, these steps are omitted.

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

$\bar{F}$ = 2.6E-07/py (BWR), and 2.03E-08/py(PWR)[a]

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

W = 1.76 man-rem/py (BWR), and 5.12E-02 man-rem/py (PWR)[a]

This assumes a dose of 6.76E+06 man-rem per event for BWRs, and 2.52E+06 man-rem per event for PWRs.

10-12. Steps Related to Adjusted-Case Values of Affected Parameters, Accident Sequences, and Release Categories:

The adjusted-case, affected core-melt frequency is estimated directly in the next step. Thus, these steps are omitted.

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

$\bar{F}^*$ = 1.6E-07/py and 9.0E-10/py (PWR)[a]

(a) See Attachment 1.

TABLE 1. (cont'd)

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

W^* = 1.08 man-rem/py (BWR), and 2.27E-03 man-rem/py (PWR)[(a)]

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

$\Delta\bar{F}$ = 1.0E-07/py (BWR), and 1.94E-08 (PWR)[(a)]

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

ΔW = 0.68 man-rem/py (BWR), and 4.89E-02 man-rem/py (PWR)

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
947	6.8E+04	0

(a) See Attachment 1.

ATTACHMENT 1 (To Table 1)

The Oak Ridge Precursor Study (Minarick and Kukielka 1982) is used here, along with data from an EPRI study (McClymont and Poehlman 1982) on anticipated transients without scram (ATWS) to calculate a base-case and adjusted-case core-melt frequency for this issue. The problem centers on the TS for shutdown of a reactor, given that a safety system is inoperable. The question is whether it is safer to shut the reactor down or continue operation. The shutdown process may, in fact, result in a transient which could call on the failed safety system.

LERs concerning this issue (Minners 1982) have dealt primarily with emergency electrical power systems and components of the ECCSs. Internal NRC concerns on this issue (Baer 1982) center primarily on components of the ECCSs. As a result, the approach used in this analysis selects specific components of the ECCSs for study.

The techniques and data presented in the Precursor Study are modified here to allow a comparison of the risk of core damage with a safety system inoperable for continued reactor operation versus immediate scram. To accomplish this, a specific system(s) must be chosen for failure, and appropriate event trees developed. Data on system failure must then be adapted to fit the need for failure frequency, failure on demand, or failure over a specified time interval.

The Baer Memorandum (1982) considers the following failure scenarios for BWRs:

1. Reactor core isolation cooling system (RCICS) inoperable

2. RCICS and high pressure coolant injection system (HPCIS) inoperable

3. RCICS and one or more residual heat removal (RHR) subsystems inoperable

4. HPCIS inoperable

5. HPCIS and one or more RHR subsystems inoperable

6. Low pressure coolant injection system (LPCIS) partially inoperable

7. LPCIS inoperable.

For this simple analysis, failure of both the RCICS and HPCIS (Case 2) are examined, along with a second case of failure of the automatic depressurization system (ADS). It would be useful to examine failure of the LPCIS (Case 7); however, no LERs exist on this system in the Precursor Study. As a result, the ADS is examined, the function of which precedes that of the LPCIS during a loss of feedwater transient.

ATTACHMENT 1. (cont'd.)

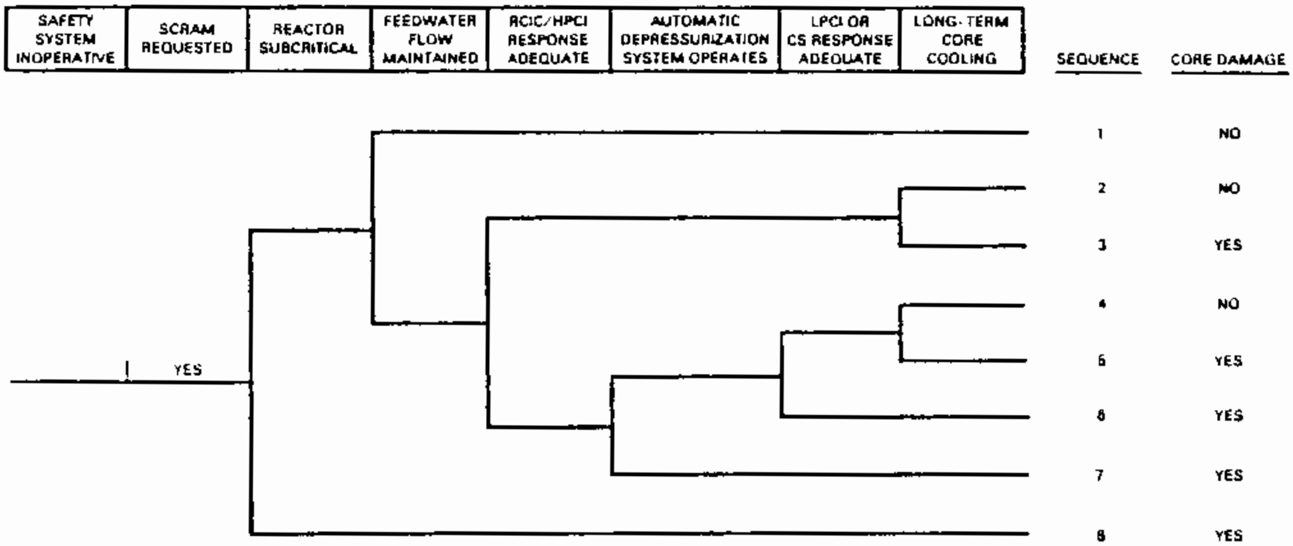


FIGURE 1.a. Scram Requested on ECCS Subsystem Failure

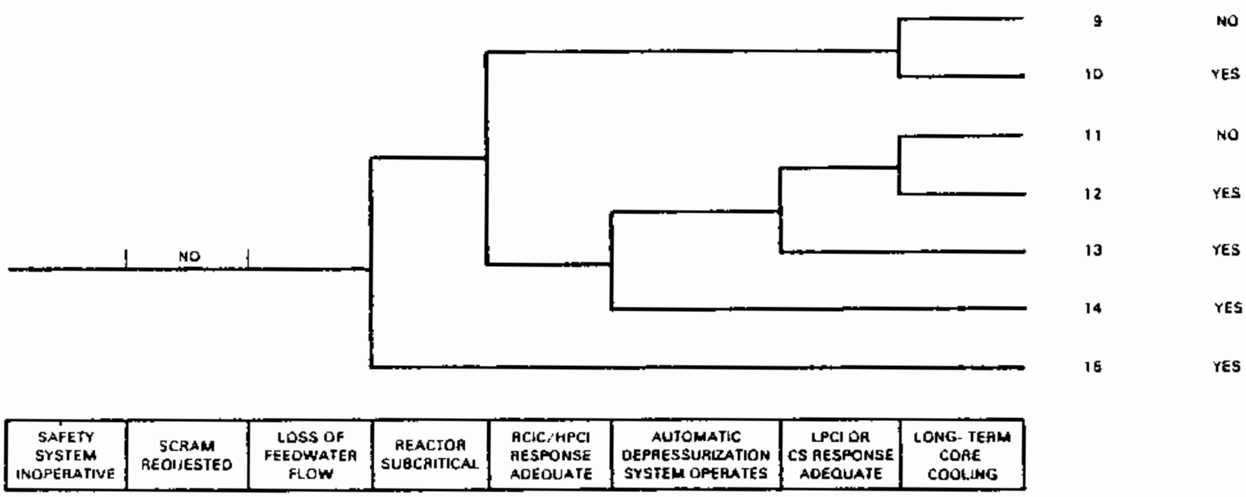


FIGURE 1.b. No Scram Requested on ECCS Subsystem Failure

To examine this issue, the generic event trees shown in Figures 1.a and 1.b were used, based on the flow logic developed in the Precursor Study for BWR transients. The upper event tree (Figure 1.a) depicts a failure of a safety system followed by a request for scram by the operator. The transient which could then follow is shown as "loss of feedwater given scram," chosen here as representative of transients which would challenge the ECCS. The lower event tree (Figure 1.b) depicts the case where operation continues. Another initiating event is then required, taken here as loss of feedwater given an ECCS subsystem failure. The data required to complete this analysis are given in Table 1.1.

ATTACHMENT 1. (cont'd.)

TABLE 1.1. BWR ECCS Subsystem Failure Data[a]

Event Description	Occurrences	Plant-Years	Demands	Failure Frequency (1/py)	Failure Probability on Demand
Loss of Feedwater	39	66	-	0.58	-
Reactor Subcritical	-	-	-	-	1.3E-06
RCICS/HPCIS Failure	4.9	99	-	0.049	0.0039
ADS Failure	4	148	148	0.027	0.027
LPCIS/LPCSS Failure	-	-	-	-	5.6E-04
Long-Term Core Cooling Failure	-	-	-	-	1.1E-04
Loss of Feedwater on Scram					0.0025[b]
Loss of Feedwater Given RCICS/HPCIS Failure (independent failure over a 1-day period)					0.0016[b]

(a) Data taken from NUREG/CR-2497 (Minarick and Kukielka 1982) unless indicated otherwise.
(b) See text for explanation.

The analysis results hinge on the probability of inducing a feedwater transient on scram. Data for this value are lacking at this time, so a value is estimated based on the ATWS report. For BWR transient category 26 (decreasing feedwater flow during startup or shutdown), the frequency reported is 0.07/py. It is assumed here that this is applicable to the 9 scrams/py for BWRs, resulting in a probability of 0.01 for a feedwater transient on scram. It is further assumed that 50% of these transients are decreases in feedwater during shutdown, with 50% of these resulting in loss of function. The probability, p, of loss of feedwater on shutdown of

$$p = (0.01/\text{shutdown})(0.50)(0.50) = 0.0025/\text{shutdown}$$

To estimate the probability of feedwater failure during an ECCS subsystem outage, a one-day failure duration is assumed. The probability of failure increases with time, so this initial calculation possibly gives a conservative estimate of the worth of this issue. The probability, p, of independent loss of feedwater over the one-day ECCS subsystem outage becomes

$$p = 1\text{-}e^{-\ t} \qquad t = 0.0004$$

where = 0.13/py
t = (1 day)/(365 days/yr) = 0.0027 yr

The appropriate data have been entered on Figures 2.a, 2.b, 3.a, and 3.b. The results are summarized below for pathways leading to core damage as a result of RCICS/HPCIS failure (Figures 2.a and 2.b) and ADS failure (Figure 3.a and 3.b).

1. RCICS/HPCIS Failure (Figure 2)

   a. Scram Requested (base case)

Core-Damage Sequence	Frequency (1/py)
5	1.3E-08
6	6.7E-08
7	3.3E-06
8	6.4E-06
	3.5E-06

   b. No Scram Requested (adjusted case)

Core-Damage Sequence	Frequency (1/py)
12	2.2E-09
13	1.1E-08
14	5.3E-07
15	2.5E-11
	5.4E-07

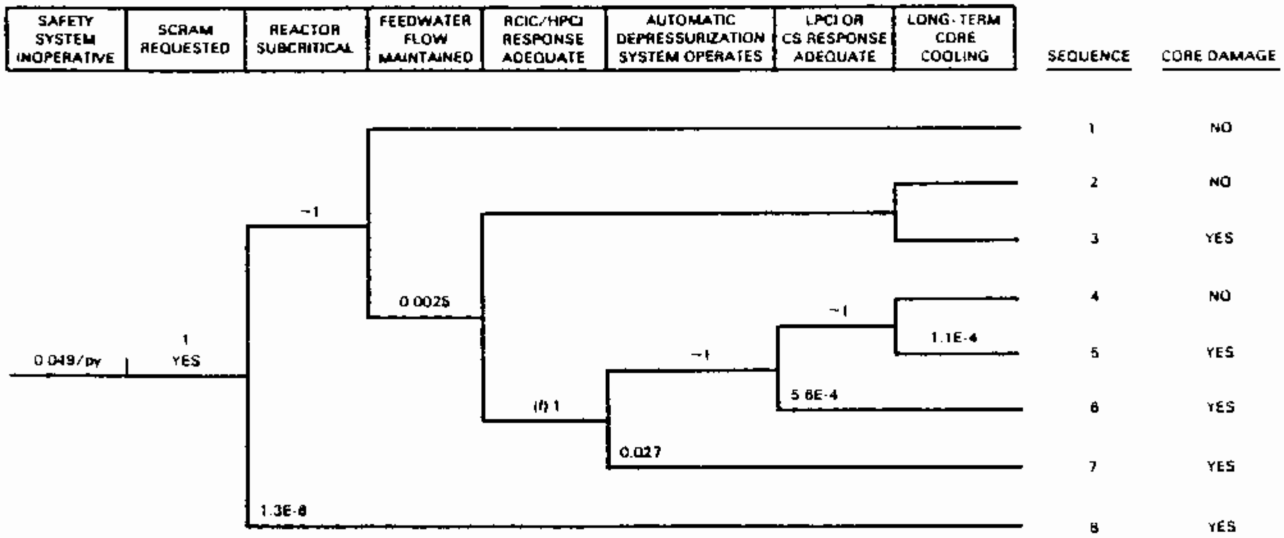


FIGURE 2.a. HPCIS/RCICS Failed, Scram Requested

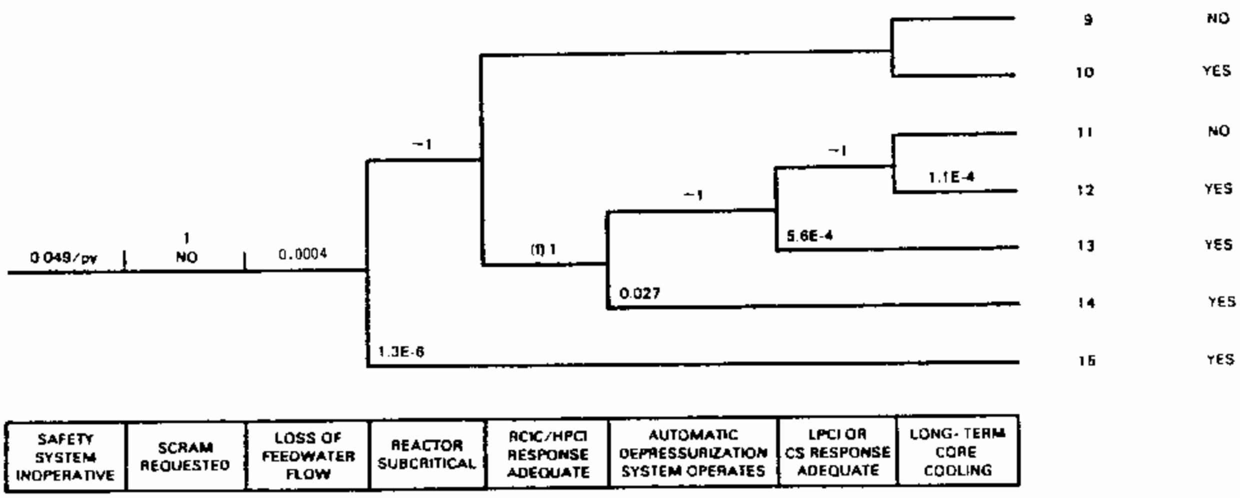


FIGURE 2.b. HPCIS/RCICS Failed, No Scram Requested

(f) Assumed failure branch (probability = 1)

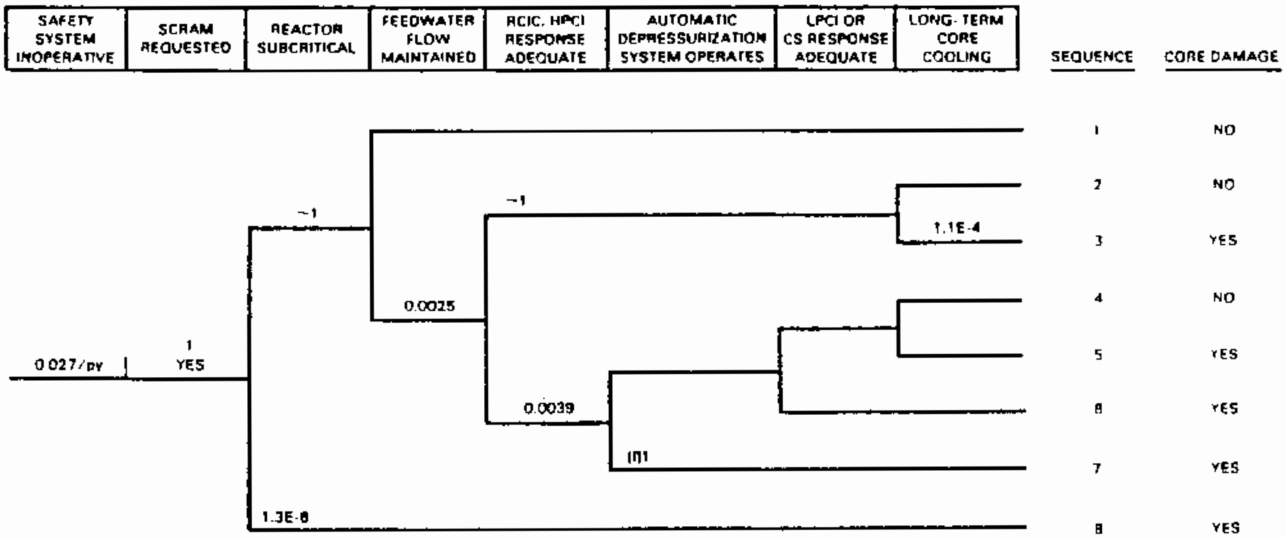


FIGURE 3.a. ADS Failed, Scram Requested

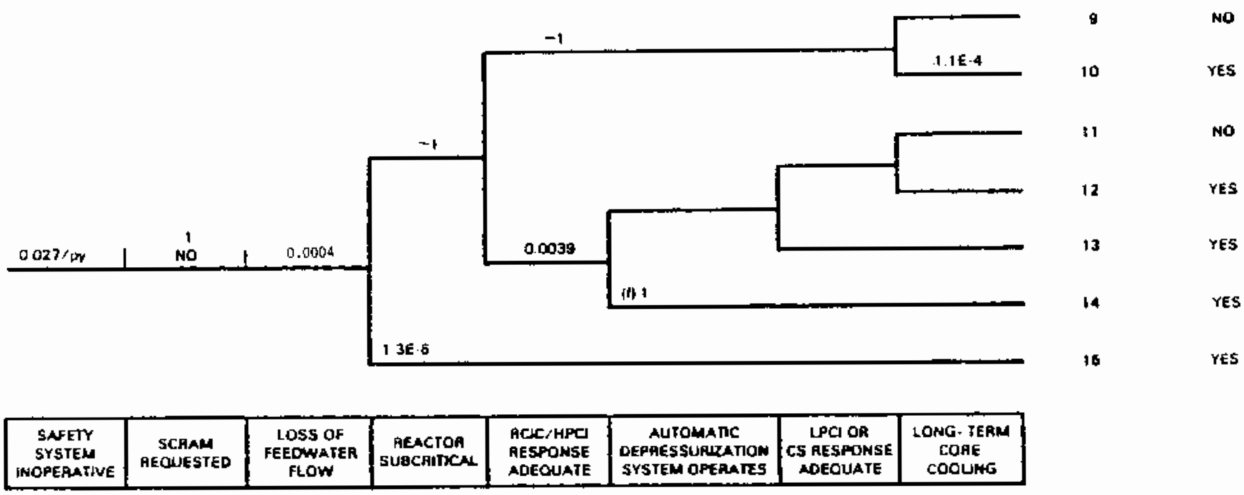


FIGURE 3.b. ADS Failed, No Scram Requested

(f) Assumed failure branch (probability = 1)

ATTACHMENT 1. (cont'd.)

2. ADS Failure (Figure 3)

a. Scram Requested (base case)

Core-Damage Sequence	Frequency (1/py)
3	7.4E-09
7	2.6E-07
8	3.5E-08
	3.1E-07

b. No Scram Requested (adjusted case)

Core-Damage Sequence	Frequency (1/py)
10	1.2E-09
14	4.2E-08
15	1.4E-04
	4.3E-08

Combining the results for the two examples, one obtains the following core-damage frequencies:

Base Case

RCICS/HPCIS Failure, Scram Requested	=	3.5E-06/py
ADS Failure, Scram Requested	=	3.1E-07/py
		3.8E-06/py

Adjusted Case

RCICS/HPCIS Failure, No Scram Requested	=	5.4E-07/py
ADS Failure, No Scram Requested	=	4.3E-08/py
		5.8E-07/py

The event trees developed for the Precursor Study give a measure of core damage only. To equate this with the core-melt frequency used in other risk studies, the above core-damage frequencies are divided by a factor of 30.[a] Thus, the base and adjusted-case, affected core-melt frequencies become 1.3E-07/py and 1.9E-08/py, respectively.

The preceding calculations were performed for BWRs but are assumed to be representative of all LWRs. Other sequences are possible, but the limited scope of this simple analysis precludes more comprehensive analysis. Thus, to allow for contributions from other possible scenarios, these base and adjusted-case, core-melt frequencies are doubled for the purposes of estimating risk reduction due to SIR. This gives the following results:

$\bar{F}$ = 2.6E-07/py

$\bar{F}^*$ = 1.6E-07/py

$\Delta\bar{F}$ = 2(5E-08/py) = 1E-07/py

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

$$W_O = \bar{F}_O R_O$$

where W_O = overall risk

$\bar{F}_O$ = overall core-melt frequency

R_O = average dose factor.

Based on Appendices A-D of NUREG/CR-2800 (Andrews et al. 1983), an average LWR Risk, $(R_O)_{LWR}$, is 3.3E+06 man-rem per event. In this case, individual calculations will be presented for both BWR and PWR plants. In this case,

(a) As reported in Nucleonics Week, October 21, 1982, an analysis of the Oak Ridge Precursor Study by the Institute of Nuclear Power Operations (INPO) claims that the chances of a severe nuclear accident were estimated 30 times too high. Furthermore, severe core damage (assumed to be analogous to that at TMI-2 in the Precursor Study) is presumably less severe than core melt, the level of core damage normally considered in nuclear power plant risk studies. Based on these considerations, it is assumed that the frequency of core damage as assessed using the Precursor Study should be divided by INPO's factor of 30 to result in the frequency of core melt.

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

$\bar{T} = 28.8$ yr (PWR) and 27.4 yr (BWR)

$W_0 = 207$ man-rem/py (PWR) and 250 man-rem/py (BWR)

$\bar{F}_0 = 8.2E\text{-}05$/py (PWR) and 3.7E-05/py (BWR)

where $R_0 = 2.52E+06$ man-rem (PWR), and 6.76E+06 man-rem (BWR)

In this issue, base case BWR risk is then

$$W_0(\text{BWR}) = (2.6E\text{-}07/\text{py})(6.76E+06 \text{ man-rem}) = 1.76 \text{ man-rem/py}$$

The adjusted case BWR risks is (1.6E-07/py)(6.76E+06 man-rem) = 1.08 man-rem/py. The change in BWR risk is then

$$\Delta(W)_{BWR} = 0.68 \text{ man-rem/py}$$

A similar analysis was undertaken for the failure of a PWR safety system, again based on the event trees developed in the Precursor Study (NUREG/CR-2497). The modified event trees are shown in Figures 4a, and 4b. Again, the first figure depicts shutdown of the reactor upon defection of the failed safety system. The shutdown process is then assumed to present a potential challenge to the feedwater systems. In Figure 4b, the plant is left operating, and the probability of a feedwater failure occuring over a 24 hour period is presented as the initiating event to the sequences.

Data for the event trees is presented in Table 1.2.

TABLE 1.2. PWR Subsystem Failure Data

Event Description	Occurrence	Plant years	Demands	Failure Frequency (1/py)	Failure Probability on Demand
Loss of Feedwater	31	213.37	---	0.15	---
Reactor Subcritical	---	---	---	---	3.6E-05
Auxiliary Feedwater (on scram failure)	6.1	206	5624	2.96E-02	1.1E-03 1.1E-02
PORV Demanded	---	---	---	---	0.9
PORV or Isolation Valve Closure	---	---	---	---	2.9E-03

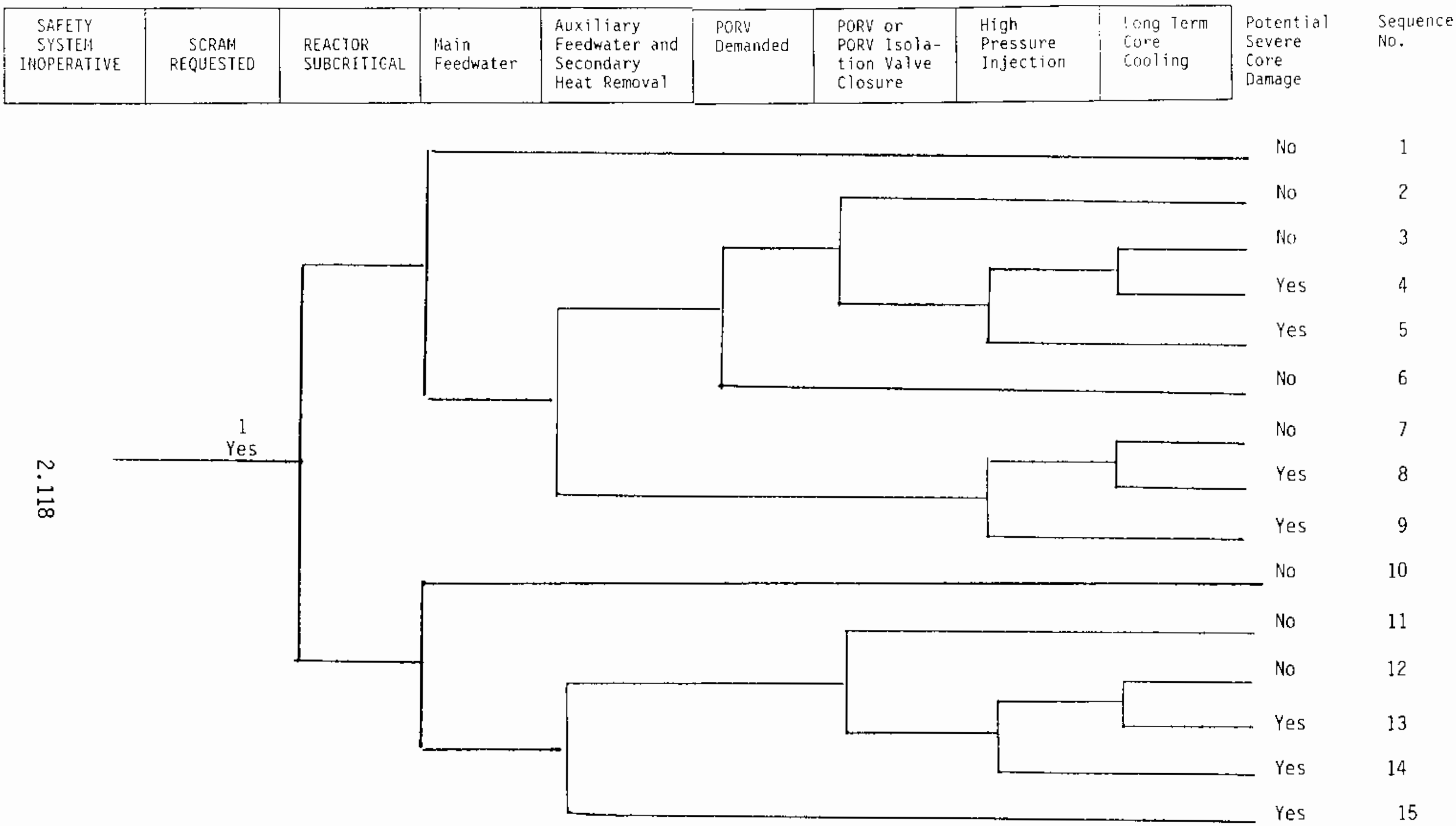


FIGURE 4.a. Safety System Inoperative, Shutdown Requested

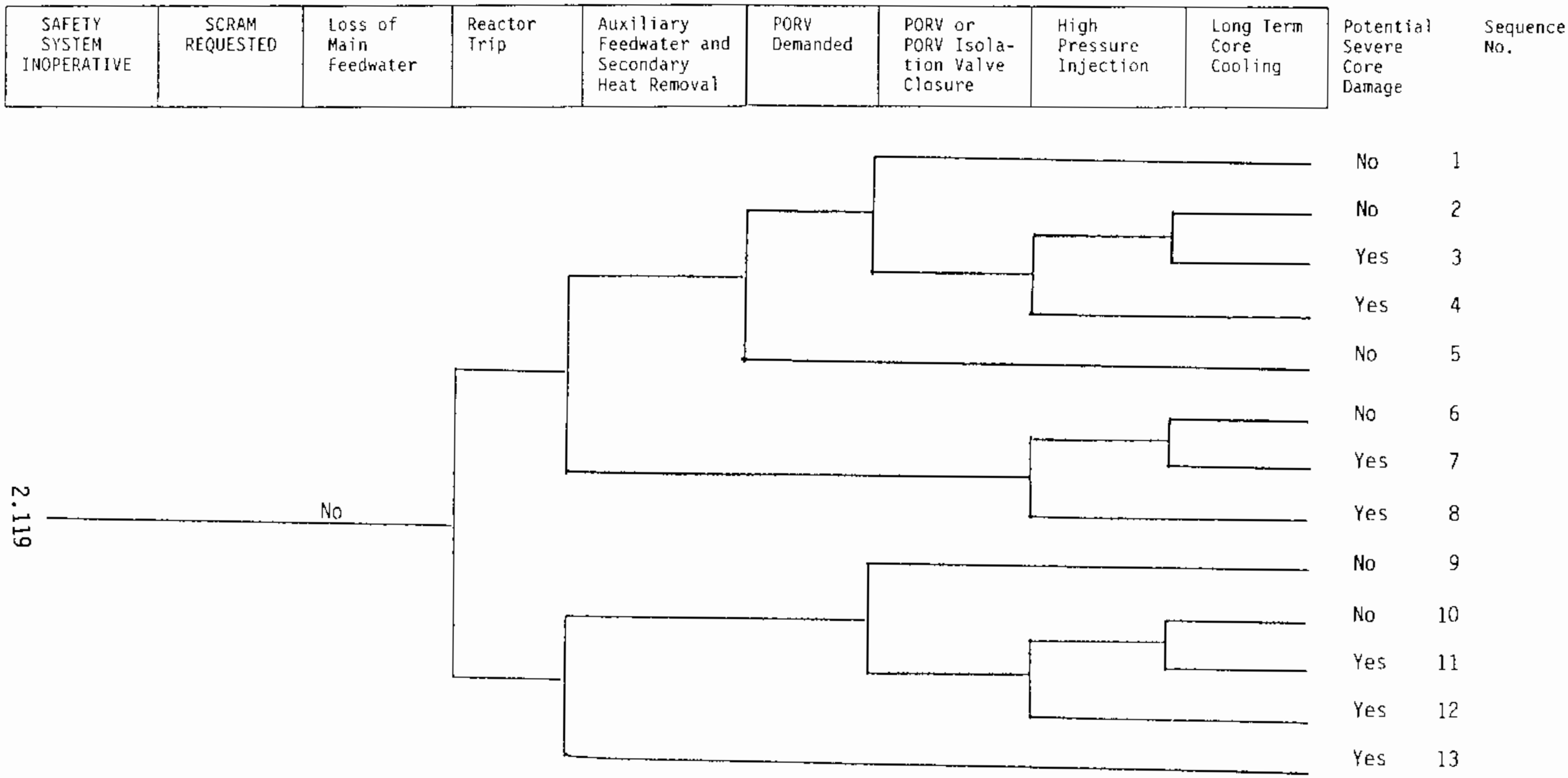


FIGURE 4.b. Safety System Inoperative, No Shutdown Requested

The analysis again hinges on the probability of inducing a feedwater transient on shutdown. As with the BWR analysis, the PWR transient categories 21 (feedwater instability on startup or shutdown, operator induced), and 27 (feedwater instability on startup or shutdown, mechanical causes) are used. The sum of these two categories is 0.36/py. Again, assuming that 50% are during shutdown, and 50% are loss of function, the estimated frequency for loss of feedwater or shutdown is put at 0.09/py. Again using the number of PWR scrams/yr (~10) as a measure of the number of shutdowns, the estimate of the probability of feedwater failure on shutdown is put at 0.009.

To estimate the probability of feedwater failure during an ECCS subsystem outage, again a one day failure duration is assumed. From the ATWS PWR category 16 (loss of feedwater), the frequency of failure is 0.15/py. Over a one hour period, the probability of failure is then

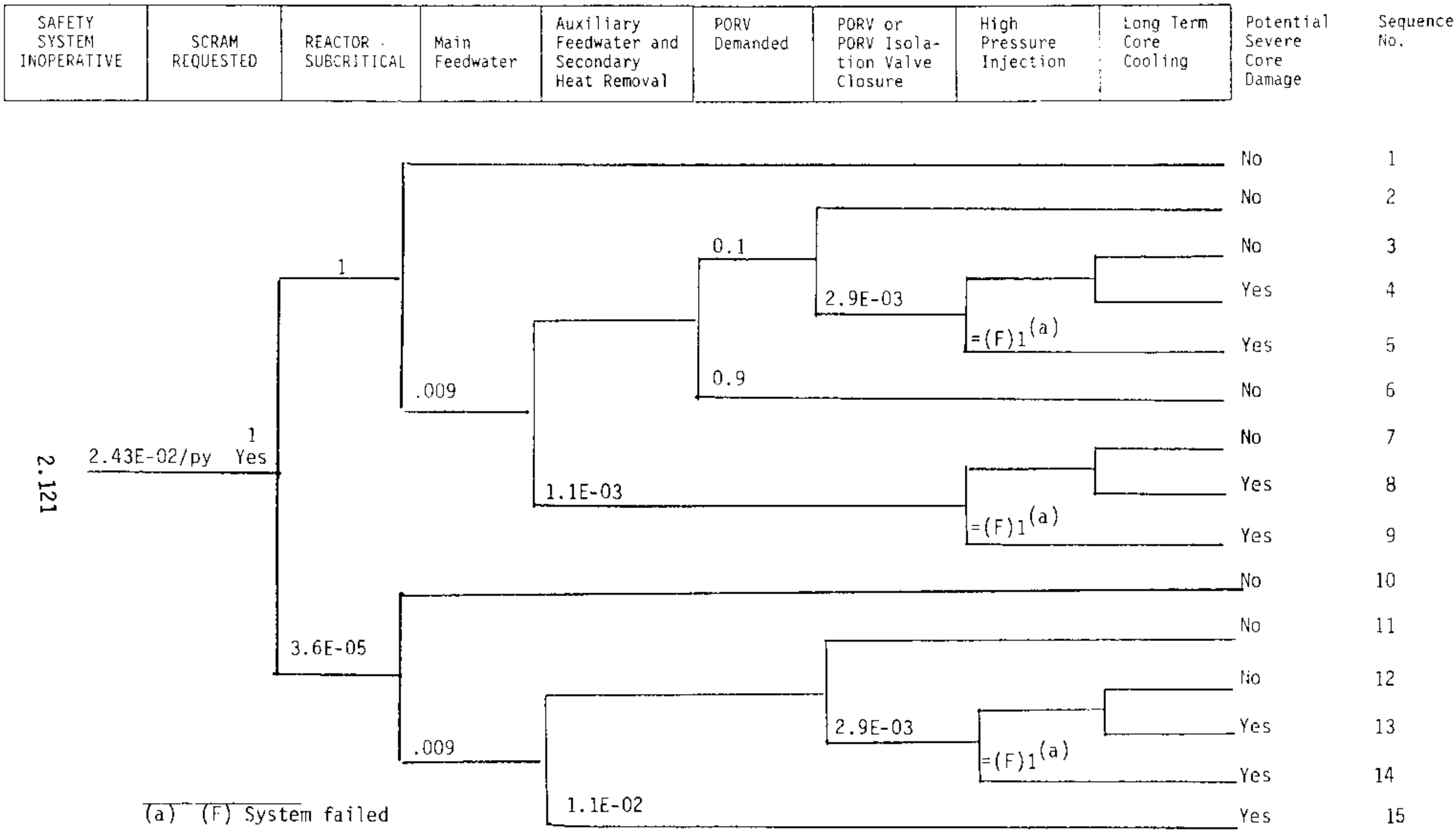
$$p = 1-e^{-t} \qquad t = (0.15/py)(1/365\ py)$$
$$= 0.0004.$$

The appropriate data have been entered on Figures 5a and 5b. The results are summarized below for the base where the high pressure injection system (HPI) is assumed to be inoperable. This simulated the incident at the McGuire plant on February 12, 1982.

HPI Failure

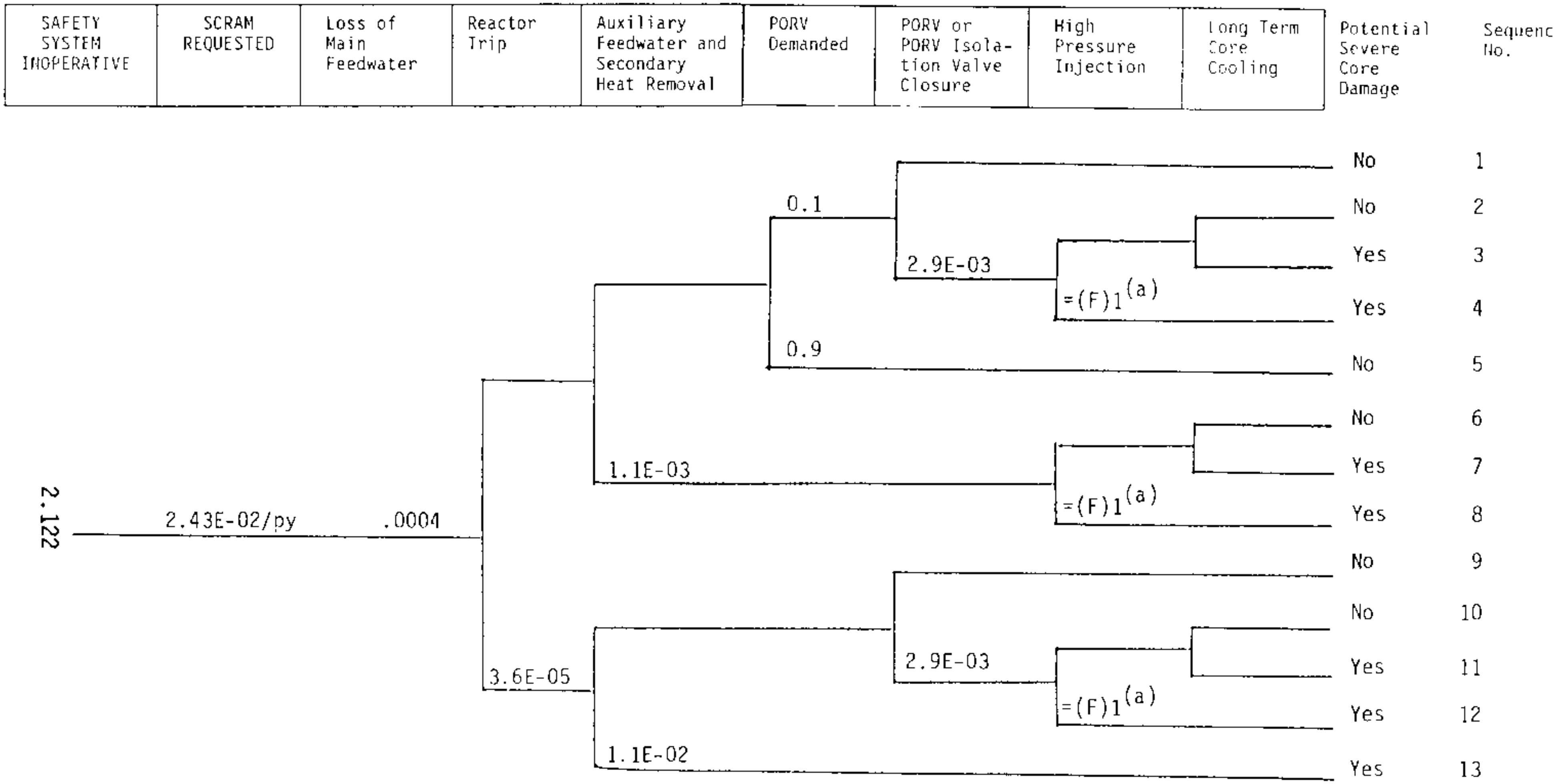
a. Shutdown Requested (base case)

Core-Damage Sequence	Frequency (1/py)
5	6.34E-08
9	2.41E-07
14	2.28E-11
15	8.66E-11
	3.05E-07



(a) (F) System failed

FIGURE 5.a. HPI Failed, Shutdown Requested



(a) (F) System failed

FIGURE 5.b. HPI Failed, No Shutdown Requested

ATTACHMENT 1. (cont'd.)

b. No Shutdown Requested (adjusted case)

Core Damage Sequence	Frequency (1/py)
4	2.87E-09
8	1.07E-08
12	1.01E-12
13	3.85E-12
	1.35E-08

Base Case Core Damage

HPI Failed, Shutdown Requested = 3.05E-07/py

Adjusted Case Core Damage

HPI Failed; No Shutdown Requested = 1.35E-08

Again to equate the measure of core damage used in the Precursor Study with the core-melt frequency used in other studies, the above frequencies are divided by a factor of 30. It is, also assumed that other sequences exist for PWRs. As with the BWR case, the change in a core-melt frequency predicted will be multiplied by a factor of 2, giving the following values:

PWR Base Case Core-Melt Frequency = 2.03E-08/py

PWR Adjusted Case Core-Melt Frequency = 9.00E-10/py

ΔPWR Core-Melt Frequency = 1.94E-08

Likewise,

$$(W_o)_{PWR} = (2.03E\text{-}08/py)(2.52E{+}06 \text{ man-rem}) = 5.12E\text{-}02 \text{ man-rem/py}$$

$$(W^*)_{PWR} = (9.00E\text{-}10/py)(2.52E{+}06 \text{ man-rem}) = 2.27E\text{-}03 \text{ man-rem/py}$$

$$\Delta(W)_{PWR} = 4.9E\text{-}02 \text{ man-rem/py}$$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Technical Specifications for Plant Shutdown when Safety Equipment is Inoperable (59)

2. Affected Plants (N):

	N
PWR	90
BWR	44

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

	$\bar{T}$
PWR	28.8
BWR	27.4

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

$\Delta(\bar{F}D_R)$ = 1.99E-03 man-rem/py (BWR), and 3.90E-04 man-rem/py (PWR)

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
3.41	44	0

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

The occupational dose increase for SIR implementation is zero because the implementation will consist of modifications to the operating plant TS.

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

Areas inside reactor containment containing the ECCS equipment (i.e., the RCICS room, HPCIS room etc.) are typically designated as "type B" (<2.5 mR/hr) or, at most, "type C" (<15 mR/hr) radiation zones, with limited access of 40 hr/wk and 6-40 hr/wk, respectively. It is assumed that dose rates due to maintenance during operation are 2.5 mR/hr higher than if the same work was performed during shutdown.

From Attachment 1, the frequencies of HPCIS/RCICS and ADS failures are estimated to be 0.049/py and 0.027/py, respectively, or a total of

0.076/py. Doubling this to account for other possible scenarios (as done for the core-melt frequency estimation in Attachment 1) gives an occurrence frequency of 0.15/py. Assuming that 24 maintenance hours are required per occurrence gives an estimate of (24)(0.15) of 4 man-hr/py of work in the containment during plant operation. This applies to PWRs as well.

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

    D_O = (0.0025 R/hr)(4 man-hr/py) = 0.010 man-rem/py

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

    $N\bar{T}D_O$ = (134)(28.3 yr)(0.010 man-rem/py) = 38 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
38	110	13

3.0 SAFETY ISSUE COSTS

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

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

    Technical Specifications for Plant Shutdown when Safety Equipment is Inoperable (59).

2. Affected Plants (N):

    All 134 LWRs, 71 operating and 63 planned.

3. Average Remaining Lives of Affected Plants (T):

	T (yr)	N
PWR	28.8	90
BWR	27.4	44

TABLE 3. (cont'd.)

Industry Costs (Steps 4 through 12)

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

$$\Delta(\bar{F}) = (\$1.65E+09)(1.0E-07/py) = \$165/py \text{ (BWR)}$$

$$= (\$1.65E+09)(1.94E-08/py) = \$32/py \text{ (PWR)}$$

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

Best Estimate	Upper Bound	Lower Bound
\$2.8E+05	\$3.6E+06	0

6. Per-Plant Industry Resources for SIR Implementation:

Cost is estimated directly in the next step.

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

A. For all LWRs with Operating Licenses:

The 71 operating LWRs are assumed to require supporting analysis, put at \$100,000/plant or \$7.1E+06. In addition, they will require a Class III amendment to their operating licenses, costing \$4000/plant. No additional hardware or training costs are foreseen beyond current expenditures.

B. For all LWRs with Construction Permits:

No additional costs are foreseen for the remaining 63 plants.

Thus, I = \$104,000/plant (operating only)

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

NI = (71)(\$104,000/plant) = \$7.38E+06

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

No additional industry labor is foreseen for this SIR. However, a decrease in outage time is expected to result since the TS change no longer necessitates plant shutdown. This decreased outage time will occur at a rate of (0.15 outage/py)(1 day/outage) = 0.15 day/py, where the outage frequency is calculated as in Step 9 of Table 2. Thus, a reduction of 0.15 day/py of outage time is assumed to result from the SIR.

TABLE 3. (cont'd.)

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

I_O = (-0.15 day/py)($3.0E+05/day) = -$4.5E+04/py

(Negative sign indicates cost savings.)

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

$N\bar{T}I_O$ = (134)(28.3 yr)(-$4.5E+04/py) = -$1.71E+08

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
-$1.6E+08	-$7.4E+07	-$2.5E+08

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

This issue will require considerable analysis to further quantify safety benefits associated with modified TS. As a first estimate, 3.5 man-yr of combined NRC and contractor effort is assumed.

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

C_D = (3.5 man-yr)($1.0E+05/man-yr) = $3.5E+05

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

A minimal amount of NRC labor, 0.5 man-day/plant, is presumed necessary to review each operating plant's TS modifications. For planned plants, these changes will be incorporated during the initial writing of the TS, so no additional NRC review is anticipated.

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

C = (0.5 man-day/plant)(1 man-wk/5 man-days)($2270/man-wk)
= $227/plant (operating only)

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

NC = 71($227/plant) = $1.61E+04

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

No additional NRC labor beyond current levels is foreseen. Thus, C_O = 0.

TABLE 3. (cont'd.)

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$3.7E+05	\$5.4E+05	\$1.9E+05

REFERENCES (For Issue 59)

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

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 60, Lamellar Tearing of Reactor Systems Structural Supports

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Concern exists over the potential for failure of PWR steam generator and reactor coolant pump supports during a seismic event due to lamellar tearing. The proposed resolution consists of two parts: 1) inspect all susceptible welds of this type using radiography or ultrasonic inspection techniques (including areas outside the weld heat-affected zone, where the phenomenon normally exists); and 2) repair/replace defective supports which are found as a result of inspection.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 2.3E+03

OCCUPATIONAL DOSES:

SIR Implementation =	710
SIR Operation/Maintenance =	0
Total of Above =	710
Accident Avoidance =	18

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:

SIR Implementation =	7.9
SIR Operation/Maintenance =	0
Total of Above =	7.9
Accident Avoidance =	1.5

NRC COSTS:

SIR Development =	0.22
SIR Implementation Support =	0.31
SIR Operation/Maintenance Review =	0
Total of Above =	0.52

LAMELLAR TEARING OF REACTOR SYSTEMS STRUCTURAL SUPPORTS

ISSUE 60

1.0 SAFETY ISSUE DESCRIPTION

Lamellar tearing is a material-cracking phenomenon which occurs beneath welds. The tearing occurs in the parent metal and generally lies parallel to the weld fusion boundary. Lamellar tearing is initiated by the decoherence or cracking of elongated inclusions. This causes voids to form and coalesce by the plastic tearing of the metallic matrix. This phenomenon is a ductile failure of the parent metal and occurs while the metal is cooling after the welding process. The tearing relieves the internal stresses caused by that process. Factors which can affect lamellar tearing susceptibility include parent materials, plate thickness, weld-bead geometry, electrode materials, joint geometry, the welding process, stress relief, and post-weld testing.

The primary difficulty with lamellar tearing is that, since it is a subsurface flaw condition, it is virtually impossible to detect by visual means. Therefore, a welded joint which appears to be unflawed and continues to perform its intended function may indeed be weakened due to the lamellar tear. For example, the welded supports for such items as the steam generator or reactor coolant pump may continue to support their loads even though they contain these lamellar tears. Thus, the tears do not represent a fracture of the joint but rather a reduction in the maximum load the joint can withstand. Under emergency loading conditions (such as a seismic event) the welded joints may fail causing a loss of function of the particular reactor system.

SAFETY ISSUE RESOLUTION

The proposed resolution of the lamellar tearing issue consists of two parts: 1) examine the steam generator and reactor coolant pump supports for lamellar tears and 2) repair/replace supports found to be defective. Radiography of the welds is already a required quality control practice, but it is not known whether these radiographs are extended to the locations outside the heat-affected zone of welds that are affected by the lamellar tearing phenomenon. Thus, Part One of the proposed safety issue resolution (SIR) is to perform radiography, ultrasonic inspection, or some other examination technique to identify structures affected by lamellar tears. Other methods to evaluate lamellar tearing were discussed in NUREG-0577 (NRC 1979). This report also identified the types of welded joints that are most susceptible to this phenomenon. These joint types could be avoided in future power plant construction.

AFFECTED PLANTS

As developed in Attachment 1 to the Public Risk Reduction Work Sheet, Part Two of the proposed issue resolution assumes that half of all PWRs are found to have defective supports as a result of the examination. Repair/ replacement is assumed to take place at these plants. SIR (both parts) is further assumed to be a one-time effort at the beginning of plant life for planned plants or at the next occurring outage for operating plants.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with resolution of this safety issue (60) 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:

   Lamellar Tearing of Reactor Systems Structural Supports (60)

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

   Half of all operating and planned PWRs are assumed to be affected. Thus:

	N	T/(yr)
Operating PWRs	24	27.7
Planned PWRs	21	30
All PWRs	45	28.8

3. Plants Selected for Analysis:

   Indian Point Unit 2 (IP2)

4. Parameters Affected by SIR:

   Steam generator (SG) failure (reactor coolant pump failure is not found in the Boolean expression for seismic core-melt frequency; see Attachment 1).

5. Base-Case Values for Affected Parameters:

   The base-case value of the affected parameter (SG) cannot be easily determined from the data contained in the IP2 study (PASNY 1982). However, the approach taken in this analysis does not require an explicit failure rate for SG failure. The lamellar tearing phenomenon is assumed to be important only during a seismic event of sufficient magnitude to cause

(a) See Attachment 1

TABLE 1. (cont'd.)

the SG supports to fracture. Therefore, the affected parameter is assumed to impact only the seismic contribution to the mean core-melt frequency. See Attachment 1 for further details.

6-7. These steps (and Steps 10 through 12) are omitted since the affected core-melt frequency is estimated directly.[a]

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

    1.4E-06/py

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

    3.5 man-rem/py

10-12. These steps are omitted. (See previous discussion in Steps 6 and 7.)

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

    7.0E-07/py

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

    1.8 man-rem/py

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

    7.0E-07/py

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

    1.8 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.3E+03	1.4E+05	0

(a) See Attachment 1

ATTACHMENT 1 (To Table 1)

Resolution of Issue 60 uses the Indian Point 2 (IP2) probabilistic risk assessment (PASNY 1982) due to its detailed assessment of seismic initiating events. Estimates of the affected public risk and affected core-melt frequency reductions are based on results of this study.

It is assumed that the reductions in affected public risk and core-melt frequency can be estimated as reductions in the seismic contribution to the total IP2 risk. The mean core-melt frequency due to seismic events at IP2 is 1.4E-04/py (PASNY 1982). There are a total of 9 minimal cut sets in the risk equation for the seismic core-melt frequency (denoted Ms), of which one cut set is dominant. A second cut set is roughly 100 times less likely than the dominant cut set and the remaining 7 contribute the balance. Among the remaining 7 is one cut set that includes failure of the steam generator. The IP2 risk study indicates that the critical item relating to SG failure during a seismic event is failure of the supports. Therefore, the SG failure rate is assumed conservatively to be entirely due to support failure; and, further more, the dominant failure mode for the SG supports is assumed to be lamellar tearing. These assumptions result in the following calculations:

$$\begin{aligned}
X &= \text{dominant minimal cut set} \\
Y &= \text{second most dominant minimal cut set} \\
Z &= \text{minimal cut set containing SG failure} \\
w_1, w_2, \ldots, w_6 &= \text{remaining six minimal cut sets} \\
y &= 0.01x
\end{aligned}$$

$$\left.\begin{array}{c} z \\ w_1 \\ w_2 \\ \cdot \\ \cdot \\ \cdot \\ w_6 \end{array}\right\} \leq (y = 0.01x)$$

Therefore, the following is true:

$$\begin{aligned}
M_s &= x + 0.01x + \alpha(0.01x) \\
&\leq 1.08x
\end{aligned}$$

where a $\leq$ 7

Thus, $x \geq M_s/1.08$. Assuming $x = M_s$ implies the following:

$$z \leq (0.01x \quad 0.01M_s \quad 1.4E\text{-}06/py)$$

This equation expresses the base-case, affected core-melt frequency due to cut set which contains SG failure.

The release categories defined in the IP2 study are considerably different from those normally used in issue analysis. The only release category contributing significantly to IP2 risk for this issue is defined below:

2RW: Applies to late overpressure failures of containment without functional sprays.

Other release categories do not contribute significantly to the public risk and will not be considered in this analysis. Therefore, the base-case, affected public risk is assumed to be entirely due to release category 2RW.

The IP2 study does not give an average public dose factor. This value is estimated for this issue as follows. The Oconee PRA, summarized in Andrews et al. (1983), will be used to calculate the average public dose factor because the IP2 plant is in a high population density area. Thus, the risk reduction results will be artifically high if IP2 is used. It is believed that Oconee is more representative of the "average plant" than IP2. The average public dose factor is calculated by first summing the products of the core-melt frequency and dose factors for each PWR release category to obtain a value for the total public risk due to Oconee. This value is divided by the sum of the core-melt frequencies to obtain a value for the average public dose factor, R. The value obtained by this procedure is R = 2.5E+06 man-rem. Thus, the base-case affected public risk, W, is equal to 1.4E-06/py (F) times 2.5E+06 man-rem (R), or 3.5 man-rem/py.

It is assumed that resolution of this issue has a relatively small impact on the likelihood of a core-melt accident due to a seismic event that involves SG failure. This assumption is largely due to the fact that no failures of this type have been documented in a nuclear power plant. Also, the steam generator is one component that is designed to withstand a highly energetic earthquake (1.8 g of peak ground acceleration, PASNY 1982). For these reasons, only a 50% reduction in the base-case frequency of the cut set in volving SG failure is assumed (i.e., the fragility curves for the SG supports are assumed to be displaced to a point where the probability of failure during a 1.8 g of earthquake is reduced to one-half of the base-case value). This reduces the likelihood of occurrence of the accident sequence to an adjusted case, affected core-melt frequency (F*) of 7.0E-07/py. The adjusted-case affected public risk (W*) equals R times F*, or 1.8 man-rem/py.

The total public risk reduction, $(\Delta W)_{total}$, is calculated assuming conservatively that 50% of the operating PWRs and 50% of the PWRs under construction will require that some SG and reactor coolant pump supports be repaired/replaced as a result of the proposed inspection. This assumption is based on a review of operating PWR system supports performed by Sandia National Laboratories (NRC 1979) which resulted in classification of about 1/2 of the PWR plants reviewed into a category defined as most susceptible to low fracture toughness. Since many of the parameters that affect fracture toughness (such as parent material, post-weld heat treatment, and plate thickness) also affect lamellar tearing susceptibility, this correlation is assumed to be justifiable for the purposes of this study. Therefore,

$$(\Delta W)_{total} = 45\ (28.8\ \text{yr})(1.8\ \text{man-rem/py}) = 2.32\text{E+03 man-rem}$$

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Lamellar Tearing of Reactor Systems Structural Supports (60)

2. Affected Plants (N):

SIR involves two parts--inspection for lamellar tearing and repair/replacement when found. All 90 PWRs (47 operating and 43 planned) will inspect. Only half of these (24 operating and 21 planned) will presumably repair.

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

	$\bar{T}$(yr)
Operating PWRs	27.7
Planned PWRs	30
All	28.8

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

$\Delta(FD_R)$ = (19,900 man-rem)(7.0E-07/py) = 1.4E-02 man-rem/py

(This reduction will be realized only at the 45 PWRs effecting repair/replacement.)

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

Best Estimate (man-rem)	Error Bounds Upper	Error Bounds Lower
18	220	0

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

Occupational doses for implementation of the issue resolution is the sum of two terms, 1) dose received during inspection (N = 47 backfit plants) and 2) dose received during support repair for those plants with lamellar tear discoveries (N = 24 operating plants, see Attachment 1). For 1), it is assumed that 2 man-wk/plant will be required to inspect and test the SG and reactor coolant pump supports. For 2), it is assumed that 6 man-wk/plant will be required to replace the defective component supports, re-weld the joints, and re-inspect the welds. Assuming a 75% utilization factor for man-power in radiation zones results in the following:

TABLE 2. (cont'd.)

Inspection only: (0.75)(2 man-wk/plant)(40 man-hr/man-wk) = 60 man-hr/plant

Inspection and repair: 0.75(8 man-wk/plant)(40 man-hr/man-wk) = 240 man-hr/plant

No dose will be accumulated at planned plants since any inspection and repair/replacement will presumably occur at the beginning of plant life, i.e., prior to initial operation.

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

It is assumed here that radiation fields of 100 mR/hr exist in the SG and reactor coolant pump enclosures.

Inspection only (23 operating PWRs): D = (60 man-hr/plant)(0.1 R/hr) = 6.0 man-rem/plant

Inspection and repair (24 operating PWRs): D = (240 man-hr/plant)(0.1 R/hr) = 24.0 man-rem/plant

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

ND = (6.0 man-rem/plant)(23 PWRs) + (24 man-rem/plant)(24 PWRs)

= 710 man-rem

9-11. No work in radiation zones would be required for SIR operation and maintenance since it is a one-time effort. Therefore, $D_O = 0$.

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
710	2100	240

3.0 SAFETY ISSUE COSTS

The resolution of this safety issue is postulated in Section 1.0. It is assumed that a 2 man-year contractor study would be required to perform further analyses to identify acceptable and non-acceptable test results. In addition, it is assumed that it will require 1 man-wk of NRC staff labor per plant to transfer the results of the contractor study to the plant personnel. Industry and NRC costs are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Lamellar Tearing of Reactor Systems Structural Supports (60)

2. Affected Plants (N):

   SIR involves two parts--inspect supports for lamellar tearing and repair/replacement of supports found to be defective. All 90 PWRs (47 operating and 43 planned) will inspect. Only half of these (24 operating and 21 planned) will presumably repair.

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

	$\bar{T}$(yr)
Operating PWRs	27.7
Planned PWRs	30
All	28.8

Industry Costs (Steps 4 through 12):

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

   $\Delta(\bar{F}A)$ = (7.0E-07/py)($1.65E+09) = $1150/py

   (This reduction wil be realized only at the 45 PWRs effecting repair/replacement.)

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

Best Estimate	Upper Bound	Lower Bound
$1.5E+06	$1.8E+07	0

6. Per-Plant Industry Resources for SIR Implementation:

   Equipment: $50,000/plant for special radiography or ultrasonic NDT equipment (estimate)

   Labor: 2 man-wk/plant for inspection

   6 man-wk/plant for repair (if required)

TABLE 3. (cont'd.)

Additional down-time: 0 (issue resolution can be performed during a refueling outage for operating plants or during construction).

Materials and fabrication: \$50,000/plant for fabrication of new supports for discovered deficiencies.

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

Inspection only (45 PWRs): \$50,000/plant (equipment)

+ (2 man-wk/plant)(\$2270/man-wk)

= \$5.5E+04/plant

Inspection and repair (45 PWRs): \$5.5E+04/plant (inspection)

+ \$50,000/plant (materials)

+ (6 man-wk/plant)(\$2270/man-wk)

= \$1.2E+05/plant

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

NI = (\$5.5E+04/plant)(45 PWRs) + (\$1.2E+05/plant)(45 PWRs)

= \$7.9E+06

9-11. No operation and maintenance requirements are foreseen for this SIR since it is a one-time effort. Therefore, $I_O = 0$.

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$7.9E+06	\$1.2E+07	\$4.0E+06

NRC Costs (Steps 13 through 21):

13. NRC Resources for SIR Development:

NRC Staff Labor = 8 man-weeks

Contractor Support (see next step for estimates of contractor costs)

TABLE 3. (cont'd.)

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

Labor = (8 man-wk)($2270/man-wk) = 1.8E+04

Contractor Support = 2.0E+05

C_D = 2.18E+05

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

1 man-wk/plant to review inspection

1 additional man-wk/plant to review repairs on plants with detected lamellar tears on component supports.

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

Inspection only (45 PWRs): C = (1 man-wk/plant)($2270/man-wk)

= $2270/plant

Inspection and Repair (45 PWRs): C = (2 man-wk/plant)($2270/man-wk)

= $4540/plant

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

NC = (45 PWRs)($2270/plant) + (45 PWRs)($4540/plant)

= $3.06E+05

18-20. No review of SIR operation and maintenance is foreseen since it is assumed that SIR is a one-time effort for all plants. Therefore,

$C_o = 0.$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$5.2E+05	$7.1E+05	$3.4E+05

REFERENCES (for Issue 60)

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.

Power Authority of the State of New York (PASNY) and Consolidated Edison Co. of New York, Inc. 1982. Indian Point Probabilistic Safety Study. New York.

U.S. NRC. 1979. Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports, NUREG-0577 (DRAFT), U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 61, SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Failure of a BWR safety relief valve to reseat following a transient could overpressure containment if its discharge line should rupture in the wetwell air space above the suppression pool. The proposed resolution is automation of the system needed to mitigate the overpressure--the Containment Spray System--along with separation of its function from that of emergency core cooling.

AFFECTED PLANTS		
BWR:	Operating = 24	Planned = 10
PWR:	Operating = 0	Planned = 0

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	2,700
OCCUPATIONAL DOSES:	
SIR Implementation =	1,600
SIR Operation/Maintenance =	540
Total of Above =	2,100
Accident Avoidance =	8.3

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	34
SIR Operation/Maintenance =	2.1
Total of Above =	36
Accident Avoidance =	0.7
NRC COSTS:	
SIR Development =	0.10
SIR Implementation Support =	0.077
SIR Operation/Maintenance Review =	0.21
Total of Above =	0.40

SRV LINE BREAK INSIDE THE BWR WETWELL AIRSPACE OF MARK I AND II CONTAINMENTS

ISSUE 61

1.0 SAFETY ISSUE DESCRIPTION

The safety/relief valves (SRVs) of a BWR provide protection against over pressurization of the reactor primary system. During normal operation the SRVs which are mounted on the main steam lines open on high pressure, permitting steam to escape from the reactor vessel. SRV piping carries the steam through the drywell, into the wetwell, and discharges it into the suppression pool, thereby condensing the steam. Failure of the steam to condense eventually leads to a rupture of the containment boundary and loss of reactor coolant inventory.

This issue examines a postulated break in the SRV discharge line in the wetwell airspace above the suppression pool of Mark I and II plants. Coupled with the line break is a failure of the relief valve to close after its actuation in response to the transient. The relief valve must remain open for a significant amount of steam to escape, bypass the pool, and threaten overpressurization of the containment vessel with rupture in approximately ten minutes (Economos 1982).

At least two points of view emerge from a review of the available background information. One, based upon a risk assessment performed under the auspices of the NRC, indicates that transient-initiated accident sequences involving the failure of an SRV to reseat combined with the rupture of the associated discharge line can occur at frequencies comparable to or greater than those of many accident sequences now considered to be significant contributors to risk. It recommends that such accident sequences be included in ongoing or future studies of Mark I and II plants. It states that more quantifiable information is necessary to sufficiently characterize the consequences of such accident sequences and recommends three areas for further work at generic and plant-specific levels (Economos 1982).

According to the second point of view, existing piping requirements and proposals to upgrade piping classification are sufficient to ensure safety. The NRC Mechanical Engineering Branch (MEB), in recognition of the potential consequences of a line break, requires additional fatigue analysis (equivalent to that required for Class 1 piping) on the SRV system of Mark I, II, and III containment configurations. The MEB is also considering proposals to upgrade the classification of the BWR SRV system. In a memorandum from T. Murley to R. Vollmer (February 4, 1981), the Safety Program Evaluation Branch agreed with the following recommendations as presented in a memorandum by R. Vollmer (December 5, 1980):

[that] CP (construction permit) applicants upgrade the system to Quality Group B and that OL (operating license) applicants and operating plants upgrade only the inservice inspection requirements to those required in ASME Code Section XI for Class 2 (Quality Group B) systems. The major improvement to be gained from this reclassification is that it would assure volumetric examination rather than surface examination of welds in the system during construction. In addition, as a system classified Quality Group B, the inservice inspection program performed during the 40-year life of a plant in accordance with ASME Section XI would require selective volumetric examination of pipe welds and volumetric or surface examination of welds in other components depending on wall thickness and location. At present, Class 3 lines do not require such examination.

The primary benefit gained from reclassification would apply to plants requesting CPs. This issue affects plants already under construction and in operation. For these plants, the proposal would require that the inservice inspection program be consistent with Quality Group B requirements during plant life. Current requirements are for periodic visual inspection of the SRV discharge piping.

SAFETY ISSUE RESOLUTION

No resolution has been proposed for this issue, although several have been discussed. By analysis (Economos 1982), the Containment Spray System (CSS) has been shown to be effective in reducing containment pressures if actuated in a timely manner. Since the CSS is manually actuated and response time may be as short as ten minutes, automatic CSS actuation is seen as a desirable aspect of the resolution. The CSS is only one of the operating modes of the residual heat removal system, and priority is given to the emergency core cooling system (ECCS), another of its operating modes. Additionally, in some plants a lockout feature (which can be overridden) prevents operator actuation of the CSS for ten minutes or more to assure maintenance of the coolant inventory.

Therefore, modification of the CSS for automatic actuation and separation of the CSS from the ECCS is assumed as the safety issue resolution (SIR) for this analysis. Automatic CSS actuation effectively eliminates the possibility of operator error. Separation of the CSS and ECCS functions assures integrity of the ECCS.

AFFECTED PLANTS

This issue affects BWR plants with Mark I and II containments. There are 24 operating plants and 10 planned plants that are affected.

2.0 SAFETY ISSUE RISK AND DOSE

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

PUBLIC RISK REDUCTION

For the public risk reduction, the affected parameter is the CSS unavailability due to operator error. Attachment 1 summarizes the assessment which identified the accident sequence used in this public risk reduction analysis. Table 1 summarizes the results of this analysis.

OCCUPATIONAL DOSE

Additional radiation will be accumulated by personnel during installation of the required equipment and its operation and maintenance. Table 2 summarizes the results of the analysis.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments (61)

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

   All BWRs with Mark I and II containment configurations. The resolution is assumed to be implemented in 1983.

		N	$\bar{T}$ (yr)
BWRs:	Operating	24	25.2
	Planned	10	30.0
	All	34	26.6

3. Plants Selected for Analysis:

   The hypothetical plant in the alternate study is explained and referenced in Attachment 1.

4. Parameters Affected by SIR:

   Parameter Z in sequence TPDZ is identified from the alternate study and explained in Attachment 1.

Symbol	Description
Z	CSS Unavailability

TABLE 1. (cont'd.)

5. Base-Case Values for Affected Parameters:

$Z = 5.0E-01$

Manual actuation of CSS is dominated by operator reliability, and its successful actuation is as likely to occur as not (Economos 1982).[a]

6. Affected Accident Sequences and Base-Case Frequencies:

TPDZ - γ' (BWR-2) = 7.5E-07/py
γ (BWR-3) = 2.5E-07/py

The frequency of TPDZ = 4.6E-07/py[a] and falls into BWR release categories 2 and 3 via containment failure modes $\gamma' = 0.75$ and $\gamma = 0.25$. These values are based on a qualitative comparison of the consequences of the TPDZ sequence with accident sequences in WASH-1400 (1975) having containment failure by overpressure. The affected release categories in WASH-1400 are BWR-2 with $\gamma' = 0.225$ and BWR-3 with $\gamma' = 0.76$. Because rupture occurs about 10 minutes after reactor scram, the consequences are expected to be more severe than sequences involving failure of the containment where rupture occurs 20-25 hours after scram. Also since the suppression pool is bypassed, no scrubbing action is performed, further increasing its severity. Hence, it is assumed that $\gamma' = 0.75$ (containment failure due to overpressure with release to atmosphere) and $\gamma' = 0.25$ (containment failure due to overpressure caused by hydrogen burning).

7. Affected Release Categories and Base-Case Frequencies:

BWR-2 = 3.45E-07/py
BWR-3 = 1.15E-07/py

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

$\bar{F} = 4.6E-07/py$

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

W = 3.0 man-rem/py

(a) See Attachment 1.

TABLE 1. (cont'd.)

10. Adjusted-Case Values for Affected Parameters:

$Z^* = 5.0E-03$

Separation of the CSS from the ECCS and automatic actuation are assumed to reduce the probability of containment overpressurization by 99%. This is primarily due to reducing dependence on operator reliability.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

TPDZ - γ' (BWR-2) = 7.5E-09/py
γ (BWR-3) = 2.5E-09/py

12. Affected Release Categories and Adjusted-Case Frequencies:

BWR-2 = 3.45E-09/py
BWR-3 = 1.15E-09/py

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

$\bar{F}^* = 4.6E-09/py$

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

$W^* = 3.0E-02$ man-rem/py

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

$\Delta\bar{F} = 4.6E-07/py$

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

$\Delta W = 3.0$ man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.7E+03	8.1E+04	0

ATTACHMENT 1 (To Table 1)

The risk assessment previously referenced (Economos 1982) forms the basis of the public risk reduction calculations. Explanations are included when modifications to this methodology are made. Following is a summary of the analysis and findings. A complete description of the scope, method of approach, the major findings, and conclusions can be found in the original reports.

The method of approach is essentially that used in the Reactor Safety Study (WASH-1400 1975) with a more limited scope. The limitations are summarized here:

1) The accident sequences were qualitatively compared to a "release category." No actual consequence calculations were performed.

2) Only the most frequent anticipated transient was used as an initiating event.

3) Additional system failures were minimized.

4) Only one accident mitigating system was considered.

5) Evaluations were restricted to BWRs which already had available risk assessments.

SUMMARY OF KEY FINDINGS

The findings of this study fall into three areas. The first is the analysis of the containment pressure response due to an SRV discharge line failure and subsequent steam bypass. Results show that for a Mark I plant (Peach Bottom) and a Mark II plant (Limerick Generating Station) with 100% steam bypass (rupture of the line), design pressure is exceeded within four minutes and rupture pressure within about 10 minutes. For 50% steam bypass, the time is about 6 minutes and 20 to 30 minutes, respectively. For a Mark II plant, the CSS is shown to be very effective in mitigating the accident, provided that it is actuated in a timely manner. "Based on these results it is concluded that a stuck open SRV combined with a ruptured SRV discharge line inevitably will lead to containment rupture by overpressure unless the CSS is actuated within about 10 minutes of the transient initiator. Also, significant design pressure exceedance will occur unless the CSS is actuated within four minutes." (Economos 1982)

The second area of study is estimation of SRV discharge line failure rates by two different methods. They include an evaluation of operating experience and use of probabilistic mechanical design methods. The report concludes "that the most appropriate value to be used for the SRV discharge line failure rate is 7.4E-05 per demand with the recognition that it probably represents an upper bound rather than a best estimate or mean value." (Economos 1982)

Finally, the report examines the contribution to risk due to the accident by determining the frequency of its occurrence and estimating the resulting consequences. Figure 1 shows the event tree which was developed and evaluated.

> Following the initiating event (in this case an anticipated tran sient) the tree shows in columnar fashion the functions that need to be successfully performed to prevent an accident (degraded core condition) from occurring. It also shows the system (or systems) that can be exploited to perform these functions. The ordering of the functions is in total agreement with event trees previously developed with the single exception of the inclusion of the "limit containment pressure" function in a transient event tree. As indicated, this represents the point of departure for the present study. This function has been placed before the coolant and heat removal functions because its failure implies very early rupture of the containment which leads, with high probability, to failure to maintain coolant inventory and, therefore, to eventual core melt down. (Economos 1982)

As noted previously, only the more dominant sequences are considered. Their frequencies in functional form are shown in the last column of Figure 1. A simplification is that only the CSS is shown to be capable of limiting containment pressure if the Vapor Suppression System (VSS) ruptures via SRV discharge lines. The symbol Z represents failure of the CSS.

DESCRIPTION OF ACCIDENT SEQUENCES

A description of each of the accident sequences considered is given below.

The TPD Sequence

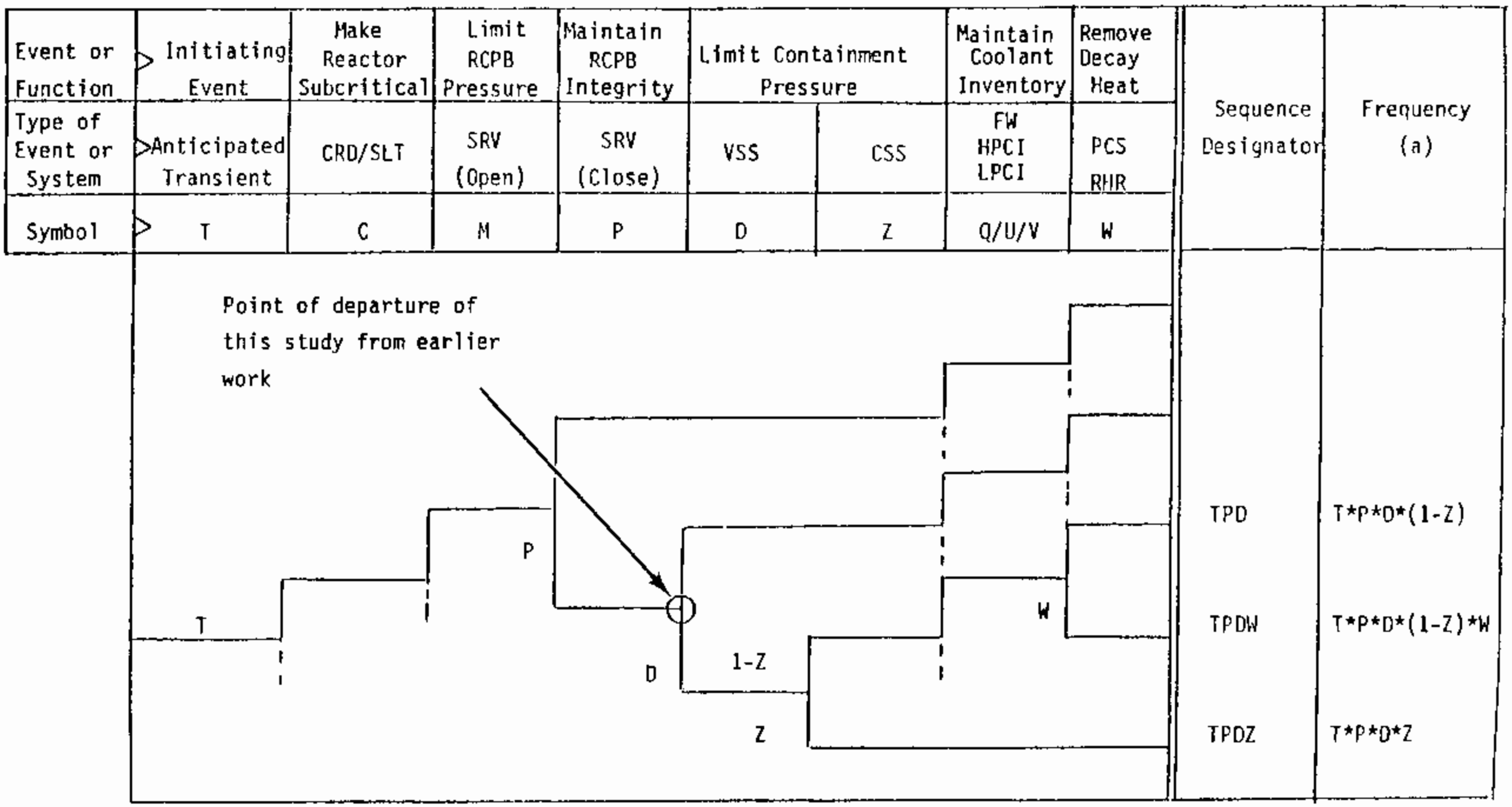
The initiating event (T) is a turbine trip with bypass followed by successful scramming of the reactor and normal actuation of the SRVs to limit reactor pressure. It is further postulated that, after reaching set point levels, one SRV fails to reseat (Event P) and the associated SRV discharge line ruptures, allowing steam to bypass to the wetwell airspace (Event D). The CSS is assumed to be activated before containment rupture, but after design pressure is exceeded, so some fission products are released.

The TPDW Sequence

This sequence is the same as the above but with failure to remove residual heat from containment (Event W), leading to high containment pressure and failure to maintain coolant inventory, leading to a core meltdown.

The TPDZ Sequence

This sequence postulates that the CSS is not activated before the con tainment ruptures, leading with high probability to degraded core conditions.



(a) The symbols T, P, etc. are used to designate both the event and its frequency.

FIGURE 1. Simplified Transient Event Tree Used in Present Study (Economos 1982)

FREQUENCIES AND PROBABILITIES OF EVENTS

The frequencies and probabilities of the events in the accident sequences are estimated below. The general approach is taken from Economos 1982. Specific event probabilities were altered in an effort to enhance the accident sequences logic.

Frequency of Event T

The anticipated transient (T) is turbine trip with bypass. This was considered to be the transient with the highest frequency of occurence (Economos 1982). This frequency is assumed to be 4 events per reactor year (EPRI 1978).[a] Inherent in this frequency is the number of valves actuated given that a transient has occurred. Therefore, N will represent the number of valves actuated by the transient. For the purpose of this analysis we assume on the average that one valve is demanded open per transient. Thus, the event in question becomes T * 1/N = 4/py.

Probability of Event P

The failure probability for event P is taken as the probability that one valve fails given that N valves were actuated by the transient. The probability of a valve failing to reseat based on a study performed by EG&G for the NRC (Miller, et al. 1980) is 3.1E-03. The event probability is also based upon the number of valves actuated and thus, the event probability becomes P*N or 3.1E-03*N.

Probability of Event D

Failure of the VSS occurs if the SRV discharge line associated with the open SRV ruptures. Given the failure of the SRV to reseat, the probability that the associated discharge line fails is the random pipe demand failure rate of 7.4E-05 (Economos 1982).

Probability of Event Z

The failure probability for Event Z is taken as Z = 5.0E-1 per demand. The report considered that the Z failure has occurred if the CSS has not been actuated within 10 minutes of the occurrence of the transient. This is a very short time in terms of operator reliability, particularly when it is recognized that the prevailing conditions are "stressful." (Containment pressure exceeding design limits within 4 minutes after the transient is judged to be "very stressful.") Under such conditions, the WASH-1400 estimate that the operator would fail to act correctly (here, actuate the CSS) lies somewhere

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(a) See also "Assessment of Transient Frequencies Based on BWR Operating Experience," an unpublished evaluation by General Electric Company submitted to the NRC (no date, referenced by Economos 1982).

between a 10 to 90% probability. There is also indication that in some plants actuation of the CSS under the condition of high drywell pressure is precluded for a period of ten minutes or more due to the so-called "lockout" feature. This logic is introduced to assure that all low pressure coolant injection pumps remain dedicated to maintaining coolant inventory.

Based on all of these considerations, manual actuation of the CSS is believed to be dominated by operator reliability, and its successful actuation is as likely to occur as not. Accordingly, the failure rate for the Event Z is assigned the value 5.0E-01 per demand (Economos 1982).

Probability of Event W

The failure probability for event W is taken as W = 1.5E-03. After qualitatively comparing features of the TPDW sequence with conditions during a LOCA, the authors judged, "The unavailability of the heat removal systems in the TPDW sequence is not likely to exceed the unavailability during a LOCA by more than an order of magnitude. The WASH-1400 value for this event was 1.25E-04 while, for the Limerick plant, it is taken as 1.6E-04. Accordingly, for the present study, we take W = 1.5E-03." (Economos 1982)

The frequencies of the dominant accident sequences identified in the report are summarized below.

$$f(TPD) = T \cdot P \cdot D \cdot (1-Z) = 4.6E\text{-}07/py$$
$$f(TPDW) = T \cdot P \cdot D \cdot (1-Z) \cdot W = 6.9E\text{-}10/py$$
$$f(TPDZ) = T \cdot P \cdot D \cdot Z = 4.6E\text{-}07/py$$

These are compared with sequences considered to be dominant risk contributors in WASH-1400 and the Limerick study. Their frequencies are shown below and comparisons are discussed further.

This Study (Economos 1982)	WASH-1400 (1975)	Limerick Study (Philadelphia Electric Co. 1982)
TPD = 4.6E-07/py	A = 1.0E-04/py	TPW = 4.0E-07/py
TPDZ = 4.6E-07/py	TW = 8.0E-06/py	TQUV = 3.0E-08/py
TPDW = 6.9E-10/py	TQUV = 2.0E-07/py	AJ = 6.4E-08/py
	AJ = 1.3E-08/py	TQW = 3.0E-09/py

Qualitatively, this study considers the consequences of the three accident sequences. Sequence TPD postulates some radioactive leakage out of containment because design pressure is exceeded. This sequence is compared to the A sequence of WASH-1400 whose consequences are considered relatively minor. Since the TPD sequence frequency is more than two orders of magnitude less, it is not judged a significant contributor to risk.

ATTACHMENT 1. (cont'd.)

The TPDW sequence has the following features in common with the TW, TPW and TQW accident sequences--all include successful scrams and all include containment failure by overpressure prior to core melt (the W failure). On this basis, one would expect the consequences for all to be essentially the same. One difference, however, can be cited which would indicate that the TPDW sequence may have more severe consequences. This relates to the fact that, due to the steam bypass, some percentage of the fission product released during core melt will not experience the scrubbing action provided by passage through the suppression pool. Since the consequences could be more severe and the frequency of this sequence is comparable to that of the TQW sequence, it is concluded that the TPDW sequence may represent a significant contributor to risk. A more detailed examination of the effects of exceeding design pressure (and temperature) and flooding effects on the heat removal systems' unavailability is needed to arrive at a more definite conclusion.

The TPDZ sequence is judged to contribute significantly to risk in all of the more severe release categories defined in WASH-1400 (Economos 1982). This is based on its relatively high frequency and the short time in which containment rupture occurs after reactor scram. The consequences are thought to be more severe than the sequences involving the W event where containment rupture does not occur until 20-25 hours after scram. Also, the suppression pool is bypassed and no scrubbing action is provided.

Based on these considerations, it is concluded that sequence TPDZ dominates the others with respect to risk. Therefore, calculations for public risk reduction are based solely on this sequence for the purpose of providing estimates for the SIR.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments (61)

2. Affected Plants (N):

34 BWRs (24 operating and 10 planned)

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

		$\bar{T}$ (yr)
BWRs:	Operating	25.2
	Planned	30.0
	All	26.6

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

$$\Delta(\bar{F}DR) = (19{,}900 \text{ man-rem})(4.6E-07/py)$$
$$= 9.2E-03 \text{man-rem/py}$$

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
8.3	49.7	0

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

Implementation of the SIR would require a new containment penetration for a suction line, pump and motor installation, routing pipe for suction and discharge lines and connection to the existing containment spray header, and installation of circuitry for its automatic actuation. Assurance of ECCS integrity is assumed to be best achieved through a separate suction line for the CSS. The following is a best estimate of labor hours required as obtained through PNL contacts with reactor personnel:

• Containment penetration	24 man-wk
• Pipe routing, radiography	72 man-wk
• Pump installation	8 man-wk
• Instrumentation	4 man-wk
	108 man-wk/plant (operating BWRs)

(108 man-wk/plant)(40 man-hr/man-wk) = 4320 man-hr/plant

TABLE 2. (cont'd.)

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

It is assumed that radiation fields of 15 mR/hr are encountered in the pump rooms and reactor enclosure (Grand Gulf PSAR).

$$D_{BWR} = (4320 \text{ man-hr/plant})(0.015 \text{ R-hr}) = 65 \text{ man-rem/plant}$$

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

$$ND = (24 \text{ operating plants})(65 \text{ man-rem/plant}) = 1560 \text{ man-rem}$$

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

It is assumed that one additional man-week per plant-year will be required for examination of equipment installed in the reactor enclosure. This applies to 34 BWRs.

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

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

Again, a 15 mR/hr radiation field is assumed.

$$D_O = (40 \text{ man-hr/py})(0.015 \text{ R/hr}) = 0.60 \text{ man-rem/py}$$

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

$$N\bar{T}Do = (0.60 \text{ man-rem/py})(34)(26.6 \text{ yr}) = 540 \text{ man-rem}$$

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.1E+03	6.3E+03	7.0E+02

3.0 SAFETY ISSUE COSTS

Results of NRC and industry cost calculations are presented in this section. Best estimates were used for labor time required for the resolution. Table 3 summarizes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments (61)

2. Affected Plants (N):

   34 BWRs (24 operating and 10 planned)

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

		$\bar{T}$ (yr)
BWRs:	Operating	25.2
	Planned	30.0
	All	26.6

Industry Costs (Steps 4 through 12)

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

$$\begin{aligned}\Delta(\bar{F}A) &= (4.6E\text{-}07)(\$1.65E+9) \\ &= \$7.6E+02/py\end{aligned}$$

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

Best Estimate	Upper Bound	Lower Bound
6.9E+05	$4.1E+06	0

6. Per-Plant Industry Resources for SIR Implementation:

   The labor costs are best estimates as obtained through PNL contacts with reactor personnel.

   For operating BWRs:

   - Labor = 108 man-wk/plant
   - Analysis, Scheduling, Planning, QA = 285 man-wk/plant
   - Replacement Power = None
     (assume work done during scheduled outages)
   - Equipment (cost estimated in next step)

   For planned BWRs:

   Labor is assumed ~20% lower; i.e., (0.80)(108) = 86.4 man-wk/plant, due to implementation in construction phase. Engineering analysis, scheduling, planning and QA are assumed to be the same,

TABLE 3. (cont'd.)

i.e., 285 man-wk/plant. Equipment costs are assumed similar to operating plants (and are estimated in the next step).

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

For operating BWRs:

Labor	= (108 man-wk/plant)($2270/man-wk)	= $2.5E+05
Analysis, etc.	= (285 man-wk/plant)($2270/man-wk)	= 6.5E+05
Equipment		= 1.2E+05
	I	= $1.02E+06/plant

For planned BWRs:

Labor	= (86.4 man-wk/plant)($2270/man-wk)	= $2.0E+05
Analysis, etc.	= (285 man-wk/plant)($2270/man-wk)	= 6.5E+05
Equipment		= 1.2E+05
	I	= $9.7E+05/plant

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

NI = (24)($1.02E+06/plant) + (10)($9.7E+05/plant)

= $3.42E+07

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

Equipment maintenance estimate of sustaining labor is 1 man-wk/py.

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

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

= $2.27E+03/py

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

($N\bar{T}I_O$) = (34)(26.6 yr)($2.27E+03/py)

= $2.05E+06

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$3.6E+07	$5.3E+07	$1.9E+07

TABLE 3. (cont'd.)

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Assuming that little resolution development has been performed, NRC staff labor = 1 man-yr.

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

$$C_D = (1 \text{ man-yr})(\$1.0E+05/\text{man-yr})$$

$$= \$1.0E+05$$

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

This is assumed to be 1 man-wk/plant.

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):

$$NC = (34)(\$2270/\text{plant})$$

$$= \$7.72E+04$$

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

0.1 man-wk/py

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

$$Co = (0.1 \text{ man-wk/py})(\$2270/\text{man-wk})$$

$$= \$227/\text{py}$$

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

$$N\bar{T}Co = (34 \text{ plants})(26.6 \text{ yr})(\$227/\text{py})$$

$$= \$2.05E+05$$

TABLE 3. (cont'd.)

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$3.8E+05	\$5.0E+05	\$2.6E+05

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USE OF EQUIPMENT NOT CLASSIFIED AS ESSENTIAL TO SAFETY IN BWR TRANSIENT ANALYSIS

ISSUE 63

ISSUE BACKGROUND

This issue addresses BWR equipment classified as "non-safety grade" and the acceptability of such equipment for use in the analysis of anticipated operational transients. These analyses are used to establish the thermal operating limits based on changes in the critical power ratio (CPR). The limits in part, minimize radioactive release during various modes of plant operability.

> "Thermal limits are provided for normal operation and transient events to maintain the integrity of the fuel cladding. The objective is achieved by limiting the fuel rod power density to avoid overstressing the fuel claddings because of fuel pellet-cladding differential expansion and by maintaining nucleate boiling around the fuel rods so that transition to film boiling is avoided." (USNRC RTC, 1982)

One of the established thermal limits is the minimum critical power ratio (MCPR).

The CPR is the ratio of fuel bundle power at which departure from nucleate boiling occurs to the actual bundle power. It serves as a measure or indication of how close a fuel bundle is to transition boiling.

MCPR

The MCPR, representing the fuel bundle which is the closest to transition boiling, is a limit imposed to avoid fuel damage as a consequence of clad overheating and is modified to account for transients. By setting the MCPR limit at 1.07, it is anticipated that 99.9% of the rods avoid transition boiling.

> The required MCPR at steady-state operating conditions is derived from the established fuel cladding integrity safety limit MCPR of 1.07, and an analysis of abnormal operational transients. To ensure that the fuel cladding integrity safety limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine those which result in the largest reduction in critical power ratio (ΔCPR). The types of transients evaluated include turbine trip without bypass valves; generator load rejection without bypass valves; feedwater controller failure; pressure regulator failure downscale; loss of feedwater heating; fuel loading error; and rod withdrawal error.

The operating MCPR limit is obtained by addition of the absolute, maximum ΔCPR value for the most limiting transient (including any imposed adjustment factors) to the fuel cladding integrity safety limit MCPR of 1.07.

The transients are categorized as rapid pressurization events (turbine trip without bypass, generator load rejection without bypass, feedwater controller failures, pressure regulator downscale failure) and non-pressurization events for the purpose of analysis. The slower, non-pressurization transients are analyzed using either the steady-state three dimensional BWR core simulator or the REDY transient model. Rapid pressurization events are analyzed using the "One Dimensional Core Transient Model" (ODYN).

The ODYN code contains a one dimensional representation of the reactor core which is coupled to a recirculation and control system model. The integrated model is based on one dimensional reactor kinetics, multi-noded thermal hydraulic and heat transfer relationships, and mechanical kinetic equations of the equipment. ODYN contains a refined reactor core description and a detailed steam line model to simulate pressure dynamics during a transient.

There are two basic methods under which ODYN calculated MCPR values can be used. Each utility has the choice of operating under one of the basic options.

Under option A, an NRC-imposed adjustment factor is applied to the MCPR for each event to account for code uncertainties. The code assumes the performance of key components at their adverse tolerances (i.e., CRD scram speed at the technical specification limit, scram setpoints at the technical specification limit, etc.).

Under option B, adjustment factors are also applied. These factors are a result of a statistical analysis of transient response based on an improved CRD scram insertion time distribution. Since these adjustment factors take credit for conservatism in the scram speed assumed for the transient analysis, each plant operating under option B must demonstrate that its scram speed is within the distribution used in the statistical analysis (USNRC RTC, 19B2).

PROBLEM

Under transient condition analyses, there appears to be concern over whether credit can be given for certain non-safety grade equipment.

The concern results from the fact that in most BWRs, the high water level (L8) trip and the turbine bypass system may not be single failure proof and the bypass system is not seismically qualified. Yet, the compounding of failures of more than one system may lead to a highly improbable event. Historically, applicants have been required to assume failure of the turbine bypass system in their analyses of turbine trip and generator load rejection events, but credit has been given for the use of the high water level (L8) trip and the turbine bypass systems for the feedwater controller failure (maximum demand) transient. This was based on the consideration that the feedwater event with failure of the bypass and L8 trip, would result in very limited, if any, fuel failures. This event would, however, become limiting for all plants for which the turbine trip without bypass transient is now limiting, and could result in a derate of some operating units if required as a backfit. (Ross 1981)

RESOLUTION

The following interim resolution has been deemed feasible:

"that credit can only be taken for these systems if the equipment is identified in plant Technical Specifications with regard to availability, setpoints, and surveillance testing (e.g., LaSalle, Zimmer, etc.). We believe that this will provide reasonable assurance of equipment operability until a clearer understanding of its reliability can be determined." (Ross 1981)

METHODOLOGY

The first approach taken in analyzing this issue was to examine three major transients and evaluate changes in CPR due to turbine trip failure at high-water level and/or failure bypass valves to open during a transient. The transients analyzed include 1) turbine trip without bypass--100% power; 2) loss of one feedwater string--100% power; and 3) feedwater control failure, high--50% power (see Attachment 1). In addition, a quantatitive approach using the guidelines (Andrews, et al. 1983) was taken in an effort to estimate an upper bound on public risk reduction assuming an improvement in L8 trip and TBP (see Attachment 2).

CONCLUSIONS

As a result of examining several BWR transients involving L8 trip and TBP, it was concluded that in no case was reactor vessel coolant inventory compromised due to failure of non-safety-grade equipment, the potential for core melt was considered negligible.

In applying an extremely conservative approach to quantifying the potential risk reduction for this issue, it was assumed that the reliability of the reactor protection logic system (RPLS) would be improved by a factor of ten for a hypothetical resolution. The resulting upper-bound estimate of the total public risk reduction is 240 man-rem.

ATTACHMENT 1

TRANSIENT #1: Turbine Trip Without Bypass--Reactor at 100% Power

The normal response of a BWR operating at 100% power during a transient involving a turbine trip without bypass is shown in Figure 1. An explanation of all major points listed in Figure 1 is given in Table 1.

The maximum effect on CPR based upon this transient is calculated in BWR technical specifications. This is shown in Figure 4 (upper right hand curves) for different core loads. It is shown that a response to this transient will not result in a CPR value of less than 1.07. It should also be noted that an additional safety factor K_f, which is based upon the maximum flow rate, is used to compute the operating MCPR. Thus, the operating MCPR is equal to or greater than the value from Figure 4 multiplied by K_f from Figure 5.

The transient analysis shown in Figure 1 provides time histories of critical parameters during a turbine trip without bypass. Limit switches on the turbine stop valves activate the reactor protection system (RPS), causing a scram and trip of the recirculation pumps. This action is in anticipation of a pressure/power spike due to shutting off the steam path through the turbine and bypass valves. If the signals from the limit switches on the stop valves (not seismically qualified) failed to provide a proper signal, within seconds the reactor would trip on high reactor pressure or high neutron flux. In addition, there would be a recirculation runback as feedwater flow dropped with the transient, proceeding as shown in Figure 1. The reactor vessel coolant inventory would not be compromised by any of the above events; thus, no chance of a core melt is foreseen. It is doubtful whether the operating CPR would even reach 1.07; thus, the fuel clad integrity would not be compromised.

In summary, for the turbine trip without bypass there is a scram and a recirculation pump trip activated by non-safety-grade equipment. Failure of these events (scram and pump trip) would not compromise the integrity of the vessel because alternate action by safety-grade equipment would activate a scram (high pressure or high neutron flux). The recirculation pumps would then run back due to the drop in feedwater. Reactor vessel level would be maintained, thereby precluding a meltdown.

TRANSIENT #2: Loss of One Feedwater Heater String--Reactor at 100% Power

The normal response of a BWR operating at 100% power with the loss of one feedwater heater string (now safety grade) is shown in Figure 2. An explanation of all major points is presented in Table 2.

The loss of feedwater heating results in a positive reactivity insertion. This causes an increase in power and, then, an increase in pressure, steam flow and feedwater flow (feedwater flow following steam flow). The additional steam flow is handled by the bypass valves. An additional increment is added to the base MCPR for this transient as shown in Figure 4 (lower left). If the bypass valves were to fail: 1) the rated amount of steam would still pass through the turbine; 2) reactor pressure would rise, and the reactor could scram on either high pressure or a signal from the flow-biased average power range monitor (APRM). As steam drops there would be a recirculation runback.

The MCPR limit would not be violated, although the bypass valves failed to operate during this transient. Reactor vessel level is not compromised during this transient; thus, no chance of core melt is foreseen.

TRANSIENT #3: Feedwater Control Failure (High)--Reactor at 50% Power

The normal response of a BWR operating at 50% power during a feedwater control failure (high) is shown in Figure 3. An explanation of all major points is presented in Table 3.

The normal transient response is that the feedwater controller fails high. This causes a positive reactivity insertion due to the addition of subcooled feedwater. Power begins to rise, steam flow begins to rise and vessel level rises rapidly. At the high-water-level trip, the main turbine trips, the feedwater turbines trip, the reactor scrams from closure of the stop valves on the turbine trip, and the recirculation pumps trip as a result of the turbine trip. After the turbine trip, the residual steam is passed through the bypass valves.

If the high-water level trip failed, the turbine would not trip. Power would increase, steam flow would increase, and the electrohydraulic control (EHC) would progressively open the control valves to regulate the flow of steam. Water could rise and eventually enter the steam liner. The excessive moisture could reach the turbine, resulting in turbine damage. The bypass valves would not be used in this case. No thermal limits would be exceeded, and no chance for a core melt is foreseen.

There is a chance that the flow-bias power trip could be reached, which would scram the system before MCPR is attained. Again, no thermal limits would be exceeded.

An alternative case would result if the turbine tripped and the bypass valves did not open. For this case the recirculation pumps would trip, feedwater pumps would trip, the reactor would scram, and power would drop. Pressure would rise until a safety relief valve (SRV) lifted. Reactor vessel water level could be maintained by feedwater pumps when water level dropped below the high-water-level trip. Again, no thermal limits would be exceeded, and no chance for a meltdown is foreseen.

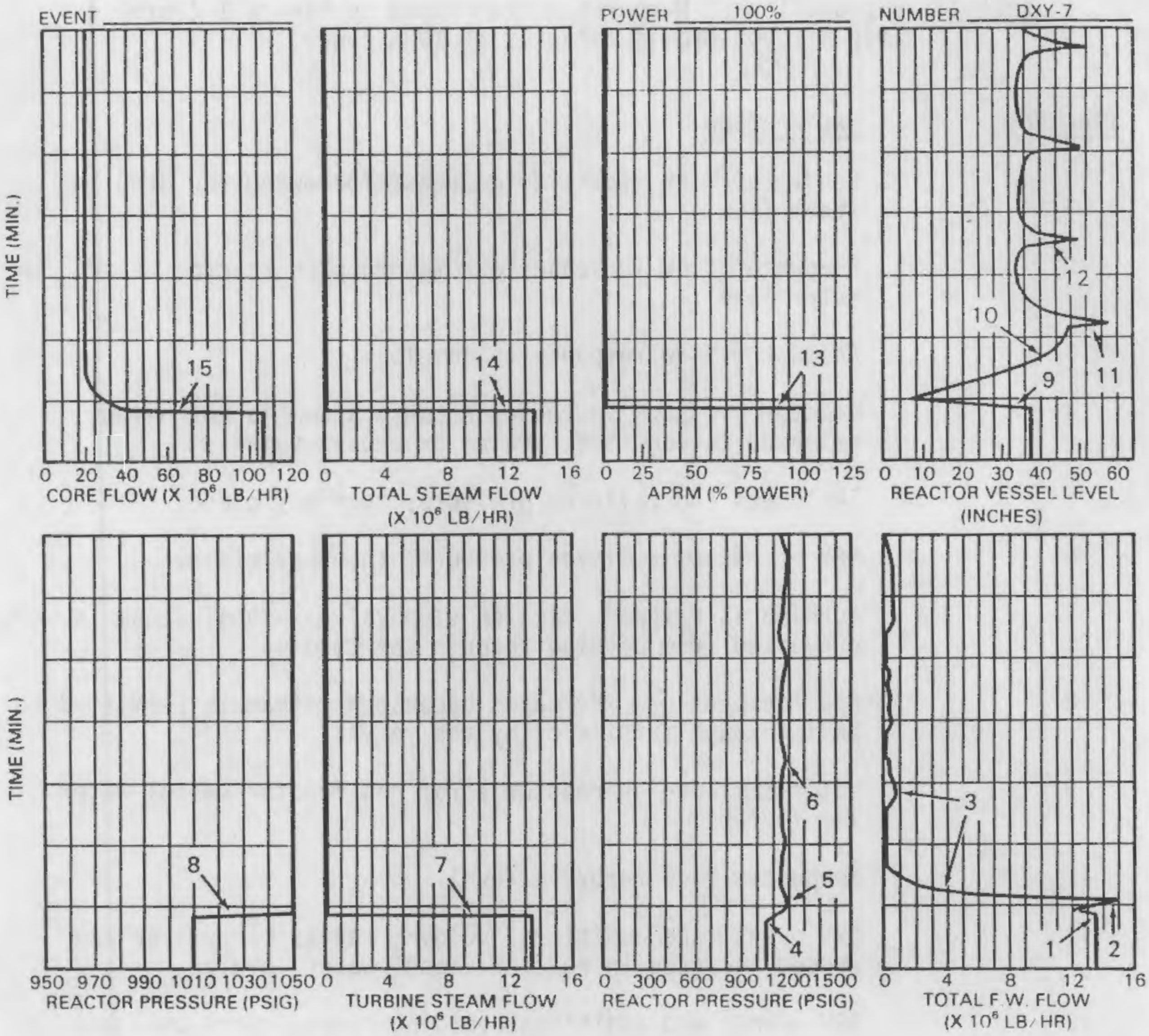

FIGURE 1. Turbine Trip Without Bypass Valves--100% Power (Browns Ferry Simulator 1983. U.S. NRC Training Center, Chattanooga, Tennessee

TABLE 1. Explanation of Numbered Items Listed in Figure 1 (Turbine Trip Without Bypass--Reactor at 100% Power)

Item No.	Explanation
1	Feedwater flow starts to decrease following the drop in steam flow.
2	Feedwater flow increases due to dropping reactor vessel water level.
3	Feedwater flow response to level.
4	Reactor pressure increases because steam is not being released through the turbine or bypass valves.
5	SRV opens and relieves pressure, then may close.
6	SRV opens and relieves pressure, then may close.
7	Turbine is tripped, closing stop valves--thus, steam is prevented from passing through the turbine.
8	Reactor pressure increases because no steam is permitted to pass through turbine or bypass valves.
9	Voids collapse on reactor scram and reactor vessel water level drops.
10	Feedwater flow recovers level.
11	SRV opens, and additional voiding causes carry over and sudden increase in reactor vessel water level.
12	SRV opens, and additional voiding causes carry over and sudden increase in vessel water level.
13	Reactor scrams on turbine stop valve fast closure (scrams on turbine trip; would scram on high pressure if scram on turbine trip failed).
14	Steam flow is stopped through the turbine and bypass valves.
15	Recirculation pump trips on turbine trip; thus, core flow reduces to natural circulation valve.

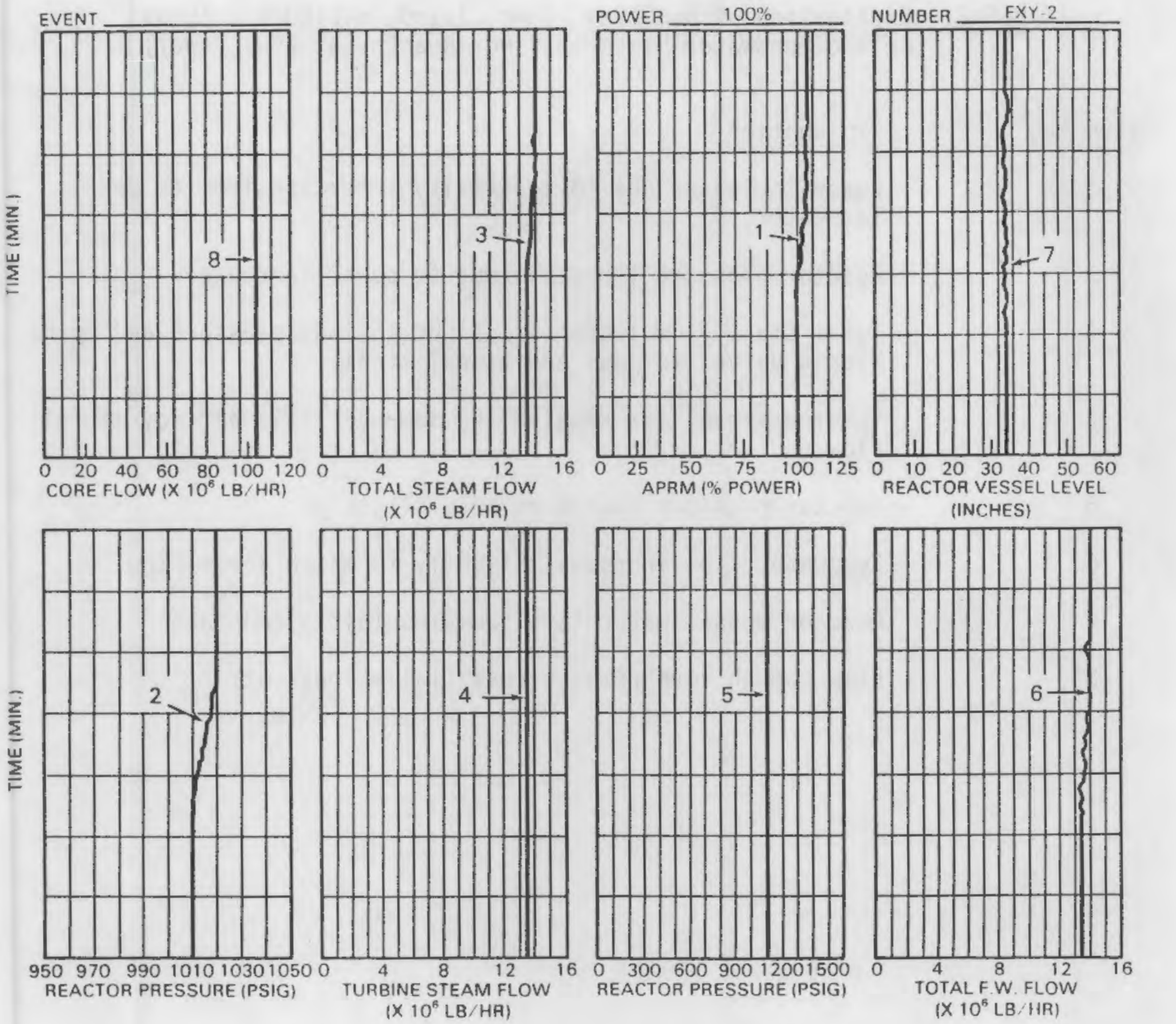

FIGURE 2. Loss of One Feedwater Heater String--100% Power (Browns Ferry Simulator 1983. U.S. NRC Training Center, Chattanooga, Tennessee)

TABLE 2. Explanation of Numbered Items Listed in Figure 2 (Loss of One Feedwater Heater String--Reactor at 100% Power)

Item No.	Explanation
1	Power increases due to reactivity insertion from colder feedwater.
2	Reactor pressure increases due to power increase.
3	Total steam flow increases as EHC follows pressure and opens bypass valves to pass additional steam.
4	Turbine steam flow remains the same as it is already at full flow.
5	Reactor pressure increases only slightly.
6	Feedwater flow increases slightly to match steam flow.
7	Reactor vessel water level remains fairly constant.
8	Flow though core remains essentially constant.

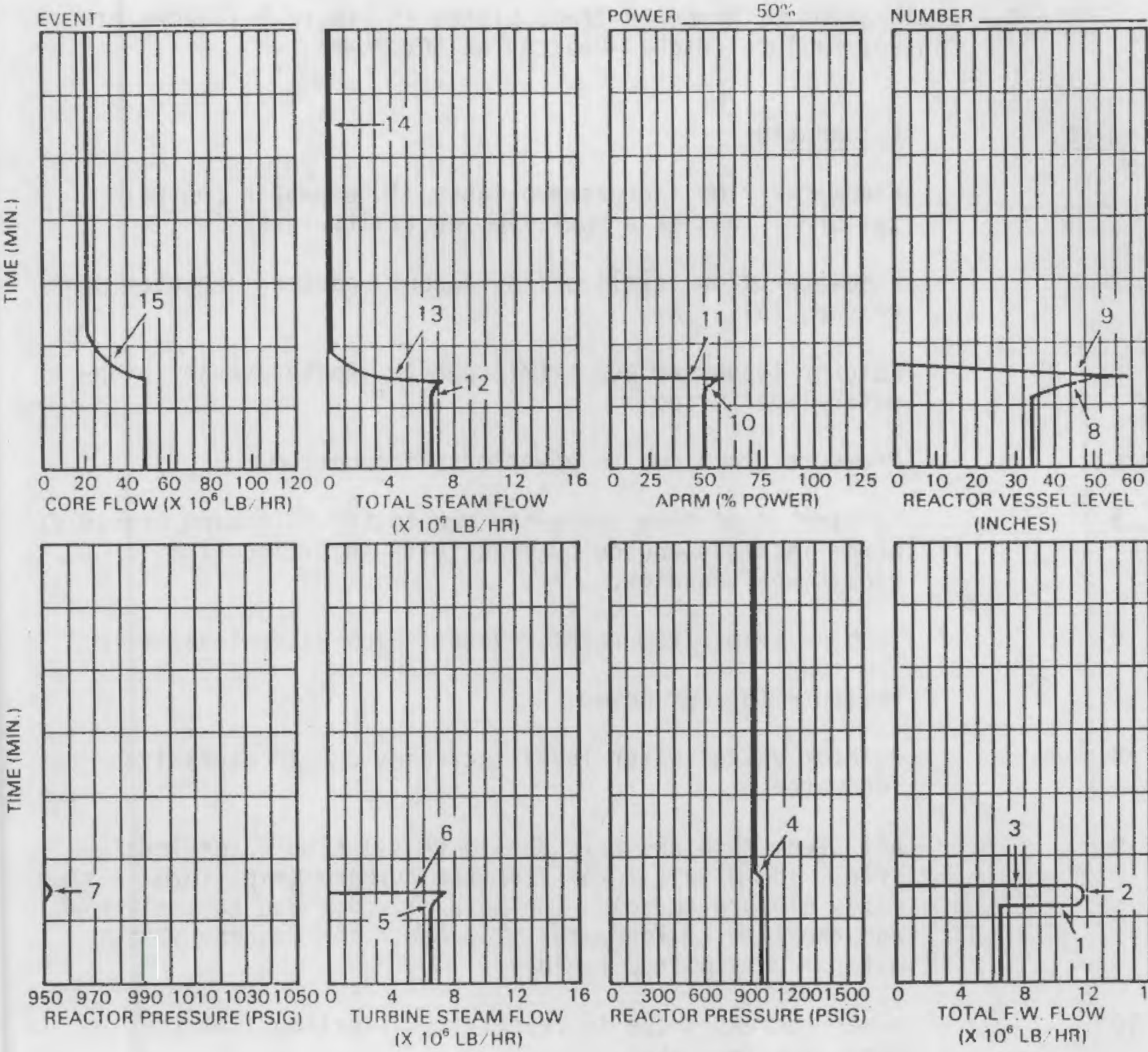

FIGURE 3. Feedwater Control Failure {High}--50% Power (Browns Ferry Simulator 1983. U.S. NRC Training Center, Chattanooga, Tennessee)

TABLE 3. Explanation of Numbered Items Listed in Figure 3 (Feedwater Control Failure {High}--Reactor at 50% Power

Item No.	Explanation
1	Feedwater flow increases because of feedwater control failure - this is a positive reactivity insertion.
2	Feedwater flow levels off as second reactor feedwater pump reaches full flow.
3	Reactor feedwater pumps trip due to reactor vessel high-water-level trip.
4	Pressure drops due to power drop from scram.
5	Turbine steam flow increases due to EHC following pressure/power increase caused by reactivity insertion from excessive feedwater.
6	Turbine trips from reactor vessel high-water-level trip.
7	Pressure follows power.
8	Reactor vessel water level increases due to excessive feedwater.
9	All steam turbines trip due to vessel high-water-level trip. There is a scram from the turbine trip (turbine stop valve closure scram). The loss of voids due to the scram and the loss of feedwater flow cause the reactor vessel water level to drop rapidly.
10	Power increases due to reactivity insertion from the excessive feedwater.
11	Power drops due to scram from turbine trip.
12	Steam flow rises due to power increase.
13	Steam flow drops because EHC is controlling steam flow by reactor pressure. Flow is through bypass valves after stop valves close.
14	EHC is controlling steam flow through bypass valves based on reactor vessel pressure.
15	Recirculation pumps trip on turbine trip in anticipation of pressure/power spike caused by turbine trip.

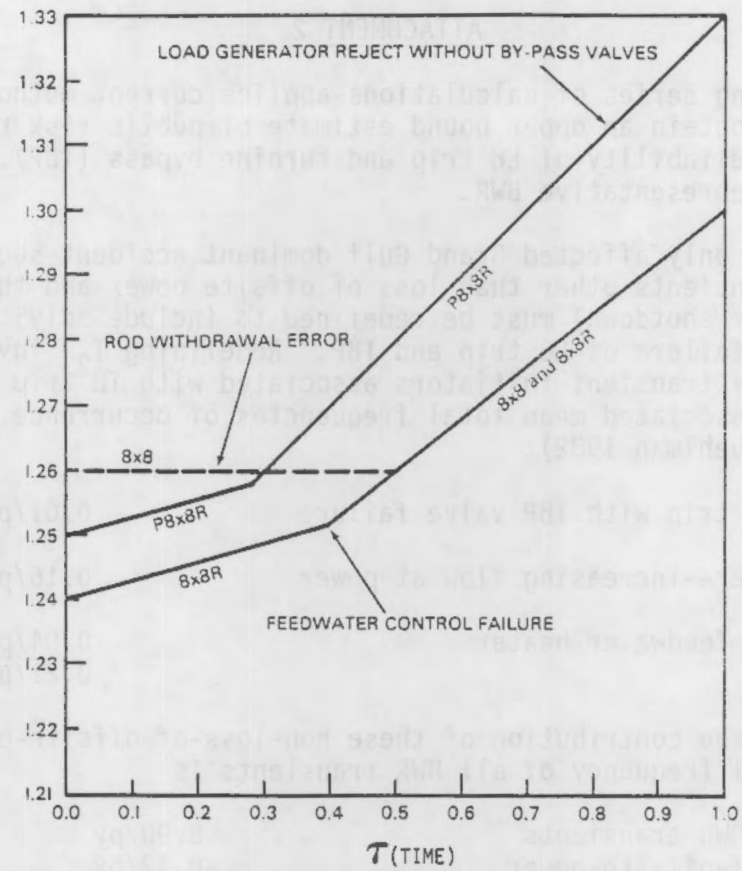

FIGURE 4. MCPR Limits--Safety Margins Introduced Due to Various Transients (TVA 1977)

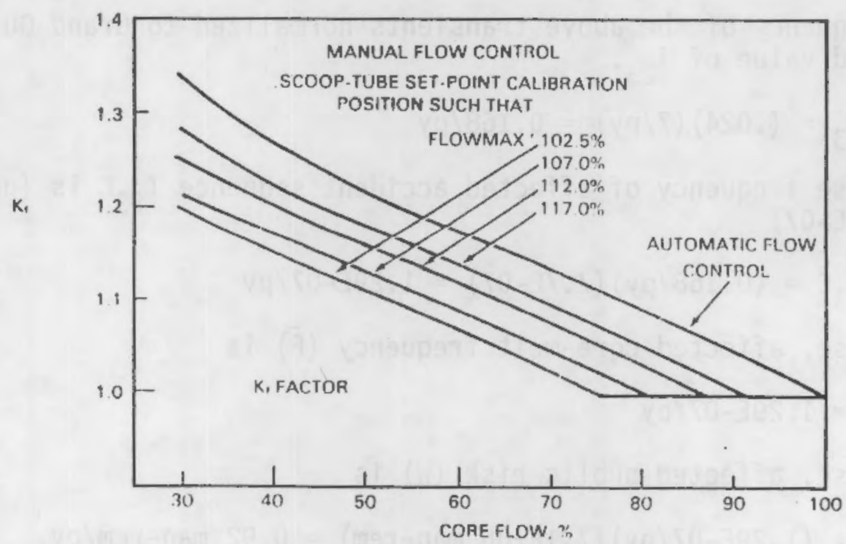

FIGURE 5. K_f Versus Core Flow Percent for Various Max Flowrates (TVA 1977)

ATTACHMENT 2

The following series of calculations applies current methodology (Andrews et al. 1983) to obtain an upper bound estimate of public risk reduction assuming an increase in reliability of L8 trip and turbine bypass (TBP). Grand Gulf is used as the representative BWR.

$T_{23}C$ is the only affected Grand Gulf dominant accident sequence for this issue. T_{23} (transients other than loss of offsite power and those requiring emergency reactor shutdown) must be redefined to include only those transients associated with failure of L8 trip and TBP. Redefining T_{23} involves identification of transient initiators associated with T8 trip and TBP. These initiators and associated mean total frequencies of occurrence are as follows (McClymont and Puehlman 1982):

1.	Turbine trip with TBP valve failure	0.01/py
2.	Feedwater--increasing flow at power	0.16/py
3.	Loss of feedwater heater	0.04/py
		0.21/py

The percentage contribution of these non-loss-of-offsite-power transients to the mean total frequency of all BWR transients is

Total BWR transients	8.90/py	
Loss-of-offsite-power	-0.12/py	
Non-loss-of-offsite-power transients	0.21/py	
	8.78/py	= 0.24 = Percentage Contribution

The frequency of the above transients normalized to Grand Gulf produces the redefined value of T_{23}.

$$T_{23} = (.024)(7/py) = 0.168/py$$

Base-case frequency of affected accident sequence $T_{23}C$ is (note: C = 7.7E-07)

$$T_{23}C = (0.168/py)(7.7E\text{-}07) = 1.29E\text{-}07/py$$

Base-case, affected core-melt frequency ($\bar{F}$) is

$$\bar{F} = 1.29E\text{-}07/py$$

Base-case, affected public risk (W) is

$$W = (1.29E\text{-}07/py)(7.1E\text{+}06\ \text{man-rem}) = 0.92\ \text{man-rem/py}$$

ATTACHMENT 2. (cont'd.)

For the adjusted case, it is assumed that availability of the reactor protection logic system will increase by a factor of 10 at most (produces extremely conservative results).

C^* (adjusted) = (1.9E-07 + 5.8E-06)(0.1) = 6.0E-07/py

$T_{23}C^*$ = (0.168/py)(6.0E-07) = 1.01E-07/py

Adjusted-case, affected core-melt frequency ($\bar{F}^*$) is

$\bar{F}^*$ = 1.01E-07/py

Adjusted-case, affected public risk (W^*) is

W^* = (1.01E-07/py)(7.1E+06 man-rem) = 0.72 man-rem/py

Reduction in core-melt frequency ($\Delta\bar{F}$) is

$\Delta\bar{F}$ = 1.29E-07 - 1.01E-07 = 2.8E-08/py

Per-plant reduction in public risk (ΔW) is

ΔW = 0.92 - 0.72 = 0.20 man-rem/py

Total Public Risk Reduction $(\Delta W)_{Total}$ is

(44 BWRs)(27.4 yr)(0.20 man-rem/py) = 240 man-rem

This should be viewed as an upper-bound value based on an extremely conservative analysis.

REFERENCES

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

McClymont, A. S., and B. W. Poehlman. 1982. ATWS: A Reappraisal, Part 3: Frequency of Anticipated Transients, Interim Report. EPRI-NP-2230, Science Applications, Inc., Palo Alto, California.

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MAKE-UP NOZZLE CRACKING IN BABCOCK AND WILCOX (B&W) PLANTS

ISSUE 69

SAFETY ISSUE DESCRIPTION

Cracks were found in the normal make-up high pressure injection (MU/HPI) nozzles of several Babcock and Wilcox (B&W) plants following an inspection of the eight B&W plants licensed to operate. These cracks appeared to be directly related to loose or missing thermal sleeves. As a result, a B&W Owner's Group Task Force was established to identify the cause of the failures and recommend modifications to eliminate future failures (Dircks, W. J. 1982).

The B&W Task Force has completed a generic investigation of the MU/HPI nozzle component cracking problem and has submitted a report containing the findings of that investigation (B&W 1983). The report presents relevant facts and probable failure scenarios, as well as recommended modifications to thermal sleeve designs, makeup system operating conditions and inservice inspection (ISI) plans. The following information presents background information and accepted issue resolutions to date.

"Site inspections conducted in February-April 1982 indicated that both the HPI only nozzles and the double-duty HPI/MU nozzles were affected. Loose, out-of-place, and cracked thermal sleeves were observed in six of the HPI only nozzles, while four of the double-duty nozzles also contained cracked safe-ends. Failure analysis indicated that the cracks were initiated on the inside diameter and were propagated by thermal fatigue. The cracked safe-end at Crystal River also contained mechanically initiated outside diameter cracking which appeared to be unrelated. Previous inspections at two plants (Davis Besse-1 and Three Mile Island-2) under construction revealed that one of the Davis Besse sleeves was loose. All four sleeves were subsequently re-rolled at Oavis Besse (hard rolled, instead of contact expanded as originally specified). Recent inspections at Midland have also shown that gaps may be present between the thermal sleeve and safe-end in the contact expanded joint. These findings along with stress analysis and testing have implicated insufficient contact expansion of the thermal sleeves as the most probable root cause of the failures (B&W 1983)."

SAFETY ISSUE SIGNIFICANCE

The incorporation of a thermal sleeve into a nozzle assembly is a common practice in the nuclear industry to provide a thermal barrier between the cold HPI/MU fluid and the hot pressure injection nozzle. This helps prevent thermal shock and fatigue of the nozzle. The purpose of the safe-end is to make the field weld easier (pipe to safe-end) by allowing similar metals to be welded. The dissimilar metal weld between the safe-end and the nozzle can then be made under controlled conditions in the vendor's shop. The use of the safe-end also eliminates the need to do any post-weld heat treating in the field. Failures in these HPI/MU nozzles may preclude the proper functioning of the ECCS and/or the normal fluid makeup to the primary system.

SAFETY ISSUE RESOLUTION

Following the investigation, B&W has made several recommendations for modifying the design, operation and inspection procedures of the HPI/MU nozzles. "(1) A hard rolled thermal sleeve design has been developed which helps prevent thermal shock to the nozzle assembly and helps reduce flow induced vibrations more effectively. (2) An increase in minimum continuous makeup flow has been suggested to help prevent thermal stratification in the MU line and more effectively cool the safe-end. (3) An inservice inspection plan has also been developed to provide a means of early problem detection (B&W 1983)." (4) A detailed stress analysis of the nozzle with a modified thermal sleeve design justifying long term operation has been suggested.

All licensees participating in the B&W Owners' Group Task Force performed the repairs to damaged components outlined in Recommendation (1). The augmented ISI program in Recommendation (3) was voluntarily implemented. The stress analysis of Recommendation (4) will be done by the affected licensees and will require an MPA for follow-up staff verification (Denton, H. R. 1984). Thus, this issue was RESOLVED and no new requirements were established.

REFERENCES

Babcock and Wilcox. 1983. 177 Fuel Assembly Owner's Group Safe End Task Force Report on generic Investigation of HPI/MU Nozzle Component Cracking. B&W Document Number: 77-1140611-00, the Babcock and Wilcox Company, Utility Power Generation Division, Lynchburg, Virginia.

Denton, H. R. 1984. "Resolution of Generic Issue 69: Make-up Nozzle Cracking in B&W Plants." September 27, 1984 Memo to T. P. Speis, U.S. Nuclear Regulatory Commission, Washington, D.C.

Dircks, W. J. 1982. "Make-up Nozzle Cracking in Babcock and Wilcox (B&W) Plants." July 23, 1982 Memo to Commissioners (SECY-82-186A), U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 70, PORV and Block Valve Reliability

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Power-operated relief valves (PORVs) installed in many operating PWRs have no reliability or operability specifications and are not of safety grade quality. Until recently, these valves were not specified for use in design basis accidents. Approximately 55% of Westinghouse and Babcock and Wilcox (B&W) plants are currently operating with PORVs blocked off to prevent excessive leakage. This may reduce plant flexibility in transient situations. This issue would establish PORVs and block valves as safety grade components for the purpose of improving the reliability of PORV/block valve closure.

AFFECTED PLANTS	PWR:	Operating = 43	Planned = 42
	BWR:	Operating = 0	Planned = 0

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	1.5E+04
OCCUPATIONAL DOSES :	
SIR Implementation =	902
SIR Operation/Maintenance =	0
Total of Above =	902
Accident Avoidance =	12

COST RESULTS ($1E+06)

INDUSTRY COSTS:	
SIR Implementation =	15.4
SIR Operation/Maintenance =	2.9
Total of Above =	18.3
Accident Avoidance =	11.0
NRC COSTS:	
SIR Development =	1.5
SIR Implementation Support =	0
SIR Operation/Maintenance Review =	0
Total of Above=	1.5

PORV AND BLOCK VALVE RELIABILITY

ISSUE 70

1.0 SAFETY ISSUE DESCRIPTION

This safety issue addresses the need to improve the reliability of the power-operated relief valve (PORV) and block valve located on the pressurizer in Westinghouse, Babcock & Wilcox (B&W), and some Combustion Engineering (CE) reactors. This improvement would help to assure 1) the opening and closing function of this valve system as dictated by various plant transients and 2) the mitigation of consequences of potential plant accidents.

The use of the PORV/block valve system can be quite flexible in a transient situation and can lessen the frequency of demands on the larger, spring-loaded safety relief valves (SRVs). In most plants, the low temperature overpressure (LTOP) protection system is also designed to use the PORVs. However, the PORVs and block valves are currently not safety components, and thereby are not specifically required to control design basis accidents. They have no set technical specifications for reliability.

PORV valves have stuck open on numerous occasions, essentially resulting in a small break loss-of-coolant accident (SBLOCA). In other cases where the PORVs have had leakage problems, the leakage has simply been stopped by closure of the block valve. It is currently estimated that approximately 55% of the Westinghouse and B&W plants are operating with the block valves closed so as not to exceed the technical specification limits for loss of coolant from the primary system.

PROPOSED ISSUE RESOLUTION

Issue resolution is assumed to be specification of the PORV/block valve combination as a safety grade system for the purpose of bringing the reliability up to a specified level. This will minimize the potential introduced for SBLOCAs, while retaining the system's flexibility to respond to a range of transients before relying on the SRVs. Less frequent operation with the block valve closed should also lessen the potential for LTOP events during startup. The increased reliability could be achieved through better initial qualifications for the valve, specified maintenance and testing requirements, and the addition of an automatic actuating circuit for the motor-operated block valve. The last essentially eliminates the potential for failure of the operator to close the block valve given leakage past the PORV. For this analysis, these improvements are assumed to be implemented.

Seven operating Combustion Engineering (CE) plants currently have PORVs in their design. It will be assumed that future CE plants would include them. The need to include PORVs in the CE PWR (Arkansas Nuclear One-Unit 2) currently lacking them is considered to be addressed as part of unresolved Safety Issue A-45, "Decay Heat Removal," and is thus not included here. The analysis for this CE plant compares the change in risks of the CE design with and without the PORV's block valves. For the Westinghouse and B&W plants, however, it has not been proposed that the PORVs be eliminated. Therefore, that scenario is not analyzed here. This analysis compares the risks presented by the existing population of Westinghouse, CE, and B&W reactors in their current configuration to the risk presented after more reliable PORVs are introduced. Details of the analysis are discussed further in Attachment 1 to the Public Risk Reduction Work Sheet.

AFFECTED PLANTS

This issue impacts operating and planned Westinghouse and B&W PWRs, and current and futire CE plants with PORVs.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose associated with the proposed safety issue resolution (SIR) are developed in this section. Results are summarized in Tables 1 and 2, respectively.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

PORV and Block Valve Reliability (70)

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

		N	$\bar{T}$ (yr)
Westinghouse PWRs:	Operating	31	27.5
	Planned	30	30
B&W PWRs:	Operating	8	27.8
	Planned	5	30
Combustion Engineering PWRs:	Operating	8	27.1
	Planned	8	30
All Affected PWRs:		90	28.7

3. Plants Selected for Analysis:

Arkansas Nuclear One 1(ANO-1, Kolb et al. 1982)

4. Parameters Affected by SIR:

Based upon the discussion in Attachment 1, the frequency of small break loss of coolant accident (SBLOCA), S2, and the parameter Q, failure of SRVs to close, will be used to model failure of the PORV to close after demand. For those plants with PORVs active, the improvement in the frequency of SBLOCA due to inadvertent PORV leakage will be examined. For those plants with block valves closed, the improvement due to reduced frequency of demand of the SRVs will be examined.

Plants with PORVs

For those plants with PORVs, the frequency of SBLOCA due to PORV leakage will be the product of PORV lift frequency, P_f, times the probability of failure to close, Q, or

$$S_2(PORV) = P_f Q$$

where

P_f = frequency of PORV lift
Q = Q(PORV) Q(block valve)
Q(PORV) = probability PORV fails to close having opened
Q(block valve) = probability of failure of operator to manually actuate the block valve, plus the probability of block valve closure failure.

Note again that some of the affected PWRs (55% of the total affected is assumed), the PORV is blocked off. For these plants, $S_2(PORV)=0$. For these plants, the frequency of SRV lift and failure to close, Q(SRV), will be of interest.

5. Base-Case Values for Affected Parameters:

For those plants with PORVs,

P_f = 1/py
Q(PORV) = 0.02
p(failure of operator manual actuation of block valve) = 0.05
p(block valve closure failure) = 0.005
Q(block valve) = 0.05 + 0.005 = 0.055
Q = (0.02)(0.055) = 0.0011
$S_2(PORV) = P_f x Q$ = (1/py)(0.0011) = 0.0011/py.

For those plants without PORVs,

Q(SRV) = 0.02. This compares to 0.05 used in the Oconee-1 PRA.

6-7. Steps Related to Affected Accident Sequences, Release Categories, and Their Base-Case Values:

From the discussion in Attachment 1, the following dominant transient sequences are assumed to be affected at ANO-1 for plants with PORVs blocked:

TABLE 1. (cont'd.)

Sequence	Base Case Frequency (per py)
T(DO1)LQ(SRV)-D3	4.0E-06
T(A3)LQ(SRV)-D3	3.3E-06
T(LOP)LQ(SRV)-D3	9.0E-07
T(PCS)LQ(SRV)-D3	8.8E-07

Sequence	Base Case Frequency (per py)
T(PCS)Q(SRV)-H1	7.2E-07
T(FIA)MQ(SRV)-H1	5.7E-07
T(FIA)MLQ(SRV)-D3	3.9E-07
T(DO2)LQ(SRV)-D3	2.2E-07
TOTAL =	1.10E-05

These frequencies are assumed to represent the base-case values at the 55% of the affected plants whose PORVs are blocked off (ANO-1 is included in this group).

At the remaining 45% of the affected plants, the base-case, affected frequencies will be assumed to be the frequency of SBLOCA due to PORV leakage, S_2(PORV), times the probability of core-melt given the SBLOCA. The ANO-1 PRA uses the B(1.66) break size to model a PORV lift. With a frequency of B(1.66)=3.1E-04/py and total core-melt for B(1.66) sequences of 1.2E-06/py, the conditional probability of core-melt given B(1.66) or PORV lift and sticking is then put at (1.2E-06/py)/(3.1E-04/py) = 3.9E-03/demand.

This compares to the probability of core-melt given SBLOCA in the WASH-1400 study with a failure probability of 9E-03/demand. The Oconee-1 PRA probability of core-melt given an S_2 break is 8.25E-03/demand.

The base-case core-melt frequency for the plants with PORVs then becomes

$$\begin{aligned} F &= S_2(\text{PORV})(3.9\text{E-03/demand}) \\ &= (0.0011/\text{py})(3.9\text{E-03/demand}) = 4.3\text{E-06/py}. \end{aligned}$$

On the average, the base-case, affected core-melt frequency becomes

$$\begin{aligned} F &= 0.55(1.10\text{E-05/py}) + 0.45(4.3\text{E-06/py}) \\ &= 8.0\text{E-06/py} \end{aligned}$$

Since all of the above sequences lead to release via categories PWR-1, 2, 5, and 7 with containment failure mode likelihoods of 1E-04, 0.5, 0.007, and 0.5, respectively, the base-case, affected release category frequencies for plants with PORVs blocked become

TABLE 1. (cont'd.)

PWR-1 = 8.0E-10/py
PWR-2 = 4.0E-06/py
PWR-5 = 5.6E-08/py
PWR-7 = 4.0E-06/py

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

$\bar{F}$ = 8.0E-06/py (composite for all affected plants)

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

W = 1.9E+01 man-rem/py (composite for all affected plants).

10. Adjusted-Case Values for Affected Parameters:

For plants with PORVs, improvement in PORV and block valve closure reliability and reduced operator error through the use of automatic actuation of the block valve are thought to give the following values:

P_f = 1/py
Q(PORV) = 0.01
p(failure of manual or automatic actuation of block valve) = 0.002
p(block valve closure failure) = 0.003
Q(block valve) = 0.002 + 0.003 = 0.005
Q = (0.01)(0.005) = 5.0E-05/demand
S_2(PORV) = P_fxQ = (1/py)(5.5E-05/demand) = 5.5E-05/py.

For those plants with block valves now closed, reintroducing the use of a more reliable PORV/block valve combination is thought to reduce the frequency of SRV lifts. Because improved the PORV/block valve combination is more likley to close that the SRV (5.0E-05/demand compared to 0.02/demand), this will reduce the net effective value of Q. The transients modeled above tend to be quite severe (loss of power conversion, etc), so it will be assumed that use of the PORVs will be effective in preventing SRV lift in 10% of the transients. The effective value of Q will then become

Q' = (0.9Q + 0.1Q(PORV/block valve)
= (0.9)(0.02) + (0.1)(5.0E-05) = 0.018.

11-12. Steps Related to Adjusted-Case Frequencies of Affected Accident Sequences and Release Categories:

Release category frequencies are estimated directly to be the following:

TABLE 1. (cont'd.)

PWR-1 = 5.5E-10/py
PWR-2 = 2.8E-06/py
PWR-5 = 3.9E-08/py
PWR-7 = 2.8E-06/py

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

For plants with PORVs, the new core-melt frequency is then

$\bar{F}$* = (5.5E-05/py)(3.9E-03/demand) = 2.1E-07/py.

For plants with block valves currently closed, the new core-melt frequency is estimated at

$\bar{F}^*$ = (1.1E-05/py)(0.018/0.02) = 9.9E-06/py.

The average adjusted-case affected core-melt frequency is then estimated at

$\bar{F}^*$ = (0.45)(2.1E-07/py) + (0.55)(9.9E-06/py) = 5.5E-06/py.

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

W* = 1.3E+01 man-rem/py

15. Reduction in Core-Melt Frequency (ΔF):

ΔF = 8.0E-06/py - 5.5E-06/py = 2.5E-06/py

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

ΔW = 1.9E+01 man-rem/py - 1.3E+01 man-rem/py = 6 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.5E+04	1.5E+06	0

ATTACHMENT 1 (To Table 1)

The introduction of a PORV/block valve system into a PWR will trade off a number of risks and benefits. Those recognized in this analysis are as follows:

- The PORV/block valve system has a probability for failure to seat after lifting, thus increasing the probability of an SBLOCA to some degree if subject to lifts.

- Replacing an existing PORV/block valve system with a more reliable system will reduce this probability of an SBLOCA to some degree.

- The PORV/block valve system allows more flexibility in small transients compared to an SRV, and thus may reduce the number of challenges to the SRVs. This may not be true of the CE plants which were designed to function without the PORV, but it is likely to be the case for Westinghouse and B&W plants. The transients modeled in the plant PRAs resulting in SRV lift tend to involve loss of feedwater or loss of the power conversion function in general, so it is uncertain if PORV action alone would prevent SRV lift under such circumstances. It will be assumed here that the use of a PORV would prevent SRV lifting in 10% of such transients, thus providing a higher likelihood of valve closure under such conditions.

The PORVs can also impact the progression of a number of other accident scenerios which are thought to be of secondary safety significance here. These include:

- Use of the PORV for low temperature overpressure (LTOP) protection. The several events that have occured in the past can be attributed to poor PORV/block valve reliability, with the result that the block valve has been closed. Although the LTOP issue is being addressed seperately in multi-plant action item (MPA) B-04 and thus will not be addressed here, the need remains to coordinate the use of PORV/block valves under various operational conditions.

- Use of PORVs during anticipated transients without SCRAM (ATWS). Successful valve response to ATWS events (~1E-05/py) is typically defined as having both PORVs and SRVs open, with partial success being PORVs or SRVs. Improved reliability of the PORV may lower the probability of failure to open to some degree, but this has typically been put at 0.01. As a result, any improvements would give reductions in core-melt frequency below 1E-07/py and thus contribute insignificantly here.

- Use of the PORV to depressurize the primary system in the event of a steam generator tube rupture (SGTR) can possibly reduce the offsite exposure resulting from the release of iodine in the coolant. The SGTR case is not considered to be significant enough to include in the calculations for risk reduction in this issue analysis. This conclusion is based on a June 1983 report by the NRC Task Force on SGTR that estimated a release of 53,600 curies of I-131 associated with an SGTR. Based on informtion given in NUREG/CR-0651 (Marsh 1980) for an SGTR at Prairie Island 1, a release of 2.1E-04 Ci of I-131 was estimated to result in a public exposure of 4.3E-06 rem to the thyroid. A conversion factor of 100 thyroid-rem/man-rem was used in Safety Issue III.A.1.3, "Maintain Supply of Thyroid Blocking Agent," indicating a public exposure of

(53600 Ci I-131/SGTR)(4.3E-06 thyroid-rem/2.1E-04 Ci I-131)
(1 man-rem/100 thyroid-rem) = 11 man-rem/SGTR.

With the assumed SGTR frequency of 1.3E-03/py estimated by the NRC Task Force on SGTR, a base-case, affected public risk of (1.3E-03/py)(11 man-rem) = 0.014 man-rem/py due to SGTR is calculated. This base-case, affected public risk is small compared to that associated with the SBLOCA pathways (calculated below) and can thus be dropped from further consideration.

The main risk associated with this issue is thus assumed to be the reduction SBLOCA frequency due to improved PORV leakage following lift. The assumptions for this are presented directly in the work sheets, and the reader is referred there for details. The remaining significant contributor to risk is then the potential reduction in SRV lifts in transients where PORV action would be sufficient, coupled with the reduced probability of PORV failure to close compared to SRVs. The transient sequences identified in the Arkansas Nuclear One 1(ANO-1) IREP study (Kolb et al. 1982) are examined. The potentially affected parameters are P and Q, defined as follows:

P = probability of failure of the SRVs/PORV to open
Q = probability of failure of the SRVs/PORV to close.

The ANO-1 study identifies the following transient sequences containing P or Q:

T(FIA)MQ-	T(LOSW)Q-	T(LOP,PCS,A3,B5,DO1,DO2)	Q-
T(FIA)MP	T(LOSW)P	"	P
T(FIA)MPC	T(LOSW)LQ-	"	PC
T(FIA)MPY	T(LOSW)LP	"	PY
T(FIA)MPYC		"	PYC
T(FIA)MLQ-		"	LQ-
T(FIA)MLP		"	LP
T(FIA)MLPC		"	LPC
T(FIA)MLPY		"	LPY
T(FIA)MLPYC		"	LPYC
T(FIA)KQ-		"	KQ-
T(FIA)KP		"	KP
T(FIA)KPC		"	KPC
T(FIA)KPY		"	KPY
T(FIA)KPYC		"	KPYC
T(FIA)KMQ-		"	KPL
T(FIA)KMP		"	KLPC
T(FIA)KMPC		"	KLPY
T(FIA)KMPY		"	KLPYC

ATTACHMENT 1. (cont'd.)

T(FIA)KMPYC
T(FIA)KMLP
T(FIA)KMLPC
T(FIA)KMLPY
T(FIA)KMLPYC

where

T(FIA) = transient occurs with all frontline systems initially available
T(LOP) = transient initiated by loss of offsite power
T(PCS) = transient initiated by total interruption of main feedwater
T(A3) = transient initiated by failure of the engineered safeguards (ES) bus A3
T(B5) = transient initiated by failure of ES bus B5
T(D01) = transient initiated by failure of ES bus D01
T(D02) = transient initiated by failure of ES bus D02
T(LOSW) = transient initiated by failure of the plant service water system
C = failure of the reactor building spray injection system
K = failure of the reactor protection system
L = failure of the emergency feedwater system
M = failure of the power conversion system
P = failure of the relief valves to open
Q = failure of the relief valves to reseat
Y = failure of the reactor building cooling system.

In the ANO-1 analysis, none of the sequences containing P or K terms were found to be significant. Only the sequences containing the Q term, which indicates SBLOCAs of equivalent diameter <1.66 in., were considered further (and only if the K term was absent). The potential for recovery was then examined for these sequences on an SBLOCA event tree. Recovery from the SBLOCA focused on the failure of 2 out of 3 high-pressure injection pumps (D3) and failure of the high-pressure recirculation system (H1). Core-melt frequencies were given only for the following sequences (Table C-9 in Kolb et al. 1982), the rest being considered insignificant:

Sequence	Frequency, per year
T(D01)LQ-D3	4.0E-06
T(A3)LQ-D3	3.3E-06
T(LOP)LQ-D3	9.0E-07
T(PCS)LQ-D3	8.8E-07
T(PCS)Q-H1	7.2E-07
T(FIA)MQ-H1	5.7E-07
T(FIA)MLQ-D3	3.9E-07
T(D02)LQ-D3	2.2E-07

The ANO-1 study actually considered only the first two sequences as dominant and dropped the other sequences in its reported results. They are included here for completeness. The assumptions as to changes in the frequency of these sequences as a result of SIR are given directly in Table 1, the Public Risk Reduction Work Sheet.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

PORV and Block Valve Reliability (70)

2. Affected Plants (N):

		N	$\bar{T}$ (yr)
Westinghouse PWRs:	Operating	31	27.5
	Planned	30	30
B&W PWRs:	Operating	8	27.8
	Planned	5	30
Combustion Engineering PWRs:	Operating	8	27.1
	Planned	8	30
All Affected PWRs:		90	28.7
Operating PWRs:		47	27.5

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

For all 90 affected PWRs, $\bar{T}$ = 28.7 yr

4. Per-Plant Occupational Dose Reduction Due to Accident Avoidance, (ΔFD_R:

ΔFD_R = (2.5E-06/py)(1900 man-rem) = 0.0048 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.2E+01	2.4E+02	0

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

It is assumed that it will take 96 man-h/plant to replace the existing PORV/block valve system with a safety grade system.

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

Radiation fields near the pressurizer during shutdown are put at 0.2 R/hr (EPRI-NP-1139, p.3-26). The estimated dose is then

D = (96 man-h/plant)(0.2 rem/h) = 19.2 man-rem/plant.

TABLE 2. (cont'd.)

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

   ND = (47 plants)(19.2 man-rem/plant) = 9.0E+02 man-rem

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

   It is assumed that additional annual inspection and testing during refueling outages to improve valve reliability will take 4 man-h/plant in radiation fields of 0.2 R/hr, giving an exposure increase of 0.8 man-rem/py.

   A review of information in EPRI-NP-1138 indicates that annual maintentance on the PORVs can be as high as 50 man-hrs/py. Although testing and inspection will add time to this, the implementation of better procedures for maintenance to improve valve reliablility is thought to reduce the time required for annual maintenance. Using a 10% improvement figure or 5 hours/py in a 0.2 R/hr radiation field indicates that doses could be reduced by 1 man-rem/py. This essentially is thought to offset any dose increase due to increased inspection and testing.

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

   $D_o = 0.$

11. Total Occupational Dose Increase for SIR Operation and Maintenance:

   $N\bar{T}D_o$ = 0 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
9.0E+02	2.7E+03	3.0E+02

3.0 SAFETY ISSUE COSTS

The industry and the NRC costs associated with the proposed SIR are developed in this section. Results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   PORV and Block Valve Reliability (70)

TABLE 3. (cont'd.)

2. Affected Plants (N):

		N	$\bar{T}$ (yr)
Westinghouse PWRs:	Operating	31	27.5
	Planned	30	30
B&W PWRs:	Operating	8	27.8
	Planned	5	30
Combustion Engineering PWRs:	Operating	8	27.1
	Planned	8	30
All Affected PWRs:		90	28.7

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

For all 90 affected PWRs, $\bar{T}$ = 28.7 yr

43 planned PWRs, $\bar{T}$ = 30 yr

47 operating PWRs, $\bar{T}$ = 27.5 yr

Industry Costs (Steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, Δ(FA):

Δ(FA) = (2.5E-06/py)($1.65E+09) = $4.13E+03/py

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

Best Estimate	Upper Bound	Lower Bound
$1.1E+07	2.0E+08	0

6. Per-Plant Industry Resources for SIR Implementation:

It is assumed that the 96 man-h/plant in a radiation field required for installation represents only 20% of the necessary utility labor for valve backfit, giving a total of 480 man-h/plant (12 man-wk/plant at $2270/man-wk) at operating PWRs. This will include management review, QA control, licensing review, and engineering for the backfit. No such labor is foreseen at planned plants. Material requirements are two safety grade PORVs and two instrumented (for automatic actuation) block valves, each costing $25,000 at operating plants. Incremental material costs above those associated with initial installation of the safety grade PORV and instrumented block valve are estimated at $50,000.

TABLE 3. (cont'd.)

The cost for placing a PORV valve in an existing CE plant has been estimated at \$4.25E+06, \$2.5E+06 in a new plant. However, the Westinghouse and B&W plants are already designed and built with the PORVs in place. As such, the large costs associated with engineering and safety review of the PORV could be avoided. However, safety analysis and review of the block valve (automatically actuated) is required. The cost is estimated at \$50,000/plant at operating PWRs. For new plants, an incremental effort above the analysis required for relief valves is put at \$5,000/plant. Finally, a Class III License Amendment at \$4000 is expected at operating plants as a result of the valve upgrades.

For forward fit plants, the new valves are assumed to add only \$25,000/valve in additional cost compared to the purchase of non-safety grade valves. An additional \$5000 for safety analysis is also included above that which would be normally expected. No additional material, labor, or licensing fees are estimated.

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

At Operating Plants:

Labor = (6 man-wk/plant)(\$2270/man-wk)	= \$ 27,200/plant
Valves = (4 valves/plant)(\$25,000/valve)	= \$100,000/plant
Material (hardware, piping, etc)	= \$ 50,000/plant
Analysis and Review	= \$ 50,000/plant
License Amendment Fee	= \$ 4,000/plant
	\$231,200/plant

At Planned Plants:

Valves (4 valves/plant)	= \$100,000/plant
Analysis and Review	= \$ 5,000/plant
	\$105,000/plant

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

NI = (47 operating plants)(\$231,200/plant) +
(43 planned plants)(\$105,000 plant) = \$1.54E+07

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

The 4 man-h/py of labor in a radiation field required for annual testing is again assumed to represent 20% of the necessary labor, giving a total of 20 man-h/py (0.50 man-wk/py).

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

I_o = (0.5 man-wk/py)(\$2270/man-wk) = \$1140/py

TABLE 3. (cont'd.)

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

$$\bar{NTI_O} = (90\ \text{plants})(28.7\ \text{yr})(\$1140/\text{py}) = \$2.9E{+}06$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$1.8E+07	2.6E+07	1.0E+07

NRC Costs (Steps 13 through 21)

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

It is assumed that the NRC will fund the generic and plant specific studies necessary to establish reliability and performance goals for the PORV and block valve. These will be published as a Regulatory Guide. This is assumed to require 1 man-yr of contractor labor. The total cost is estimated at 3 man year or \$300,000. This cost will be spread over the 90 operating and planned plants, for a per plant cost of \$3,333. The plant specific reviews at each plant are further estimated to require 6 man-wk at \$2270/man-wk. The total is then \$1.70E+04/plant, or \$1.5E+06 for 85 plants.

15-20. Steps Related to NRC Costs for Support of SIR Implementation and Review of SIR Operation and Maintenance:

All of these costs are expected to be carried by the utilities or to be included in routine onsite NRC inspections. Thus, no additional NRC costs are identified.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$1.5E+06	3.1E+06	0

REFERENCES (For Issue 70)

EPRI-NP-1139, "Limiting Factor Analysis of High Availability Nuclear Plants," Electric Power Research Institute, August 1979.

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Minarick, J., and C. Kukielka. 1982. Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report. NUREG/CR-2497, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

WASH-1400. 1975. Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants. (NUREG-75/014), U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 71, Failure of Resin Demineralizer Systems

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Literature reviews suggest a need for additional licensing attention with regard to resin bed demineralizer systems. It is considered possible that failure of associated process systems could result in "initiating" events which are not bounded by current licensing requirements. Due to possible impact on plant safety, a review by technical branches is requested. Resolution should determine whether FSAR Chapter 15 assumes adequate failure assumptions, and whether corrections should be made to systems of concern.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	4.9E+03
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	1.6E+01

COST RESULTS ($1E+06)

INDUSTRY COSTS:	
SIR Implementation =	0.77
SIR Operation/Maintenance =	-1.9E+02
Total of Above =	-1.9E+02
Accident Avoidance =	1.3
NRC COSTS:	
SIR Development =	0.20
SIR Implementation Support =	0.10
SIR Operation/Maintenance Review =	0
Total of Above =	0.30

FAILURE OF RESIN DEMINERALIZER SYSTEMS

ISSUE 71

1.0 SAFETY ISSUE DESCRIPTION

"A recent search of the literature and the LERs suggests that additional licensing attention is needed for certain ancillary power plant equipment. Available information shows that failures of resin bedtype demineralizer subsystems have occurred within the process systems--both nuclear and non nuclear--of nuclear power facilities. These process systems, by definition, do not directly perform any Reactor Protection (RP) or Engineered Safeguards Features (ESF) functions, yet their failure has seriously impaired the capability of those systems to perform, by rendering their redundant trains inoperable. In addition, it is possible that the failure of such process systems could result in 'initiating events' which are not bounded by the current licensing basis for the facility, and may thereby result in the facility being inadequately protected by available RP and ESF Systems, even though they remain fully operable" (Speis 1982).

Possible causes for the observed situations include a single passive failure of structures supporting the demineralizer resin and single operator error or malfunction of an automatic reprocessing system. The basic concern is that these failures are not presently postulated in Chapter 15 (transient and accident analyses) review of the Final Safety Analysis Report (FSAR).

PROPOSED ISSUE RESOLUTION

A re-evaluation has been suggested when considering operating history in conjunction with potential consequences of postulating failures. The specific request for review should accomplish the following:

1. Determine that the present failure assumptions for FSAR Chapter 15 evaluations are adequate and that operating history does not warrant considering different failures, such as demineralizer resin beds.

2. Propose new failure assumptions to be made for Chapter 15 evaluations that are consistent with operating data.

3. Propose corrections to those systems whose failure could result in unacceptable consequences (Speis 1982).

AFFECTED PLANTS

This issue affects all PWRs and BWRs.

2.0 SAFETY ISSUE RISK AND DOSE

The public risk reduction and occupational dose are estimated in this section.

PUBLIC RISK REDUCTION

It is assumed that resolution of this issue will serve to identify a need for procedural changes with the possibility of minor system hardware changes in demineralizer systems. A reduction in public risk will result from a decreased probability of a common-cause failure mode representing loss of Low Pressure/Containment Spray Recirculation System (LP/CSRS) in PWRs. A decrease in the number of transients requiring shutdown resulting from malfunctions in demineralizer systems in PWRs and BWRs will also affect public risk. 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:

   Failure of Resin Demineralizer Systems (71)

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

		N	$\bar{T}$ (yr)
PWRs:	Operating	47	27.7
	Planned	43	30.0
	All PWRs	90	28.8
BWRs:	Operating	24	25.2
	Planned	20	30.0
	All BWRs	44	27.4

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

   Grand Gulf 1 - representative BWR

4. Parameters Affected by SIR:

   During reprocessing of a demineralizer subsystem on a PWR facility during residual heat removal (RHR) operation at cold shutdown, a system malfunction occurred. The malfunction during the demineralizer

TABLE 1. (cont'd.)

backflushing process rendered all RHR pumps inoperable. It is, therefore, assumed that demineralizer malfunction can result in common-cause failure of the LP/CSRS. The only system failure of concern in recirculation mode are the containment spray recirculation and emergency coolant recirculation systems.

In BWRs, a large number of situations have been reported in which resins have intruded into the reactor from either the full-flow condensate polisher system or the reactor water cleanup (RWCU) system (Electric Power Research Institute 1980). According to Speis, "Types of failures causing these intrusions have generally not been described; particular instances include a suspected tear in the resin trap on the related RWCU system, and a valving error during resin transfer. No related impacts on other plant systems are reported in the reference" (Speis 1982).

Based on the redefinition of Oconee 3 parameters related to a potential loss of LP/CSRS, a common-cause parameter Z is defined as an affected parameter for this SIR (PWRs only). Z represents loss of LP/CSRS due to demineralization system malfunction or operator error. See Attachment 1 for details related to parameter Z. In addition, T_3' and T_{23}' are affected parameters for Oconee 3 and Grand Gulf 1, respectively.

5. Base-Case Values for Affected Parameters:

Oconee 3:

$Z = 7.86E\text{-}06/py$[a]

$T_3' = 0.34/py$

Note: T_3' includes transients requiring shutdown due to malfunctions in demineralizer systems. The number of events (number of scrams due to abnormal water chemistry events/years of operation) was obtained for BWRs from EPRI NP-1603 (Electric Power Research Institute 1980) and was assumed to be identical for PWRs.

Grand Gulf 1:

$T_{23}' = 0.34/py$

(T_{23}' was assumed to be identical to T_3' above).

(a) See Attachment 1.

TABLE 1. (cont'd.)

6. Affected Accident Sequences and Base-Case Frequencies:

	Sequence	Frequency (1/py)
Oconee 3:		
T_2MQFH	γ(PWR-2)	5.90E-09
	β(PWR-4)	8.61E-11
	ϵ(PWR-6)	5.90E-09
S_3FH	γ(PWR-2)	5.10E-09
	β(PWR-4)	7.45E-11
	ϵ(PWR-6)	5.10E-09
S_2FH	α(PWR-1)	3.14E-11
	γ(PWR-2)	6.28E-10
	β(PWR-4)	2.29E-11
	ϵ(PWR-6)	2.51E-09
T_3MLUO	γ(PWR-3)	4.68E-08
	β(PWR-5)	6.83E-10
	ϵ(PWR-7)	4.68E-08
Grand Gulf 1:		
$T_{23}PQI$	α(BWR-1)	1.80E-09
	δ(BWR-2)	1.80E-07
$T_{23}PQE$	γ(BWR-3)	1.31E-08
	δ(BWR-4)	1.31E-08
$T_{23}QW$	δ(BWR-2)	5.83E-07
$T_{23}C$	δ(BWR-2)	2.62E-07

7. Affected Release Categories and Base-Case Frequencies:

Category	Frequency (1/py)
PWR-1	3.14E-11
PWR-2	1.16E-08
PWR-3	4.68E-08
PWR-4	1.84E-10
PWR-5	6.83E-10
PWR-6	1.35E-08
PWR-7	4.68E-08
BWR-1	1.80E-09
BWR-2	1.03E-06
BWR-3	1.31E-08
BWR-4	1.31E-08

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

$\bar{F}_{PWR}$ = 1.19E-07/py

$\bar{F}_{BWR}$ = 1.05E-06/py

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

W_{PWR} = 3.12E-01 man-rem/py

W_{BWR} = 7.39 man-rem/py

10. Adjusted-Case, Values for Affected Parameters

An LER search on demineralizer malfunctions revealed 161 LERs of interest between 1974 and 1983. The SIR is assumed to reduce malfunctions related to inadequate procedures or technical specifications and improperly trained personal by as much as 50 percent. Examples of some possible malfunction corrections, as a result of SIR, are listed below.

- corrected flow paths to prevent intrusions--many times from operator error (e.g., mixing of main condensate water with DWST (demineralizer water storage tank))
- review and revision of chemistry procedures (e.g., displacement of chloride from the purification ion exchange resin--this occurs when hydrazine decomposes to form ammonia which, in the presence of boric acid, forms ammonia borate)
- reduce incidents of condensate polisher bypass during startup
- reduce chance of contamination of DW system when attaching hose between DW and spent-fuel pool
- improved methods for anticipating resin bed depletions

10. Adjusted-Case, Values for Affected Parameters

- improved methods for monitoring system pressure during sluicing operation--reduces chance of exceeding relief valve set point and releasing gas to stack
- improved procedures to double check the status of all valves to prevent such occurrences as unacceptably low tank levels or water backing into other systems (e.g., lower-pressure air blower)
- spring fastening devices for screw type filter elements to prevent resin passage attributed to vibration and pressure surges.

TABLE 1. (cont'd.)

The affected values for adjusted-case parameters are as follows:

Oconee 3:

$Z = 3.93E\text{-}06/py$

$T_3' = 0.17/py$

Grand Gulf 1:

$T_{23}' = 0.17/py$

11. Affected Accident Sequences and Adjusted-Case Frequencies:

		Sequence	Frequency (1/py)
Oconee 3:			
	T_2MQFH	γ(PWR-2)	2.95E-09
		β(PWR-4)	4.30E-11
		ϵ(PWR-6)	2.95E-09
	S_3FH	γ(PWR-2)	2.56E-09
		β(PWR-4)	3.73E-11
		ϵ(PWR-6)	2.56E-09
	S_2FH	α(PWR-1)	1.57E-11
		γ(PWR-2)	3.14E-10
		β(PWR-4)	1.15E-11
		ϵ(PWR-6)	1.26E-09
	T_3MLUO	γ(PWR-3)	2.34E-08
		β(PWR-5)	3.42E-10
		ϵ(PWR-7)	2.34E-08

TABLE 1. (cont'd.)

Grand Gulf 1:

$T_{23}PQI$	(BWR-1) (BWR-2)	8.99E-10 8.99E-08
$T_{23}PQE$	(BWR-3) (BWR-4)	6.55E-09 6.55E-09
$T_{23}QW$	(BWR-2)	2.91E-07
$T_{23}C$	(BWR-2)	1.31E-07

12. Affected Release Categories and Adjusted-Case Frequencies:

Category	Frequency (1/py)
PWR-1	1.57E-11
PWR-2	5.82E-09
PWR-3	2.34E-08
PWR-4	9.18E-11
PWR-5	3.42E-10
PWR-6	6.77E-09
PWR-7	2.34E-08
BWR-1	8.99E-10
BWR-2	5.12E-07
BWR-3	6.55E-09
BWR-4	6.55E-09

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

$\bar{F}*_{PWR}$ = 5.95E-08/py

$\bar{F}_{BWR}$ = 5.26E-07/py

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

$W*_{PWR}$ = 1.56E-01 man-rem/py

$W*_{BWR}$ = 3.69 man-rem/py

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

$\Delta\bar{F}_{PWR}$ = 5.95E-08/py

$\Delta\bar{F}_{BWR}$ = 5.24E-07/py

TABLE 1. (cont'd.)

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

ΔW_{PWR} = 0.156 man-rem/py

ΔW_{BWR} = 3.70 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
4.9E+03	2.9E+05	0

ATTACHMENT 1 (To Table 1)

All Oconee 3 parameters related to potential loss of LP/CSRS are redefined to include a common-cause failure Z, which represents loss of LP/CSRS due to demineralization system malfunction. The resulting affected accident sequences and cut sets (which include Z) are as follows:

$\underline{T_2MQFH}$

$T_2 \cdot M \cdot \bar{P} \cdot Q \cdot Z$

$\underline{S_3FH}$

$S_3 \cdot Z$

$\underline{S_2FH}$

$S_2 \cdot Z$

T_2MQH and S_3H were also considered, but were eliminated because they contained identical cut sets to those found in accident sequences T_2MQFH and S_3FH plus the latter represent core melt sequences without core spray available and this situation would also be true for common-cause failure Z cutsets.

To estimate the failure probability of Z, it is necessary to consider available historical data. These data include one observed failure of LP/CSRS (both trains). Given this failure and 349 expended PWR plant years, assuming a 35-year life and subtracting future operating years (Andrews et al. 1983, Table C.2), the failure probability based on historical data is:

$$\lambda = 1 \text{ failure}/349 \text{ PWR-yr} = 2.87\text{E-}03 \text{ failures/year}$$

Converting this failure rate into unavailability (failure probability) for parameter Z, an estimated downtime of 24 hours/transient/year is used. Therefore, the following unavailability estimate for LP/CSRS exists:

$$Z = \lambda t = [2.87\text{E-}03 \text{ failures/yr}][(24 \text{ hr/transient})/(8760 \text{ hr/yr}] = 7.86\text{E-}06$$

This is taken as the base-case value for the parameter Z. Substituting this value for Z and using the original values for the remainder of the cut-set elements results in the following base-case frequencies for the affected accident sequences:

ATTACHMENT 1. (cont'd.)

Affected Sequences	Base-Case Frequency (1/py)
T_2MQFH:	
$T_2 \cdot M \cdot P_1 \cdot Q \cdot Z$	5.90E-09
S_3FH:	
$S_3 \cdot Z$	5.11E-09

Affected Sequences	Base-Case Frequency (1/py)
S_2FH:	
$S_2 \cdot Z$	1.57E-09

Issue resolution is assumed to reduce the unavailability by 50 percent. This assumption is based on the engineering judgment that if appropriate training and a conscientious maintenance program are instituted, a 50 percent reduction in unavailability can be achieved.

Affected Sequences	Base-Case Frequency (1/py)
T_2MQFH:	
$T_2 \cdot M \cdot \bar{P}_1 \cdot Q \cdot Z$	5.90E-09
S_3FH:	
$S_3 \cdot Z$	5.11E-09
S_2FH:	
$S_2 \cdot Z$	1.57E-09

ATTACHMENT 1. (cont'd.)

OCCUPATIONAL DOSE

Since this safety issue resolution (SIR) involves procedural changes for the most part, it is assumed that personnel exposure to radiation zones due to implementation and operation/maintenance would be negligible. This assumption is further supported by the fact that demineralizer systems are located in high-maintenance/low-exposure areas of the plant. The results of the occupational dose estimates are summarized in Table 2. Table 2 includes occupational dose reduction due to accident avoidance.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Failure of Resin Demineralizer Systems (71)

2. Affected Plants (N):

   All PWRs and BWRs are affected.

		N
PWRs:	Operating	47
	Planned	43
	All PWRs	90
BWRs:	Operating	24
	Planned	20
	All BWRs	44

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

		$\bar{T}$ (yr)
PWRs:	Operating	27.7
	Planned	30.0
	All PWRs	28.8
BWRs:	Operating	25.2
	Planned	30.0
	All BWRs	27.4

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

   ΔPWR: (19,900 man-rem)(5.95E-08/py) = 1.18E-03 man-rem/py

   ΔBWR: (19,900 man-rem)(5.24E-07/py) = 1.04E-02 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
16.0	190	0

TABLE 2. (cont'd.)

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

Resolution of this issue involves procedural and technical-specification changes with minor hardware changes (e.g., bypass hoses, gauges for monitoring water levels, spring-type fasteners for screw-in-type filter elements). Any minor hardware fixes would be completed during regularly scheduled resin bed replacement. Therefore, no occupational dose increase due to SIR implementation is anticipated.

9-11. Steps Related to Occupational Dose Increase for SIR Operation and Maintenance:

It is anticipated that issue resolution accomplished by decreasing demineralizer system failures would result in fewer plant shutdowns. However, because demineralizer systems are located in high-maintenance areas, the reduction in occupational dose is expected to be negligible.

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)
negligible

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost analyses are included in this section. Best estimates are used for labor time and replacement power costs needed for issue resolution. Table 3 includes the results of this analysis.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Failure of Resin Demineralizer Systems (71)

2. Affected Plants (N):

		N
PWRs:	Operating	47
	Planned	43
	All PWRs	90

TABLE 3. (cont'd.)

BWRs:	Operating	24
	Planned	20
	All BWRs	44

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

		$\bar{T}$ (yr)
PWRs:	Operating	27.7
	Planned	30.0
	All PWRs	28.8
BWRs:	Operating	25.2
	Planned	30.0
	All BWRs	27.4

Industry Costs (Steps 4 through 12)

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

$\Delta\bar{F}A$ PWR: (\$1.65E+09)(5.95E-08/py) = \$9.82E+01/py

$\Delta\bar{F}A$ BWR: (\$1.65E+09)(5.24E-07/py) = \$8.65E+02/py

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

Best Estimate	Upper Bound	Lower Bound
\$1.3E+06	\$1.6E+07	\$0

6. Per-Plant Industry Resources for SIR Implementation:

Training maintenance personnel for procedural changes due to issue resolution should require minimal staff labor time. It is estimated that one man-week will be required to accommodate all procedural changes in a program specifically designed for maintenance personnel, appropriate shift supervisors, engineers, and technical advisors.

In addition, minor hardware changes anticipated will require some flow-path alterations, filter revisions, and monitoring equipment. An estimate for installation of such equipment is two man-weeks. Such installation is assumed to be completed during scheduled shutdowns.

TABLE 3. (cont'd.)

BWRs:	Operating	24
	Planned	20
	All BWRs	44

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

		$\bar{T}$ (yr)
PWRs:	Operating	27.7
	Planned	30.0
	All PWRs	28.8
BWRs:	Operating	25.2
	Planned	30.0
	All BWRs	27.4

Industry Costs (Steps 4 through 12)

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

$\bar{FA}$ PWR: (\$1.65E+09)(5.95E-08/py) = \$9.82E+01/py

$\bar{FA}$ BWR: (\$1.65E+09)(5.24E-07/py) = \$8.65E+02/py

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

Best Estimate	Upper Bound	Lower Bound
\$1.3E+06	\$1.6E+07	\$0

6. Per-Plant Industry Resources for SIR Implementation:

Training maintenance personnel for procedural changes due to issue resolution should require minimal staff labor time. It is estimated that one man-week will be required to accommodate all procedural changes in a program specifically designed for maintenance personnel, appropriate shift supervisors, engineers, and technical advisors.

In addition, minor hardware changes anticipated will require some flow-path alterations, filter revisions, and monitoring equipment. An estimate for installation of such equipment is two man-weeks. Such installation is assumed to be completed during scheduled shutdowns.

In the event that changes are made in FSAR Chapter 15 or in technical specifications, an amendment fee assessment may be required. For purposes

TABLE 3. (cont'd.)

In the event that changes are made in FSAR Chapter 15 or in technical specifications, an amendment fee assessment may be required. For purposes of this analysis, it is assumed a license amendment is applicable and the associated fee is estimated at $4000 per reactor.

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

   I = (3 man-wk/plant)($2270/man-wk)+($4000/plant) = $1.08E+04/plant

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

   The implementation costs are related to changes in operating plants and would be considered in original design, construction and training for planned plants. Therefore, implementation costs for the SIR are applicable to operating plants only.

   NI = ($1.08E+04/plant)(71 plants) = $7.7E+05

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

   A reduction in plant down-time is expected due to resolution of this issue. If 0.34 scrams/year are reduced by 50 percent (see Step 10 of Public Risk Reduction Work Sheet and Attachment 1) due to resolution of this issue, a cost savings is realized.

   Maintenance labor savings is not considered significant for this issue resolution.

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

   Base Case

   (0.34 scram/py)(1 day/scram)($300,000/day) = $1.02E+05

   Adjusted Case

   (0.50)(0.34 scrams/py)(1 day/scram)($300,000/day) = $5.10E+04/py

   I_O = $5.10E+04 - $1.02E+05 = $5.10E+04/py

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

   $N\bar{T}I_O$ = -$5.10E+04/py [(90)(28.8) + (44)(27.4)] = -$1.9E+08

TABLE 3. (cont'd.)

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
-\$1.9E+08	\$9.6E+07	-\$2.9E+08

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Resolution of this issue involves a review by responsible technical branches to determine whether demineralization system failures have been properly acknowledged under Chapter 15 evaluations. This issue assumes a review of two man-years to specifically accomplish the following:

1. Determine that the present failure assumptions for Chapter 15 evaluations are adequate and that operating history does not warrant considering different failures, such as demineralizer resin beds.

2. Propose new failure assumptions to be made for Chapter 15 evaluations consistent with operating data.

3. Propose corrections to those systems whose failure could result in unacceptable consequences (e.g., lose all ECCS).

This review will include all technical assistance required to review historical data.

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

C_D = (2 man-yr)(\$100,000/man-yr) = \$2.00E+05

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

Resolution of this issue assumes some procedural changes applicable to all plants. Proposals and reviews of such changes (e.g., Chapter 15 revisions are estimated to be on the order of one man-year over all plants. This would include any necessary case-by-case reviews.

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

C = (1 man-yr/134 plants)(\$100,000/man-yr) = \$7.46E+02/plant

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

NC = \$1.00E+05

TABLE 3. (cont'd.)

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

No reviews of operation and maintenance are anticipated beyond those presently in existence.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$3.0E+05	\$4.12E+05	\$1.88E+05

REFERENCES (For Issue 71)

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.

Electrical Power Research Institute (EPRI). 1980. Water Quality in Boiling Water Reactors. EPRI-NP1603. Prepared by Radiological and Chemical Technology, Inc., San Jose, California.

Speis, T. P. 1982. "Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety." Memorandum to W. V. Johnston, U.S. Nuclear Regulatory Commission, Washington, D. C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 74, Reactor Coolant Activity Levels for Operating Reactors

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Radioactivity entrained in the coolant of light water reactors is subject to accidental release and can subsequently expose the public. Limiting conditions of operation (LCOs) are typically set on allowable activity levels in the coolant, with Iodine-131 being the dominant factor. However, at some plants, LCOs on the dose equivalent I-131 activity are either nonexistent or inadequate. Standard Technical Specifications (STSs) have been proposed but not yet implemented at all plants. This issue would finally implement the STSs for I-131 for all the remaining plants in question. This issue also contributes to iodine-spiking concerns covered by Issue 65.

AFFECTED PLANTS	BWR:	Operating = 20	Planned = 0
	PWR:	Operating = 11	Planned = 0

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	0
OCCUPATIONAL DOSES =	
SIR Implementation =	16
SIR Operation/Maintenance =	2700
Total of Above =	2700
Accident Avoidance =	0

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	4.5
SIR Operation/Maintenance =	6.1
Total of Above =	11
Accident Avoidance =	0
NRC COSTS:	
SIR Development =	0.025
SIR Implementation Support =	0
SIR Operation/Maintenance Review =	0
Total of Above =	0.025

REACTOR COOLANT ACTIVITY LEVELS FOR OPERATING REACTORS

ISSUE 74

1.0 SAFETY ISSUE DESCRIPTION

In the early days of light water reactor operation, limiting conditions of operation (LCOs) for coolant activity were determined on a plant-by-plant basis, postulating accidents such that subsequent releases and exposures were an appropriately small fraction of the 10 CFR 100 guidelines. Gross activity limits were typically specified, with an assumed isotope spectrum. The limiting accident was represented by the radiological consequences of a postulated steam generator tube rupture in a PWR or a steam line break in a BWR. Similarly, the allowable secondary activity in a PWR was limited by a postulated secondary coolant steam line break. Many plants continue to sample only for gross activity, not identifying specific iodine isotope concentrations.

In May of 1974, standard technical specifications (STSs) were proposed for limiting the dose equivalent Iodine-131 coolant activity concentrations (Denton 1974). The purpose was to establish uniform concentration limits for all plants. In addition to standard monitoring, sampling, and reporting requirements, the STSs would promote uniform characteristics in shielding, personnel protection, and coolant cleanup system capacity.

The basis for the new LCOs was the requirement that doses resulting from steam generator tube rupture and steam line break accidents be below the 10 CFR 100 guidelines even for the worst sites meteorologically. The proposed limiting equilibrium concentrations were 1.0E-06 and 0.1E-06 Ci/gram of I-131 in the primary and secondary coolant, respectively, for PWRs, and 0.2E-06 Ci/gram of coolant for BWRs.

Transient-induced spiking of the iodine concentrations in the coolant can occur. The LCOs recognize this, and allow for elevated concentrations for a period not to exceed 1/20th, or 5 percent, of the plant's annual operating time. This is developed fully in Issue 65, "Iodine Spiking."

Implementation of the above LCOs may require additional sampling and isotopic analysis capability at some plants. Operating characteristics may be modified as plants approach the LCO. Current PWR primary I-131 concentrations are in the 0.01E-06 to 0.1E-06 Ci/gram level, giving a margin of at least one order of magnitude below the LCO. There are some exceptions, such as the Ginna plant, which is approximately a factor of 2 below the LCO. Current BWRs operate with I-131 concentrations of approximately 0.01E-06 Ci/gram of coolant, which is a factor of 20 below the LCO. In all likelihood, the new LCO will have no impact on observed iodine concentrations. Prudent management of coolant activities, even in the absence of an LCO, has resulted in these low levels being observed. This careful management is based primarily on the desire to control occupational radiation exposures. However, situations can be postulated currently where a plant could operate with iodine concentrations above the proposed LCO. Again, this is expected to be a spiking problem covered in Issue 65.

AFFECTED PLANTS

The proposed LCOs have been implemented at a number of sites over the years, but only when other changes to the plant technical specifications were being made. As a result, a number of plants currently operate under no LCOs, or LCOs that are considered inadequate. The most recent tally puts this number at 10 PWRs with no LCOs, and one PWR and 20 BWRs with inadequate LCOs (see Attachment 1).

PROPOSED RESOLUTION

The proposed safety issue resolution (SIR) would implement the above LCOs, requiring the affected plants to make modifications to their technical specifications. Sampling frequency would also be standardized, and sampling specifically for I-131 would be required.

2.0 SAFETY ISSUE RISK AND DOSE

The new LCOs require sampling for dose equivalent iodine once every 14 days in PWRs and 31 days in BWRs, and gross activity sampling at least every 72 hours. The potential reduction in public dose associated with this issue is difficult to quantify. As mentioned in Section 1.0, all PWRs and BWRs currently operate at levels below the standard LCO. Most are one order of magnitude below this, and some as much as two orders of magnitude below the LCO. The potential does exist for a plant to operate in excess of the standard LCO, but this is expected to be an iodine-spiking problem covered in Issue 65. Because the current operating levels of iodine observed are so far below the proposed LCO, no modification in plant operating behavior or water chemistry is expected, other than the new sampling and analysis required. No reduction in plant risk is therefore expected.

A risk reduction could be postulated by assuming a distribution for the coolant activity with time, and further assuming that increased sampling would lower this concentration. Any breach of the cooling system would then result in a lower release. However, the levels currently observed are so low that modifications to operating procedures would occur only when concentrations approached the limits, which would be expected to represent a relatively small fraction of the total operation time. Issue 65 assumes that spikes in concentration coincide with plant power transients. However, in this issue, transients are independent of the rise in concentration. The small probability of the two simultaneous events, coupled with a relatively small change in the concentrations that could be expected with a new procedure indicates that any risk reduction would be minimal.

Based on the above assumptions, the public risk reduction for Issue 74 is assumed to be zero, and no Public Risk Reduction Work Sheet is prepared. Occupational dose analysis is performed, and results are summarized in Table 1.

TABLE 1. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Reactor Coolant Levels for Operating Reactors (74)

2. Affected Plants (N):

PWR: Operating 11 Planned 0
BWR: Operating 20 Planned 0

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

PWR: 10 yr
BWR: 10 yr

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

Since there is no reduction in core-melt frequency, U = 0.

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

It is estimated that equipment would be added for the additional sampling requirements in one-half (16) of the plants. One man-week for installation in a 25 mR/hr field is estimated.

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

D = (1 man-wk/plant)(40 man-hr/man-wk)(0.025 R/hr) = 1 man-rem/plant

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

ND = (1 man-rem/plant)(16 plants) = 16 man-rem

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

Additional sampling of the primary coolant may be necessary to bring all plants into compliance with the new STS. Sampling for gross activity is required 3 times per 7 days, with the maximum period between samples not to exceed 72 hours. The sampling frequency for dose equivalent iodine is once every 14 days in PWRs, 31 days in BWRs. This is assumed to amount to the following number of samplings for each reactor type:

TABLE 1. (cont'd.)

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

PWR = [(3 gross activity samplings/7 days) + (1 dose equivalent iodine sampling/14 days)] (365 days/yr)
= 183 samplings/yr

BWR = [(3 gross activity samplings/7 days) + (1 dose equivalent iodine sampling/31 days)] (365 days/yr)
= 168 samplings/yr

Assuming that the sampling and analysis take 2 hours in a 25 mR/hr field, this translates to 366 man-hours/py for PWRs, and 336 man-hours/py for BWRs.

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

$$(D_o)_{PWR} = (366 \text{ man-hr/py})(0.025 \text{ R/hr}) = 9.15 \text{ man-rem/py}$$

$$(D_o)_{BWR} = (366 \text{ man-hr/py})(0.025 \text{ R/hr}) = 8.40 \text{ man-rem/py}$$

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

$$\begin{aligned} N\bar{T}D_o &= [(11 \text{ PWRs})(9.15 \text{ man-rem/py}) + (20 \text{ BWRs}) \\ &= (8.40 \text{ man-rem/py})] (10 \text{ yr}) \\ &= 2690 \text{ man-rem} \end{aligned}$$

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2700	8100	900

3.0 SAFETY ISSUE COSTS

Industry and NRC costs associated with resolving Issue 74 are estimated in this section. Results are summarized in Table 2.

TABLE 2. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Reactor Coolant Activity Levels for Operating Reactors (74)

2. Affected Plants (N):

   PWRs: Operating 11 Planned 0
   BWRs: Operating 20 Planned 0

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

   PWRs: 10 yr
   BWRs: 10 yr

Industry Costs (Steps 4 through 12)

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

   Since there is no reduction in core-melt frequency, H = 0.

6. Per-Plant Industry Resources for SIR Implementation:

   For each affected plant, one man-month is assumed for the preparation of the new STS, plus $4000 for submittal of the STS to the NRC. One-half (16) of these plants are assumed to require additional sampling and analysis equipment, estimated at $250,000, plus one man-month for installation of the new equipment (one man-week of which was assumed to be in a radiation zone). Thus, 16 plants require two man-months each of staff labor plus $254,000 each to implement the SIR. The remaining 15 plants require only man-month each of staff labor plus $4000 each to implement the SIR.

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

   I(15 plants preparing and submitting new STSs)
   = (1 man-mo/plant)(1 man-yr/12 man-mo)($1.0E+05/man-yr)
   + $4000/plant = $1.23E+04/plant

   I(16 plants preparing and submitting new STSs and adding equipment) = (2 man-mo/plant)(1 man-yr/12 man-mo) ($1.0E+05/man-yr) + $2.54E+05/plant = $2.71E+05/plant

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

   NI = 15($1.23E+04/plant) + 16($2.71E+05/plant) = $4.52E+06

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

   This was estimated in Step 9 of Table 1 to be 366 man-hr/py for PWRs, and 336 man-hr/py for BWRs.

TABLE 2. (cont'd.)

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

$$(I_o)_{PWR} = (366 \text{ man-hr/py})(1 \text{ man-wk}/40 \text{ man-hr})(\$2270/\text{man-hr}) = \$2.08E{+}04/\text{py}$$

$$(I_o)_{BWR} = (366 \text{ man-hr/py})(1 \text{ man-wk}/40 \text{ man-hr})(\$2270/\text{man-hr}) = \$1.91E{+}04/\text{py}$$

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

$$N\bar{T}I_o = [(11 \text{ PWRs})(\$2.08E{+}04/\text{py})+(20 \text{ BWRs})(\$1.19E{+}04/\text{py})](10 \text{ yr}) = \$6.10E{+}06$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$1.1E+07	\$1.4E+07	\$6.8E+06

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

The new LCOs have been developed. One quarter of a man-year is estimated for finalizing and publishing the STSs.

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

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

15-20. Steps Related to NRC Cost for Support of SIR Implementation and Review of SIR Operation and Maintenance:

The new LCOs have been implemented at plants in the past with no difficulty. No need for site-specific support is seen. Similarly, no additional labor is foreseen beyond the normal inspection and review requirements. Thus, $C = C_o = 0$.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$2.5E+04	\$3.8E+04	\$1.3E+04

ATTACHMENT 1 (To Issue 74)

In the March 4, 1982, NRC memorandum from R. Wayne Houston to Thomas Novak,[a] several PWRs are listed as having no limits on iodine activity concentrationJ Haddam Neck; Indian Point 2; Kewaunee; Oconee 1, 2, and 3; Point Beach 2; Rancho Seco; Three Mile Island-1; and Zion 1 and 2. In addition, several units were listed as having limits higher than the proposed standard, or requiring less surveillance: Arkansas Nuclear One 1, Fort Calhoun, Turkey Point, and R.E. Ginna. For BWRs, it was stated that 20 plants have what are considered to be inadequate LCOs.

To date, several PWRs have adopted the new standard LCO while modifying their technical specifications at the same time as changes for some other, non-related, issue. This reduces the number of PWRs with no LCOs to 10, and those with inadequate LCOs to one. Twenty BWRs still have LCOs that are considered inadequate.

All of the PWRs listed above have remaining operating lifetimes in excess of 20 years. Any modification to the LCOs could be assumed to apply to this period. However, these were being adopted as changes to other technical specifications were made. Over a 20-year period, it is likely that all of the above plants would have adopted the LCOs. It will therefore be assumed that this adoption of the LCOs would have occurred linearly over the 20-year period. This effectively halves the number of years to 10 that the new standard will be in place by immediate adoption, as opposed to adoption when changes to other technical specifications are made.

For BWRs, a list of affected plants was not available. However, the average remaining operating lifetime from NUREG/CR-2800 (Andrews et al. 1983) is again over 20 years (25.2 yrs), so the above arguments can again be applied, resulting in 20 BWR units with an effective issue lifetime of 10 years.

REFERENCE (For Issue 74)

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.

Denton, H. R. May 20, 1974. Internal Atomic Energy Commission Memorandum to Joseph M. Hendrie. Proposede Technical Specification for Primary and Secondary Coolant Activity Limits. U.S. Nuclear Regulatory Commission, Washington, D.C.

(a) R. Wayne Houston. March 4, 1982. Memorandum to Thomas Novak. U.S. Nuclear Regulatory Commission, Washington, D.C.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 85, Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

The reliability of vacuum breaker valves, used to relieve vacuum formation resulting from steam condensation in BWR discharge lines after a safety relief valve opening and closure, has become an item of concern. The reliability of these vacuum breakers can be improved by establishing adequate design criteria and operability limits from dynamic model development, redesign, and validation testing. Vacuum breakers could then be modified or replaced to satisfy operability limits where necessary.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	38
OCCUPATIONAL DOSES:	
SIR Implementation =	34
SIR Operation/Maintenance =	170
Total of Above =	200
Accident Avoidance =	0

COST RESULTS ($1E+06)

INDUSTRY COSTS:	
SIR Implementation =	2.2
SIR Operation/Maintenance =	0.27
Total of Above =	2.5
Accident Avoidance =	0
NRC COSTS:	
SIR Development =	0.10
SIR Implementation Support =	0.10
SIR Operation/Maintenance Review =	0.27
Total of Above =	0.47

RELIABILITY OF VACUUM BREAKERS CONNECTED TO STEAM DISCHARGE LINES INSIDE BWR CONTAINMENTS

ISSUE 85

1.0 SAFETY ISSUE DESCRIPTION

The reliability of vacuum breaker (VB) valves connected to steam discharge lines inside BWR containments could be improved by 1) establishing design criteria and operability limits for these valves based on dynamic model development and validation, and 2) modifying or replacing VBs to satisfy these operability limits.

VB valves are provided to prevent vacuum formation in a pipeline resulting from steam condensation. Safety relief valves (SRVs) are mounted on the main steam line inside a BWR drywell. Each SRV discharge is piped through its own discharge line (tailpipe) to a point below the minimum water level in the primary containment suppression pool. A VB valve is installed on each discharge line to admit drywell air into the discharge line after SRV actuation and closure. This prevents a vacuum from forming in the discharge line as a result of the condensation of leftover steam. Water in the suppression pool then cannot be drawn up into the line. The VB valve is similar to a swing check valve, with a disk that swings on a hinge pin to open, and a spring to return the disk to a closed position.

Review of recent Licensee Event Reports (LERs) has shown that several VB valves from various vendors and in different plants have failed to operate properly indicating a potential generic design problem. The major concerns are that failure of a VB valve in an open or partially open position may cause steam discharge to the drywell. This increased steam challenge may affect safety-related valves and instruments as well as increase the temperature and pressure of the drywell environment. Additionally, it is possible that a subsequent SRV actuation associated with a damaged VB valve in a closed or nearly-closed position would reduce vacuum relief capability and could lead to hydrodynamic loads on the discharge line piping and to the suppression chamber in excess of design conditions, or may cause water hammer damage to the SRV.

Based on the information provided for this safety issue, it appears that the cyclic impact of the disk on the VB valve seat due to steam discharge and condensation during SRV actuation or leakage presents a loading condition which has not been adequately addressed in VB valve design and qualification requirements.

A damaged VB valve has also been found on a high-pressure coolant injection (HPCI), exhaust system. This VB is similar in size and design to the VBs on SRV discharge lines.

Failure of HPCI or RCIC turbine exhaust line VB's in the closed or near closed position can result in excessive hydrodynamic loads on the turbine exhaust line or the containment wet well structure or water hammer damage to the HPCI or RCIC Turbine System. Failure of these VB's in the open position can present the possibility of excessive containment pressurization. These failure modes have previously been addressed in Generic Issue 61, "SRV Discharge Line Failure in the Wetwell Airspace of Mark I and II Containments," and are not considered in this issue analysis. As a result, this issue is limited to only the effects of SRV VB failures.

Presently, there are no specific safety requirements for VB valves; and existing Standard Review Plans, Regulatory Guides, or rules do not address the operability of the VB valve. To assist in resolving this safety issue, a better understanding of the safety-related consequences of a failed 1VB valve would be helpful. T. Barkalow of the Tennessee Valley Authority has done a failure evaluation of the consequences of VB failures.[a] In addition, a preliminary case study report by S. Rubin of the Office for Analysis and Evaluation of Operational Data (AEOD) has recently been issued which illustrated the significant consequences VB valve failures can have in the course of an event involving system interactions.[b] However, in order to more effectively define the fundamental safety requirements for these VB valves, further evaluation is needed of the steam environment presented to the drywell from a failed (e.g., partially opened) VB valve, to determine whether any additional hydrodynamic loads to the discharge line and suppression chamber will be introduced.

It was assumed that the reliability of VB valves connected to steam discharge lines inside BWR containments would be improved through the establishment of adequate operability limits and design criteria for these valves. Currently, no safety requirements exist that ensure that VB valves will perform their opening and closing functions during plant transients. Since there is evidence to suggest that at least part of the cause of VB valve damage could be attributed to design deficiency, a validated VB computer model could be helpful for establishing the actual dynamic transient loads VB valves may receive. The presence of inadequate load specifications for VB valves also has implications concerning the adequacy of the qualification program(s) for these valves. Though no specific requirements or VB valve design criteria and qualification testing are established, dynamic computer models may be available to assist in design development, and testing programs are being used by some nuclear power plant licensees.[c]

(a) LER 83-007 (Updated Report) June 30, 1983 - Browns Ferry Unit 1. Basically, this report evaluated the effects of VB valve failure after SRV actuation.

(b) Preliminary Case Study Report for the Edwin I. Hatch Unit No. 2 Plant Systems Interaction Event on August 25, 1982 by AEOD August 1983. This report gave evidence indicating that the drywell pressure exceeded technical specifications due to steam discharge from an open or partially opened (failed) VB valve in conjunction with a SRV actuation.

(c) For example, the Hope Creek Generating Station's FSAR Section 3.9.1.2.5.4 describes two verified vacuum relief valve computer codes to analyze reflood and clearing transients in relief valve discharge lines equipped with VB valves. FSAR Section 3.9.3.2.7.2 lists the tests and analysis used for their VB valves.

Implementation of the VB model requires BWR plants to evaluate their VB valves on the basis of the new design criteria, and to modify or replace VB valves to satisfy the operability limits where necessary. Such modifications could include various types of engineering. The magnitude of the modifications would vary for each BWR, depending upon the final design criteria and the existing installations.

All BWR plants using VB valves connected to SRV steam discharge lines inside containment would be affected by this safety issue resolution (SIR). This resolution encompasses all BWRs.

2.0 SAFETY ISSUE RISK AND DOSE

PUBLIC RISK REDUCTION

This SIR will improve the reliability of VB valves resulting in a reduced frequency of failure. Current information indicates that the effect of a malfunctioning VB valve by itself is not a concern for public safety. In addition, if a VB valve has failed open, and a subsequent SRV actuation occurs (opening and closing), there will be some steam discharge to the drywell, but this event would not be detrimental (lead to core-melt). However, if a VB valve has failed open with a subsequent SRV opening, and the SRV fails open or does not properly reseat, a continuous steam blowdown to the drywell will occur. This event in conjunction with failures on other systems has some potential for core-melt. This situation results in LOCA conditions to the drywell, potentially causing high drywell pressure, temperature, or equipment failure, and contributing to possible containment failure sequences due to overpressurization.

For this fail-open mode of VB valve failure, two Grand Gulf accident sequences are affected. They are T_1PQE and $T_{23}PQE$, assumed to lead to release via the BWR-4 release category. This is the least severe of the BWR core-melt release categories and is judged to represent best the potential containment failure mode for these sequences. The additional VB valve failure will result in two new accident sequences, T_1PXQE and $T_{23}PXQE$, where X is the probability of a VB valve failure. This probability is the number of pertinent VB valve failures divided by the number of SRV demands over the data base and was determined as follows. From information received, over a 4.5 year time period (3/79 to 9/83) there have been 24 VB valve failures recorded from 24 operating BWRs. Of these 24 failures, 7 went undetected until a subsequent SRV actuation occurred. The SRV demand per BWR per year was assumed to be seven, considering the available information sources.[a] Thus the 7 pertinent VB valve failures divided by (24 BWRs times 4.5 years times 7 SRV demands per BWR per year) gives a probability of X = 0.0093.

(a) These sources included engineering judgment of NRC personnel and data from NUREG/CR-1363 (Hubble 1980), NUREG-0626 (NRC 1980), and BNL-31940 (BNL 1982).

The failure sequences of potential loading of the suppression chamber beyond design limits due to a failed VB valve in the closed position are considered in this part of the analysis. These possible sequences could also contribute to containment failure due to overpressurization. Attachment 1 contains these possible sequences.

For this analysis, the SIR is assumed to reduce VB valve failures by 90%.

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

OCCUPATIONAL DOSE

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

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments (85)

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

		N	$\bar{T}$(yr)
BWRs:	Operating	24	25.2
	Planned	20	30.0
	Total	44	27.4

3. Plants Selected for Analysis:

   Grand Gulf 1 - representative BWR

4. Parameters Affected by SIR:

   This issue involves a failed VB valve in conjunction with a malfunctioning SRV and failure of other systems leading to containment failure. A new parameter "X" has been identified for VB valve failure in the open or partially opened position. This new parameter "X" is also involved with VB valve failures in the closed or partially closed position sequences.

5. Base-Case Values for Affected Parameters:

   X = 0.0093 (see text).

   Also see Attachment 1.

TABLE 1. (cont'd.)

6. Affected Accident Sequences and Base-Case Frequencies:

The new accident sequences for VB failure in the open position are: T_1PXQE and T_{23}PXQE and are assumed to lead to releases via BWR-4 release category. In effect, each minimal cut set of the T_1PQE and T_{23}PQE sequences has the parameter X added to it, thereby creating the two new sequences T_1PXQE and T_{23}PXQE. Their frequencies are as follows:

$$T_1PXQE = \underbrace{(2.3E\text{-}07/py)}_{T_1PQE\ \text{frequency}} \quad \underbrace{(0.0093)}_{X} = 2.1E\text{-}09/py$$

$$T_{23}PXQE = \underbrace{(5.4E\text{-}07/py)}_{T_{23}PQE\ \text{frequency}} \quad \underbrace{(0.0093)}_{X} = 5.0E\text{-}09/py$$

Both contribute to BWR-4.

The new accident sequences for VB failure in the closed or nearly closed position are:

$$T_1(SRV)_1(SRV\text{-}2)_1(X)(Y)(FCON)(CM) = (0.2)(1.0)(1.0)(0.0093)(0.01)(1.0E\text{-}04)(0.1)$$

$$= 1.86E\text{-}10/py$$

$$T_{23}(SRV)_{23}(SRV\text{-}2)_{23}(X)(Y)(FCON)(CM) = (7.0)(0.8)(0.8)(0.0093)(0.01)(1.0E\text{-}04)(0.1)$$

$$= 4.166E\text{-}09/py$$

Both contribute to BWR-2.

See Attachment 1 for the development of these sequences.

7. Affected Release Categories and Base-Case Frequencies:

BWR-2 = 1.86E-10/py + 4.166E-09/py = 4.35E-09/py

BWR-4 = 2.1E-09/py + 5.0E-09/py = 7.1E-09/py

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

$\bar{F} = 0$

Issue resolution will affect the containment failure mode likelihood (which the term X, in effect, represents), not the core-melt frequency.

TABLE 1. (cont'd.)

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

$$W = (4.35E\text{-}09/py)(7.1E{+}06 \text{ man-rem}) + (7.1E\text{-}09/py)(6.1E{+}05 \text{ man-rem})$$

$$= 0.0352 \text{ man-rem/py}$$

10. Adjusted-Case Values for Affected Parameters

For this analysis, it is assumed that the issue resolution will result in a reduction in the frequency of VB valve failure by 90%. Thus X* = 0.10(0.0093) = 9.3E-04. This assumption is based on engineering judgment.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

T_1PXQE = 2.1E-10/py $\qquad$ $T_{23}PXqE$ = 5.0E-10/py

$T_1(SRV)_1(SRV\text{-}2)_1$ $\qquad$ $T_{23}(SRV)_{23}(SRV\text{-}2)_{23}$

(X)(Y)(FCON)(CM) $\qquad$ (X)(Y)(FCON)(CM)

= 1.86E-11/py $\qquad$ = 4.166E-10/py

12. Affected Release Categories and Adjusted-Case Frequencies:

BWR-2 = 4.35E-10/py

BWR-4 = 7.1E-10/py

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

$\bar{F}^* = 0$

Core-melt frequency is unaffected. Risk reduction will arise from reduced likelihood of containment failure.

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

W* = 3.09E-03 man-rem/py + 4.3E-04 man-rem/py = 3.5E-03 man-rem/py

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

$\Delta F = 0$

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

ΔW = 0.035 man-rem/py - 0.0035 man-rem/py
= 0.0315 man-rem/py

TABLE 1. (cont'd.)

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
38	1.3E+03	0

<u>ATTACHMENT 1 (To Table 1)</u>

No Grand Gulf dominant risk sequences were found that were appropriate for a VB valve failed in the closed or nearly closed position. Thus, two new accident sequences were identified. They are as follows:

$$(T_1)(SRV)_1(SRV\text{-}2)_1(X)(Y)(FCON)(CM)$$

and

$$(T_{23})(SRV)_{23}(SRV\text{-}2)_{23}(X)(Y)(FCON)(CM)$$

where:

T_1 = 0.2/py = frequency of a transient initiated by loss of offsite power; from the Grand Gulf PRA.

SRV_1 = 1.0/event = probability of SRV actuation for a T_1 transient; 1.0/event was assumed since loss of offsite power will result in closure of the MSIVs.

$SRV\text{-}2_1$ = 1.0/event = probability that an SRV, once actuated in response to T_1, will undergo a second opening; an upper bound of 1.0/event was assumed to bound this part of the analysis.

X = 0.0093/demand = probability of VB failure; derived from previous discussion.

Y = 0.01/VB failure = probability that a given VB failure will result in the valve remaining closed or nearly closed; 0.01/event was assumed because, although it is possible for this failure to occur, that is not expected. No instance of this type of failure has been observed.

FCON = 10^{-5} to 10^{-4}/demand = probability that the suppression chamber (wetwell) fails due to increased hydrodynamic loads. This range was assumed after discussion with members of the NRR staff from the Containment Systems Branch, Generic Issues Branch, Mechanical Engineering Branch, and the Structural and Geotechnical Engineering Branch.

CM = 0.1/event = probability that the transient escalates into a severe core-damage event because of suppression chamber failure. The same value was used in the evaluation of Generic Issue 61, "SRV Discharge Line Failure in the Wetwell Airspace of Mark I and II Containments." It was obtained by adjusting the probability of core-melt derived for a PWR with loss of recirculation coolant (0.25/demand) to account for the significantly greater volume of water available for injection from the condensate storage system in the BWR design as well as the availability of more pathways for getting that water to the reactor.

T_{23} = 7.0/py = frequency of all other transients resulting in reactor shutdown; from the Grand Gulf PRA.

SRV_{23} = 0.8/event = probability of SRV actuation for a T_{23} transient; 0.8/event was assumed because most BWRs can accommodate a trip from about 50% power with adequate heat rejection through the turbine bypass, and many trips occur during startup or at low power.

$SRV\text{-}2_{23}$ = 0.8/event = probability that a SRV once actuated in response to a T_{23} transient will undergo a second opening; 0.8/event was assumed to provide an upper bound of the effect for this part of the analysis.

Since failure of the suppression chamber due to hydrodynamic loading would occur early in the transient, it is assumed that the consequences of these failure sequences would be best approximated by the BWR-2 Release Category.

It should be noted that two design features of some BWR plants were not considered, which would result in a reduction of the calculated public risk. These are parallel VB valves and SRV low-low level reset logic. About half of the BWR plants have two VB valves per SRV discharge line in a parallel flow path arrangement (i.e., a redundant VB valve). Also, about half of the BWR plants have adopted a low-low reactor coolant level SRV reset logic as a means of reducing the number of second SRV openings.

In addition, the failure of a VB valve in the closed or nearly closed position, combined with a subsequent SRV actuation, would increase the load on the SRV discharge line. This type of sequence can fall into the category of Generic Issue 61, "SRV Discharge Line Failure in the Wetwell Airspace of Mark I and Mark II Containments". After discussion with NRC it has been decided that failure sequences involving VB valves in the closed or nearly closed position will be addressed in Generic Issue 61. However, this analysis was not changed to reflect this decision by NRC.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments. (85)

2. Affected Plants (N):

		N
BWR:	Operating	24
	Planned	20
	Total	44

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

		$\bar{T}$ (yr)
BWR:	Operating	25.2
	Planned	30.0
	All BWRs	27.4

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

$\Delta \bar{F}D_R = 0$

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

$\Delta U = 0$

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

The assumption of one VB for each SRV discharge line gives an average of 8 VBs per plant (165 lines in 22 operating plants). The labor in radiation zones for VB valve modification/replacement is assumed to be one man-week per plant.

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

The average dose rate around the VBs during plant shutdown is estimated to be approximately 35 mrem/hr.

D = (1 man-wk/plant) (40 man-hr/man-wk)(0.035 rem/hr)
= 1.4 man-rem/plant
(operating plants only)

TABLE 2. (cont'd.)

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

$ND = (1.4 \text{ man-rem/plant})(24 \text{ plants}) = 34 \text{ man-rem}$

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

It is assumed that further modifications and maintenance will acquire approximately 4 man-hr/py (10% of original implementation labor). This labor for operation and maintenance will apply to all BWRs.

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

$D_o = (4 \text{ man-hr/py})(0.035 \text{ rem/hr}) = 0.14 \text{ man-rem/py}$

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

$N\bar{T}D_o = (44 \text{ plants})(27.4 \text{ yr})(0.14 \text{ man-rem/py}) = 169 \text{ man-rem}$

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
200	610	68

3.0 SAFETY ISSUE COSTS

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

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments. (85)

2. Affected Plants (N):

		N
BWR:	Operating	24
	Planned	20
	Total	44

TABLE 3. (cont'd.)

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

		$\bar{T}$ (yr)
BWR:	Operating	25.2
	Planned	30.0
	All	27.4

Industry Costs (Steps 4 through 12)

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

$\Delta\bar{F}A = 0$

Core-melt frequency is unaffected.

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

$\Delta H = 0$

6. Per-Plant Industry Resources for SIR Implementation:

Modifying or replacing VBs on steam discharge lines to meet newly developed design criteria and operability limits is expected to require testing, labor, material, and QA control.

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

Resource	Cost ($/plant)
Testing	2K
Labor = 1 (man-wk/plant)($2270/man-wk)	2.3K
Materials = (8 VB valves/plant)[a]($5,000/valve)	40K
QA	5K
	I = 49.3K

These costs are based on conversations with Browns Ferry staff in reference to their recent testing and maintenance on the VB valves. These costs are believed conservative because cost savings due to avoided maintenance on a new valve design were not included.

(a) Assumes an average of 8 VB valves/plant over all BWRs

TABLE 3. (cont'd.)

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

$NI = (44 \text{ plants})(\$4.93E+04/\text{plant}) = \$2.2E+06$

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

It is estimated that approximately 4 man-hr/py will be required for operation (testing) and maintenance.

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

$$I_o = (4 \text{ man-hr/py})(1 \text{ man-wk}/40 \text{ man-hr})(\$2270/\text{man-wk}) \approx \$227/\text{py}$$

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

$$N\bar{T}I_o = (44 \text{ plants})(27.4 \text{ yr})(\$227/\text{py}) = \$2.7E+05$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$2.5E+06	\$3.6E+06	\$1.4E+06

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

The cost for development of new design criteria and establishment of operability limits for VBs will be sponsored by the NRC. It is assumed that a major portion of dynamic model development and engineering data is already available for use in establishing adequate VB design criteria.

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

$$C_D = (1 \text{ man-yr})(\$1.0E+05/\text{man-yr}) = \$1.0E+05$$

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

To support implementation by the industry, 1 man-week per plant is assumed.

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

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

TABLE 3. (cont'd.)

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

$$NC = (44 \text{ Plants})(\$2270/\text{plant}) = \$1.0E+05$$

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

Approximately 0.10 man-week per plant year is estimated for followup on operation and maintenance.

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

$$C_o = (0.10 \text{ man-wk/py})(\$2270/\text{man-wk}) = \$227/\text{py}$$

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

$$N\tilde{T}C_0 = (44 \text{ plants})(27.4 \text{ yr})(\$227/\text{py}) = \$2.7E+05$$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$4.7E+05	\$6.2E+05	\$3.2E+05

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 94, Additional Low-Temperature-Overpressure Protection for Light Water Reactors

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

This issue is concerned with low-temperature-overpressure (LTOP) transients in operating PWRs where the pressure-temperature limits on the reactor cooling systems are exceeded. Resulting thermal and pressure stresses if combined with a critical size crack could result in a brittle failure of the reactor vessel. Failure of the reactor vessel could make it impossible to provide adequate coolant to the core and result in major core damage or core-melt accident. Several resolutions of this generic issue have been suggested by Reactor Systems Branch (RSB) and Analysis and Evaluation of Operational Data (AEOD). These are discussed in the Proposed Resolution section of this report.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	6.5E+03
OCCUPATIONAL DOSES:	
SIR Implementation =	9.1E+02
SIR Operation/Maintenance =	0
Total of Above =	9.1E+02
Accident Avoidance =	31.2E+02

COST RESULTS ($1E+06)

INDUSTRY COSTS:	
SIR Implementation =	12
SIR Operation/Maintenance =	0
Total of Above =	12
Accident Avoidance =	9.5
NRC COSTS:	
SIR Development =	0.038
SIR Implementation Support =	0.47
SIR Operation/Maintenance Review =	0
Total of Above =	0.51

ADDITIONAL LOW-TEMPERATURE-OVERPRESSURE

PROTECTION FOR LIGHT WATER REACTORS

ISSUE 94

1.0 SAFETY ISSUE DESCRIPTION

This issue is concerned with pressurized water reactor PWR low-temperature-overpressure (LTOP) transients where the pressure/temperature on the reactor cooling system exceeds the technical specifications. Overpressure events primarily result from the loss of letdown flow with continued charging flow, inadvertent safety injection, or a heat-up transient caused by starting a reactor coolant pump with the secondary coolant system temperature higher than the primary temperature. Resulting thermal and pressure stresses may propagate cracks existing in pressure vessels of decreased fracture toughness. Neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. The continuation of overpressure transient events and especially the two instances at Turkey Point Unit 4 on November 28 and 29, 1983, during which the pressure exceeded technical specification limits (415 psig below 355°F) by about 700 and 325 psi, respectively, may indicate a potential weakness in the present overpressure protection criteria or its implementation which warrants further consideration. This issue is a concern only for operating PWRs. It is not significant for BWRs. BWRs operate with a large portion of the water inventory inside the pressure vessel at saturated conditions. Any sudden cooling will condense steam and result in a pressure decrease, so simultaneous creation of high pressure and low temperature is improbable.

PROPOSED RESOLUTION

The NRC staff is currently evaluating the LTOP issue. Current NRC staff requirements are defined in standard review plan (SRP) 5.2.2 "Overpressure Protection" and its attached BTP-RSB-5-2 "Overpressure Protection of Pressurized Water Reactors While Operating at Low Temperatures". Resolution of this generic issue was suggested by RSB and AEOD (Sheron 1984, Heltemes 1983) to include all or some of the following proposed new requirements:

a. Amend the standard technical specifications (STS) and the SRP to require each licensee to identify the criteria used to determine if and when the LTOP protection system (LTOP-PS) set points need to be adjusted to account for the irradiation induced embrittlement of the reactor vessel. The LTOP-PS consists of components that are required to prevent over pressurization of the primary system during low-temperature conditions.

b. Make more use of the relief valves in the residual heat removal system (RHRS) for LTOP protection by raising the set points for the auto-closure of the isolation valves.

c. Allow no plant operation in the "water solid" condition with either train of the LTOP-PS out of service.

d. Allow no plant operation in the "water solid" condition with a safety injection (SI) pump in service.

e. Require the LTOP-PS to be fully safety grade.

f. Upgrade the STS to be consistent with the resolution of this generic issue. Require all operating reactors to upgrade their technical specifications to the STS for LTOP-PS.

The impact of all of the above proposed resolutions are analyzed in this report. The costs of implementing these resolutions, the induced occupational doses incurred as the result of implementing these resolutions, and their impact on public risk reductions are also analyzed and reported in this study. Since discussions with the NRC reactor license examiners revealed that implementation of proposed resolutions C and D will not have any cost effects on the industry, these two proposed resolutions were excluded from the cost analysis.

Many of the corrective actions outlined above serve only to decrease the probability of occurrence of a LTOP event. They do not, however, reduce the consequences of a reactor vessel fracture or a core-melt. Reducing the probability of a vessel failure is accomplished by adjusting pressure/temperature limits specified in the technical specifications, flux reductions, and thermal annealing of the pressure vessel. The analyses of these issues is beyond the scope of this report. The methods of reducing the probability of vessel fracture due to LTOP transients are being analyzed in an ongoing NRC project concerned with the pressurized thermal shock (PTS) issue.

AFFECTED PLANTS

This safety issue is a concern only for operating PWRs. Since the existing data about the frequency of LTOP involves only operating PWRs, the planned PWRs were excluded from this analysis. The same requirements will apply to those plants as soon as they are on line. The number of operating PWRs is assumed to be 47. The average years of operation for these plants is 9 years, and their remaining lifetime is assumed to be 26 years. These values are extracted from Guidelines for Nuclear Power Plant Safety Issue Prioritization (Andrews et al. 1983). These data were adjusted by 1-1/2 years since the guidelines are 1-1/2 years old.

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:

Additional Low-Temperature-Overpressure Protection for Light Water Reactors (94)

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

	N*	$\bar{T}$ (yr)
PWR	47	26

*30 for the Oconee Type (1 PWR) and 17 for the "High" Type (2 PWR)

3. Plants Selected for Analysis:

Oconee-Type PWRs
High-Type PWRs - Those with higher reactor vessel copper and nickel contents than Oconee

4. Parameters Affected by SIR:

LTOP events were not considered in the original Oconee-3 study. The original study was modified by adding a LTOP sequence category "LTOP" for the base-case. See Attachment 1.

5. Base-Case Values for Affected Parameters:

See Step 6.

6. Affected Accident Sequences and Base-Case Frequencies:

Average frequencies for a LTOP event and vessel fracture are estimated to be 4.5E-06/py and 6.6E-06/py for Oconee-3 and "High" Case, respectively. The term "High" Case refers to a plant with a higher reactor vessel copper and nickel content than Oconee. This and the derived values above are discussed in more detail in Attachment 1. A breach of the reactor pressure vessel is assumed to result in core-melt with a probability of 1.0. The containment failure modes, likelihoods, and release categories are assumed to be the same as for sequence S_1D in Appendix A of the Guidelines for Nuclear Power Plant Safety Issue Prioritization (Andrews et al. 1983).

		Oconee-3/py	"High" Case/py
LTOP-	(PWR-1)	4.5E-08	6.6E-08
LTOP-	(PWR-3)	9.0E-07	1.3E-06
LTOP-	(PWR-5)	3.3E-08	4.8E-08
LTOP-	(PWR-7)	3.6E-06	5.3E-06

TABLE 1. (cont'd.)

7. Affected Release Categories and Base-Case Frequencies:

	Oconee-3/py	"High" Case/py
(PWR-1)	4.5E-08	6.6E-08
(PWR-3)	9.0E-07	1.3E-06
(PWR-5)	3.3E-08	4.8E-08
(PWR-7)	3.6E-06	5.3E-06

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

$\bar{F}_{Oconee}$ = 4.5E-06/py

$\bar{F}_{High}$ = 6.6E-06/py

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

W_{Oconee} = 5.1 man-rem/py

W_{High} = 7.4 man-rem/py

10. Adjusted-Case Values for Affected Parameters:

The frequency of a LTOP event is assumed to be reduced to 0.001/py.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

		Oconee-3/py	"High" Case/py
LTOP-	(PWR-1)	4.5E-09	6.6E-09
LTOP-	(PWR-3)	9.0E-08	1.3E-07
LTOP-	(PWR-5)	3.3E-09	4.8E-09
LTOP-	(PWR-7)	3.6E-07	5.3E-07

12. Affected Release Categories and Adjusted-Case Frequencies:

	Oconee-3/py	"High" Case/py
PWR-1	4.5E-09	6.6E-09
PWR-3	9.0E-08	1.3E-07
PWR-5	3.3E-09	4.8E-09
PWR-7	3.6E-07	5.3E-07

TABLE 1. (cont'd.)

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

$\bar{F}^*_{Oconee}$ = 4.5E-07/py

$\bar{F}^*_{High}$ = 6.6E-07/py

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

W^*_{Oconee} = 0.51 man-rem/py

W^*_{High} = 0.74 man-rem/py

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

$\Delta\bar{F}_{Oconee}$ = (4.5E-06/py)-(4.5E-07/py) = 4.05E-06/py

$\Delta\bar{F}_{High}$ = (6.6E-06/py)-(6.6E-07/py) = 5.9E-06/py

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

ΔW_{Oconee} = (5.10/py)-(0.51 man-rem/py) = 4.59 man-rem/py

ΔW_{High} = (7.40 man-rem/py)-(0.74 man-rem/py) = 6.67 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
6.5E+03	2.2E+05	0

ATTACHMENT 1 (To Table 1)

Before 1979, there were 30 reported events in PWRs where the pressure/temperature of the reactor coolant system violated Technical Specifications. After 1979, following changes to operating procedures and the implementation of Overpressure Mitigation System (OMS), there have been two reported events of overpressure excursions at low temperatures. Since 1979, PWRs have accumulated approximately 20 plant years of operating time. Therefore, the currently expected frequency of overpressure excursion events is .01/py.

A potential safety effect of a LTOP event is that if combined with a critical size crack it could result in a brittle failure of the reactor vessel. The Vessel Integrity Simulation Analysis (VISA) code (D.L. Stevens et al. 1983) was used to quantify the failure probability of the reactor vessel due to a LTOP event.

Past experiences with the pressurized thermal shock (PTS) issue and LTOP events analysis have revealed that the copper and nickel contents of a reactor vessel and the fluence level have a direct effect on the integrity of the reactor vessel. The higher the copper and nickel contents and the higher the fluence levels, the more likely the vessel will fail due to an LTOP event. Therefore, two types of reactor vessels have been analyzed in this study. The first one is Oconee-3 power plant. The copper and nickel contents of the Oconee-3 reactor vessel and its fluence level are similar to those of the majority of the operating PWRs in the U.S. The copper and nickel contents of the Oconee-3 vessel and its fluence level are as follows:

$$CU\% = 0.20 \quad NI\% = 0.63 \quad \text{Fluence} = 2.9E{+}18 \text{ neutrons/cm}^2$$

The second type of vessel analyzed is the one with a high copper and nickel content. The following values were the copper and nickel contents used for the "High" case.

$$CU\% = 0.35 \quad NI\% = 1.00 \quad \text{Fluence} = 2.9E{+}18 \text{ neutrons/cm}^2$$

From the information available on 38 PWR reactors, the copper and nickel content of 14 of these reactors was higher than the Oconee-3 copper and nickel content. This translates to approximately 38 percent of the PWRs having higher copper and nickel content than Oconee-3. It was thus assumed that 38 percent of the operating PWRs or 17 PWRs are similar to "High" plant and the remaining 30 PWRs are similar to Oconee-3.

Modifications were also made to VISA in order to obtain estimates of the reactor vessel failure probability. The main modification was to revise the flaw size distribution used in the code. A listing of the flaw size distribution used in VISA follows:

ATTACHMENT 1. (cont'd.)

Flaw Size (inches)	Probability
0.000	0.91767661
0.125	0.05507015
0.250	0.02256947
0.500	0.00422063
1.000	0.00036718
1.500	0.00007085
2.000	0.00001667
2.500	0.00000500
3.000	0.00000250
3.500	0.00000083

This distribution was used during the preliminary stages of our analysis. However, due to the small probabilities given in the distribution, no reactor vessel failure was simulated by VISA. Therefore, to get any kind of an estimate, we assumed the probability of a 1/4 T (thickness) flaw is 1.0. For our purposes the thickness of the reactor vessel was assumed to be 8 inches. The failure probability results were then reduced by a factor of 2256 to adjust for the new flaw size distribution. This value was obtained by the ratio of the results given by two VISA runs. The first with the adjusted flaw size distribution and the second with the original flaw size distribution. These estimates were then multiplied by six to account for the six welds on the reactor vessel belt line.

The pressure/temperature limits of the two representative PWRs were constructed using the given copper and nickel contents of the pressure vessel, fluence, the guidelines given in Appendix G to Section III of the ASME code and Revision 2 of the Regulatory Guide 1.99.

The reactor vessel and weld materials have toughness properties which are defined by the nil ductility transition reference temperature RT(NDT). The higher the copper and nickel content, the higher the RT(NDT). The RT(NDT) also increases with fluence or the cumulative exposure of the vessel failure due to a pressure spike is a function of the initial temperature T in relation to RT(NDT) or T-RT(NDT). The failure probability is also a function of the initial pressure and the change in pressure.

Based upon a review of the previous overpressure events prior to 1978, it was found that 30 percent reached a peak pressure that was between 1100 psia and 2486 psia. In another 5 percent of these events, the peak pressure was between 950 psia and 1200 psia. In the remaining 65 percent, pressure was prevented from exceeding 950 psia by operator actions. Thus, a series of VISA code runs were made at 2485 psia, 1200 psia, and 950 psia to obtain the probability of reactor vessel failure as a funcion of T-RT(NDT).

At the midlife of the vessels the fluence is estimated to be 8.5E+18 neutrons per square centimeter. This fluence converts to a RT(NDT) or 267^{o}F for the "high" vessel and 231^{o}F for the Oconee-type vessel using the methodology contained in Rev. 2 to Regulatory Guide 1.99. If it is assumed that the starting temperature is 110°F, the T-RT(NDT) value is about -150°F for the "high" vessel and -120°F for the Oconee-type vessel. Table A.1 presents the vessel failure probabilities for these values of T-RT(NDT).

TABLE A.1. Vessel Failure Probability

Peak Pressure	Probability of failure, per event: Oconee Vessel	"High" Vessel
2,485 psia	1.5E-03	2.2E-03
1,200 psia	7E-07	7E-06
950 psia	<1E-09	<1E-09

The predicted probability of failure at 2485 psia peak surge at the end of life fluence (1.4E+19 neutrons/cm^2) for Oconee-type vessel is 2.6E-03 and 2.7E-03 for the "High" type vessel.

The average failure frequency for the Oconee-type vessels is calculated to be 4.5E-06/py and 6.6E-06/py for the "High" type vessel.

It is assumed that a breach of the reactor pressure vessel would result in core-melt with a probability of 1.0. Therefore, for our base case the core-melt frequency for Oconee-3 is 4.5E-06/py and for the "High" plant is 0.6E-06/py. It is necessary to mention that implementation of proposed new requirements (discussed in Proposed Resolution Section) by RSB and AEOD will only decrease the frequency or probability of a LTOP event from occurring. It does not, however, affect the probability of a vessel failure once the LTOP has occurred.

Since the main reasons for the occurrence of a LTOP event are either equipment malfunctions or human error, it was assumed that implementation of the proposed new requirements will decrease the frequency of a LTOP event by a factor of 10. It is believed that implementation of all of the proposed resolutions will significantly reduce the human error and will improve the reliability of the protection system. Therefore a 90 percent improvement factor is reasonable and the adjusted core-melt frequency is:

Oconee-3 = (0.1)(4.5E-06/py) = 4.5E-07/py

"High" Plant = (0.1)(6.6E-06/py) = 6.6E-07/py

The public risk reduction analysis was done by assuming that the containment failure modes, likelihoods, and release categories are the same as for sequence S_1O in Appendix A of the Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. (Andrews et al. 1983). These are as follows:

	Oconee-3		"High" Plant	
	Base-Case (/py)	Adjusted-Case (/py)	Base-Case (/py)	Adjusted-Case (/py)
PWR-1 =	4.5E-08	4.5E-09	6.6E-08	6.6E-09
PWR-3 =	9.0E-07	9.0E-08	1.3E-06	1.3E-07
PWR-5 =	3.3E-08	3.3E-09	5.0E-08	5.0E-09
PWR-7 =	3.6E-06	3.6E-07	5.3E-06	5.3E-07

The public dose consequence factors were taken from Appendix D of Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. (Andrews et al. 1983). Table A.2 represents these factors.

TABLE A.2. Public Dose Consequence Factors[a]

Category	Whole Body Dose Consequence Factor (man-rem) Core-Melt
PWR-1	5.4E+06
PWR-2	4.8E+06
PWR-3	5.4E+06
PWR-4	2.7E+06
PWR-5	1.0E+06
PWR-6	1.5E+05
PWR-7	2.3E+03

(a) From CRAC, with guidelines and quantities of radioactive isotopes used in WASH-1400. Estimates are based on the meteorology of a typical midwest site (Byron-Braidwood) with a uniform population density of 340 people/square mile, no evacuation and 50-mile radius mode.

The base case public risk was estimated by multiplying the release category values by the values given in Table A.1. Therefore the base case public risk is as follows:

Oconee-3 Base-Case Public risk = (W_1) = (4.5E-08/py)(5.4E+06 man-rem)+(9.0E-07/py)(5.4E+06 man-rem)+(3.3E-08/py)(1.0E+06 man-rem)+(3.6E-06/py)(2.3E+03 man-rem) = 5.1 man-rem/py

"High" Plant Base-Case Public Risk = (W_2) = (6.6E-08/py)(5.4E+06 man-rem)+(1.3E-08/py)(5.4E+06 man-rem)+(5.0E-08/py)(1.0E+06 man-rem)+(5.3E-06/py)(2.3E+03 man-rem) = 7.4 man-rem/py

ATTACHMENT 1. (cont'd.)

The adjusted case public risk (W^*_1) for Oconee-3 and (W^*_2) for "High" Plant were calculated in a similar manner and their values are 0.51 man-rem/py and 0.74 man-rem/py, respectively.

The occupational exposure as the result of implementing the proposed requirements are calculated by estimating the time required to make the plant/equipment modifications. The time required for raising the set points for the auto-closure of the isolation values (Requirement B) is estimated to be 1 man-day. This estimate was derived after discussions with the NRC License examiners and other industry contacts.

The occupational exposure and the costs involved in implementing Requirement E, requiring the LTOP system to be fully safety grade, were taken from the results of analysis done on Safety Issue 70, which estimates the costs and the occupational exposures involved in requiring the power operated relief valves (PORVs) which are the main part of the LTOP-PS to be fully safety grade. Although the PORVs are used in LTOP and HPIC Systems, its full dose and cost estimates were taken for this safety issue.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Additional Low-Temperature-Overpressure Protection for Light Water Reactors. (94)

2. Affected Plants (N):

47 Operating PWRs*

*30 for the Oconee-Type (1 PWR) and 17 for the "High" Type (2 PWRs)

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

The average remaining life for each of the 47 PWRs is assumed to be 26 years.

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

Oconee-Type: (19,900 man-rem)(4.05E-06/py) = 8.1E-02 man-rem/py

"High" Type: (19,900 man-rem)(5.90E-06/py) = 1.2E-01 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
1.2E+02	7.7E+02	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 raising relief valves set points will require 8 man-hr. The dose exposure inside the containment building (shutdown mode) is given as 25 mR/hr in Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. (Andrews et al. 1983). Therefore total occupational dose for this task (per-plant) is:

$$D = (2.5E\text{-}02R/hr)(8 \text{ man-hr plant}) = 0.2 \text{ man-rem/plant}$$

Safety issue 70 estimates the occupational dose for making the PORVs that are used in LTOP Protection System fully safety grade to be 19.2 man-rem/plant. Therefore total occupational dose for these two modifications (per plant) is:

$$D = (19.2 \text{ man-rem/plant})+(0.2 \text{ man-rem/plant}) = 19.4 \text{ man-rem/plant}$$

TABLE 2. (cont'd.)

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

$$ND = (47)(19.4 \text{ man-rem/plant}) = 910 \text{ man-rem}$$

9-11. Steps Related to Occupational Dose Increase for SIR Operation and Maintenance:

The equipment modifications are assumed not to require any additional operation/maintenance beyond that normally required for the vessel. Therefore, $D_o = 0$.

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
9.1E+02	2.7E+03	3.0E+02

3.0 SAFETY ISSUE COSTS

Results of the analysis of industry costs and NRC costs are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Additional Low-Temperature-Overpressure Protection for Light Water Reactors (94)

2. Affected Plants (N):

47 Operating PWRs*

*30 for the Oconee-Type and 17 for the "High Type"

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

		$\bar{T}$ (yr)
PWRs:	Operating	26

TABLE 3. (cont'd.)

Industry Costs (Steps 4 through 12)

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

$$\bar{F}A_{1PWR} = (\$1.65E+09)(\$4.05E-06/py) = \$6.7E+03/py$$

$$\bar{F}A_{2PWR} = (\$1.65E+09)(\$5.90E-06/py) = \$9.7E+03/py$$

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

Best Estimate	Upper Bound	Lower Bound
\$9.5E+06	\$1.8E+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 industry will expend resources for LTOP analysis, technical specifications adjustments, and plant/equipment modifications. Amending STS and SRP is estimated to require 8 man-wk/plant. Therefore this cost is estimated to be:

(8 man-wk/plant)(\$2270/man-week) = \$1.82E+04

Adjusting the set points on relief valves is estimated to require 8 man-hr. This estimate was derived after discussions with NRC license examiners. Therefore this cost is:

(8 man-hr/plant)(1 man-wk/40 man-hr)(\$2270/man-wk) = \$4.54E+02

Upgrading STS was estimated to take 2 man-wk and the costs associated with that is (2 man-wk)(\$2270/man-wk) = \$4.5E+03. The cost of requiring LTOP system to be fully safety grade is assumed to be similar to the results obtained in Safety Issue 70. Although PORVs are used for both the LTOP and HPIC systems, the full cost of requiring them to be fully safety grade (done in Issue 70) was assumed for LTOP. This cost is estimated to be \$2.37E+05/plant. Therefore the total cost of SIR Implementation per-plant is:

(\$1.82E+04)+(\$4.54E+02)+(4.5E+03)+(\$2.37E+05) = \$2.6E+05

TABLE 3. (cont'd.)

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

$NI = (47\ PWRs)(\$2.6E+05/plant) = \$1.2E+07$

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, $I_o = 0$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
1.2E+07	1.8E+07	6.0E+06

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Cost is estimated directly in Step 14.

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

It is estimated that total NRC labor requirement for SIR development is 8 man-wk. This translates to (8 man-wk) ($2270/man-wk) = $1.82E+04. The contract for SIR Research Program is assumed to be $2.0E+04. Therefore total NRC cost for SIR Development is:

($1.82E+04) + ($2.0E+04) = $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 reviewing the required LTOP analysis and addressing licensing codes and standards issues are estimated to be $1.0E+04. This estimate includes a required 4 man-wk/plant of NRC labor at $2270/man-week.

TABLE 3. (cont'd.)

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

$$NC = (\$1.0E+04/plant)(47) = \$4.7E+05$$

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 vessel. Therefore, $C_o = 0$.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$5.1E+05	\$7.4E+05	\$2.7E+05

REFERENCES (for Issue 94)

Appendix G to the Section III of the ASME Code.

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 99 RCS/RHR Suction Line Interlock on PWRs

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

In the existing PWRs, two interlock channels are provided so that one channel is used to interlock the operation of one residual heat removal (RHR) suction valve and the other channel is used for the other valve. When either channel is in a tripped state its associated suction valve will automatically close if it is open at the time. Since the relays used for this interlock are deenergized to initiate valve closure, a loss of the instrument bus used for either channel will result in a loss of RHR cooling due to inadvertent closure of one of the suction valves. The main safety concern in this issue is that the loss of one instrument bus will result in the automatic closure of one of the RHR suction line isolation valves. When in the RHR cooling mode, such closures can result in RHR pump damage and loss of decay heat removal by the RHR system. The NRC staff has identified a four part proposed resolution to deal with this safety issue. This resolution is discussed in the subsection entitled Proposed Safety Issue Resolution in section 1.0 of this report.

AFFECTED PLANTS	PWR:	Operating = 30	Planned = 28
	BWR:	Operating = 0	Planned = 0

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 2.0E+04

OCCUPATIONAL DOSES:

SIR Implementation =	1.3E+02
SIR Operation/Maintenance =	0
Total of Above =	1.3E+02
Accident Avoidance =	1.47E+02

COST RESULTS ($1E+06)

INDUSTRY COSTS:

SIR Implementation =	2.7
SIR Operation/Maintenance =	0
Total of Above =	2.7
Accident Avoidance =	12.2

NRC COSTS:

SIR Development =	0.018
SIR Implementation Support =	0.79
SIR Operation/Maintenance Review =	0
Total of Above =	0.81

RCS/RHR SUCTION LINE INTERLOCK ON PWRs

ISSUE 99

1.0 SAFETY ISSUE DESCRIPTION

It has been the common perception that the RHR interlocks provide overpressure protection for the RHR system so that a reactor coolant system (RCS) pressure transient will not result in a LOCA outside containment when in the RHR mode of cooling. While it is true that the interlocks provide an automatic closure signal to the RHR suction valves on high reactor coolant system pressure, overpressure protection of the RHR system is provided by relief valves. The purpose of the interlocks is to assure that there is double barrier (two closed valves) between the RCS and the RHR system when the plant is at normal operating condition and also to preclude conditions that could lead to a LOCA outside of containment due to operator error. The interlock safety function is not to isolate the RHR system from the RCS when the RHR system is operating in decay heat removal mode.

In existing PWRs, two interlock channels are provided such that one channel is used to interlock the operation of one RHR suction valve and the other channel is used for the other valve. When either channel is in a tripped state its associated suction valve will automatically close if it is open. Since the relays used for this interlock are deenergized to initiate valve closure, a loss of the instrument bus used for either channel will result in a loss of RHR cooling due to inadvertent closure of one of the suction valves.

PROPOSED SAFETY ISSUE RESOLUTION

The main safety concern related to this issue is that the loss of one instrument bus or disablement of one logic channel will result in automatic closure of one of the RHR suction line valves. When in the RHR cooling mode, a closure such as described above gives rise to the potential for RHR pump damage and loss of decay heat removal by the RHR system.

The NRC Staff has become increasingly aware of the number of events which have resulted in the loss of RHR cooling during plant shutdown conditions. Therefore, a four part proposed resolution has been recommended. This proposed resolution is briefly described below:

A. Design Bases. This action addresses the review and documentation of the design basis of RHR suction valve interlocks.

B. Interim Operation Procedures. This action addresses temporary interim modifications and operating procedures to reduce the potential for inadvertent isolation of the RHR suction valves during the RHR mode of cooling as well as assuring that a challenge will not occur which would require the interlock function.

C. System Design Modifications. This action addresses permanent modifications to reduce the potential for inadvertent isolation of the RHR system.

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 PWRs. The BWRs are not affected by this issue.

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:

   RCS/RHR Suction Line Interlock on PWRs (99)

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

		N	$\bar{T}$ (yr)
PWRs:	Operating	30	27.7
	Planned	28	30.0

3. Plants Selected for Analysis:

   Sequoyah 1 is chosen as the representative PWR.

4. Parameters Affected by SIR:

   RHR/RCS Suction Line Interlock was not considered in the Sequoyah 1 original study (Carlson et al. 1981). A new sequence is developed to analyze the impacts of inadvertent closure of the RHR Suction Line valve on plant safety. See Attachment 1.

5. Base-Case Values for Affected Parameters:

   See Step 6.

TABLE 1. (cont'd.)

6. Affected Accident Sequences and Base-Case Frequencies:

Weighted average frequency for the core-melt due to the loss of RHR cooling is estimated to be 4.8E-06/py. This value is derived in Attachment 1. The containment failure modes, likelihoods, and release categories are assumed to be the same as for T_1MLU of Oconee RSSMAP.

Sequence	Base-Case Frequency
RHR	4.8E-06/py

7. Affected Release Categories and Base-Case Frequencies:

PWR-3 = 2.4E-06/py

PWR-5 = 3.5E-08/py

PWR-7 = 2.4E-06/py

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

$\bar{F}_{PWR}$ = 4.8E-06/py

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

W_{PWR} = 13 man-rem/py

10. Adjusted-Case Values for Affected Parameters

See Attachment 1

11. Affected Accident Sequences and Adjusted-Case Frequencies:

Sequence	Adjusted-Case Frequency (1/py)
RHR	3.6E-07/py

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-3 = 1.8E-07/py

PWR-5 = 3.0E-09/py

PWR-7 = 1.8E-07/py

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

$\bar{F}^*_{PWR}$ = 3.6E-07/py

TABLE 1. (cont'd.)

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

$$W^*_{PWR} = 1\text{man-rem/py}$$

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

$$\Delta\bar{F}_{PWR} = (4.8E\text{-}06/\text{py}) - (3.6E\text{-}07/\text{py})$$

$$= 4.4E\text{-}06/\text{py}$$

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

$$\Delta W_{PWR} = (13 \text{ man-rem/py}) - (1 \text{ man-rem/py})$$

$$= 12 \text{ man-rem/py}$$

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.0E+04	6.5E+05	0

ATTACHMENT 1 (To Table 1)

In the existing PWRs, two interlock channels are provided so that one channel is used to interlock the operation of one RHR suction valve and the other channel is used for the other valve. When either channel is in a tripped state its associated suction valve will automatically close. Since the relays used for this interlock are deenergized to initiate valve closure, a loss of the instrument bus used for either channel will result in a loss of RHR cooling due to inadvertent closure of one of the suction valves.

The primary safety concern in this issue is that the loss of one instrument bus will result in the automatic closure of one of the RHR suction line isolation valves. When in the RHR cooling mode, this type of closure gives rise to the potential for RHR pump damage and loss of decay heat removal by the RHR system.

For the purposes of this analysis the Sequoyah 1 has been selected as the representative PWR.

To quantify the risks associated with the loss of RHR cooling , an attempt was made to quantify the probability of a core-melt resulting from the loss of RHR cooling. An event tree was developed based on information obtained from Sequoyah 1 Reactor Safety Study Methodology Applications Program (RSSMAP) PRA (Carlson et al. 1981) and also the discussions held with industry experts. This event tree is representative of the sequence of events postulated to occur from the time RHR cooling is lost to a possible core-melt. Figure 1 shows this event tree.

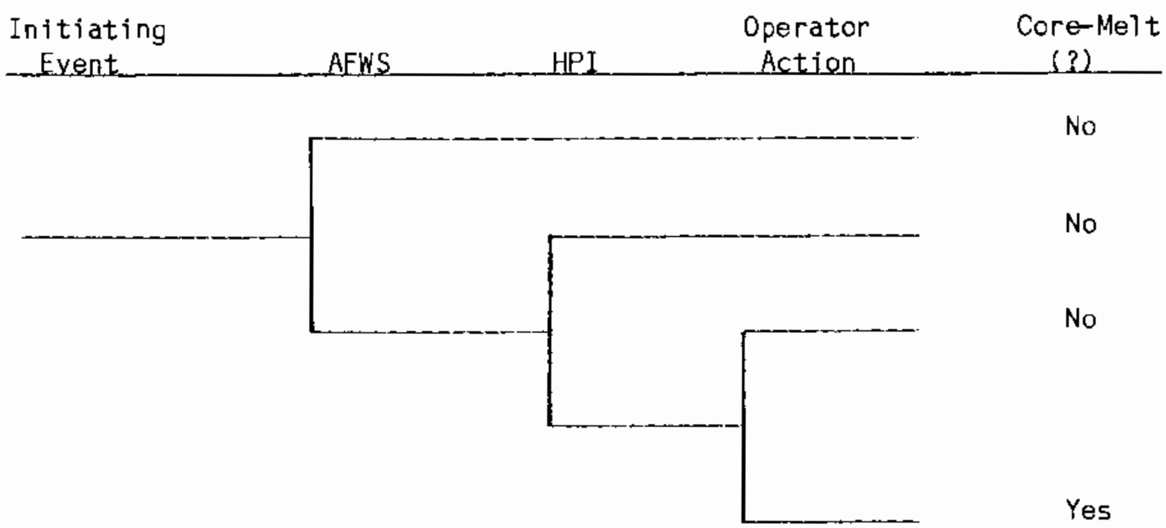

FIGURE 1. Postulated Event Tree For Accident Progression To Core-melt Due To The Loss Of Decay Heat Removal System

Following is a discussion of the events and their assumed probabilities:

INITIATING EVENT

The initiating event is the inadvertent closure of the RHR suction line valve while in the RHR cooling mode. As was mentioned previously, in PWRs two interlock channels are provided for the operation of two RHR suction valves. Since the relays used for these interlocks are deenergized to initiate valve closure, a loss of the instrument bus used for either channel or disablement of one logic channel will result in the automatic closure of one of the RHR suction line isolation valves.

A review of the existing literature about the instrument busses and their failure modes was conducted. This subject has been analyzed in Residual Heat Removal Experience Review and Safety Analysis, Pressurized Water Reactors, NSAC-52 (Vine et al. 1983). Two principle causes of bus failures were identified. These two are: 1) failure to provide DC power on demand as characterized by the loss of charger output coincident with the unavailability of DC power from the batteries; 2) operational, test, or maintenance errors resulting in the loss of DC power during normal plant operation. Of the 27 recorded losses of RHR flow due to suction valve closure, two had occurred as the result of a pressure rise in the primary, and the other 25 events resulted from causes other than an actual pressure rise. These 25 events occurred during 206 reactor years of operating experience at pressurized water reactors. The frequency of unplanned RHR suction valve closure is then estimated to be 0.12/py. Of the 25 events, 22 involved closure of only one valve and three events resulted in closure of both valves. Thus, 88 percent of the reported events were independent channel failures and 12 percent can be potentially classified as common cause related.

AUXILIARY FEEDWATER SYSTEM

The auxiliary feedwater system (AFWS) is one of the systems providing heat transfer to the environment during cooldown of the RCS to around 150^{O}F and 400 psia. The Sequoyah 1 AFWS consists of two 440-gpm electric pumps and one 880-gpm turbine-powered pump, along with associated piping, valves, and controls. The system delivers feedwater from two 385,000-gallon condensate storage tanks to the secondary side of the four steam generators in the event that normal feedwater is lost. Reviewing the available literature has made it clear that in an event when the RHR cooling capability is lost, the steam generators can be used to remove the generated heat. If this is the case, then there is no danger of overheating and core-melt. But, if the steam generators can not remove the generated heat, then there is a possibility of core overheating and core-melt.

There are several mechanisms by which the steam generators can fail to remove the generated heat. The most important ones are steam generator tube rupture (SGTR) or the failure of the AFWS to deliver water to the steam generators. The latter of these two is considered to be the dominant reason for failure of steam generators to remove the generated heat. SGTR was considered to be highly unlikely in this case and has not been considered. Instead the failure of delivering water to the steam generators is analyzed in more detail.

Unavailability of AFWS has been analyzed (Sequoyah 1 RSSMAP PRA) (Carlson et al. 1981) for three initiating events: small pipe break or transients excluding loss of offsite power; loss of offsite power; and high energy break. The associated probabilities for each one of these events are <1.0E-05, 1.90E-02, and 2.2E-02, respectively. Therefore, the total probability for the unavailability of the AFWS is estimated to be 4.1E-02. To be conservative in this analysis, it is assumed that the probability of AFW system unavailability is 0.1.

HIGH PRESSURE INJECTION SYSTEM (HPIS)

The HPIS design used in this analysis consists of three electric motor-driven high pressure injection pumps, associated piping, valves and controls. These pumps normally draw water from the refueling water storage tank and inject this borated water into the reactor cold legs at normal primary system pressure. For most small loss of coolant accidents and transient conditions requiring high pressure makeup water, the flow from one charging pump is sufficient for successful operation.

In the case the AFWS is not able to adequately remove the generated heat, decay heat, the HPIS injects water into the primary system to keep the pressure from getting too high and also to keep the water temperature down thereby providing more time for the operator to take action; i.e., opening the valve manually.

The failure of the HPIS is estimated by the ORNL Precursor Study (Minarick and Kukielka 1982) to be 1.3E-03.

OPERATOR ACTION

The operator can end this transient by manually opening the valve. Reviewing the related literature, Residual Heat Removal Experience Review and Safety Analysis, NSAC/52 (Vine et al. 1983), and also the discussions with NRC licensed examiners have made it clear that if the operator recognizes the problem correctly, given adequate time he/she can successfully re-open the valve. A review of the documented events, NSAC/52, revealed that approximately 96 percent of the time the operator has been able to open the valve manually. Therefore, a probability of 0.04 is assigned to unsuccessful operator attempt to re-open the valve manually.

It is necessary to mention that the timing of the valve closure is most important. If the reactor has been in the RHR mode of cooling for some time, then the operator has more time to react, but if the valve is closed as soon as the RHR cooling mode is initiated, then the operator has less time to lose since even with the reactor shut down there is still a significant amount of heat being generated by the delay neutrons. This decay heat is about 2-5 percent of the plant thermal output.

Applying probabilities developed above to the events shown in Figure 1 gives a frequency of (0.12/py)(.1)(.01)(0.04) or 4.8E-06/py for core-melt caused by the loss of RHR cooling. Therefore, a base-case core-melt frequency of 4.8E-06/py is obtained.

The adjusted-case, core-melt frequency is obtained by considering how much of an impact the proposed resolutions will have on preventing inadvertent closure of the RHR suction line valves.

Following are more detailed discussions of the safety issue resolutions proposed by the NRC staff.

A. REVIEW AND DOCUMENTATION OF THE DESIGN BASIS OF RHR SUCTION VALVE INTERLOCKS

A review of the design of RHR suction valve interlocks is required to be completed and a report submitted to document the design basis and conclusion of this review. The report must include the following information:

1. A description of the RHR suction valve interlock including a functional block diagram of the system and associated electrical schematics.

2. A failure mode and effects analysis of the interlock system.

3. A scenario description including control room indication and alarms as well as operator or other systems response to:

   a) An overpressure transient during the RHR mode of cooling.

   b) Inadvertent closure of the RHR suction valve(s) during the decay heat removal mode when the RHR system is in service.

4. A description of actions performed by components of RHR suction valve interlocks which are not directly related to the inter-lock function, e.g., use of the same RCS pressure measurements for the mitigation of low temperature/overpressure (LTOP) of the RCS during low temperature operation or other control or alarm function.

5. If the loss of an instrument bus associated with RHR interlocks will result in inadvertent isolation of the RHR system, a description of any other consequential failures of control or indications systems which would occur due to the loss of this power source, i.e., LTOPs, RHR minimum flow recirculation control, RCS temperature pressure indication or alarms, etc.

ATTACHMENT 1. (cont'd.)

6. A description of the use of power lockout for RHR suction valves for all modes of operation.

7. A summary of the design basis of RHR interlock including the extent of conformance to and applicable regulatory guidance to the plant design.

B. INTERIM OPERATING PROCEDURES

If a loss of an instrument bus would result in inadvertent isolation of RHR suction valves during the RHR cooling mode, interim operating procedures shall be instituted to preclude such events. These might include: the use of power lockout for RHR suction valves, temporary modifications to defeat the autoclosure feature of RHR interlocks, and appropriate instructions and training for plant operators on the impact and consequences of actions taken.

C. SYSTEM DESIGN MODIFICATIONS

The design of RHR suction valves interlocks shall be modified so that single failures will not result in inadvertent valve closure during the RHR cooling mode. One function of the autoclosure feature is to preclude an operator error which could result in one of two series valve being inadvertently left open during a plant startup. In most interlock designs this is a single channel of protection and the single failure criterion is not applicable to the design. Therefore, one solution would be to configure the logic so that it would operate on the basis of 2 out of 2 rather than 1 out of 1. In addition to this modification, several other plant modifications could be beneficial. For example, loss of RHR flow alarms may be beneficial to indicate a loss of RHR cooling . Additional pressure alarms may be beneficial to indicate pressure transients.

D. TECHNICAL SPECIFICATION CHANGES

Many plant technical specifications do not address RHR interlocks requirements for operability or surveillance. Therefore, the NRC requires that changes to the plant technical specifications be proposed to address RHR interlock requirements.

In a report done by Electric Power Research Institute (Vine et al. 1983) it was concluded that

> "...Significant improvements in residual heat removal system operations can be achieved by improved shutdown plant administrative controls and a limited number of potentially cost effective plant modifications....".

The proposed resolutions discussed above will provide for better plant administrative controls and certain other beneficial plant modifications. The improvement factor in the system reliability (less frequent inadvertent valve closure) is estimated by assuming how much of an improvement each one of the proposed safety resolutions will make in reducing the possibility of inadvertent closure of RHR valve.

Proposed safety issue resolution part A, Review and Documentation of the Design Basis of RHR Valve, is believed not to have any impact on the resolution of this safety issue. This part of the proposed resolution is merely a review and documentation of the existing system and serves no purpose in resolving this issue. Therefore, an improvement factor of zero is assigned to this part of the proposed resolution. The same argument can be made about proposed safety issue resolution part D, Technical Specification Changes. This part of the proposed resolution is merely a documentation of the changes made to the system or the plant procedures and therefore has no impact on resolution of this issue. An improvement factor of zero is also assigned to this resolution part. The greatest contribution to the resolution of this safety issue is believed to come from proposed resolution part C, System Design Modifications and proposed resolution part B, Interim Operating Procedures. These two resolution parts address the two main causes of inadvertent closure of the RHR valve, human error and system malfunctions.

Proposed resolution part C, recommends the installation of a new logic system where two signals are required to close the RHR valve. This is referred to as a two-out-of-two logic. The present logic system operates on a one-out-of-one signal.

The total unavailability of the original system can be represented by Q_t. This is estimated to be .11/py. This estimate was derived by assuming that 22 out of the 25 documented events are independent (one valve closure) while the other 3 events are common cause failures (both valves). (Therefore .11/py = 22 independent events out of 206 plant years.)

The unavailability of each signal can be represented by Q_A. Accounting for two systems, will make the total unavailability of the system to be $2Q_A$.

The unavailability of the modified system (parallel configuration) can be presented as $Q_t=Q_A^2$. From above we have that $Q_A=Q_t/2$. Therefore, the unavailability of the modified system is $Q_t^2/4$. Substituting for Q_t, or .11 (22 independent events out of 206 plant years), in the above relationship will give us the new frequency of inadvertent closure of one valve. This value is then .003. Adding this to the .015/py frequency of inadvertent closure of both valves (common cause) will give us the adjusted-case initiation frequency of (.015/py + .003/py) or .018/py. This represents an improvement factor of approximately 85 percent in comparison to the original (base-case) initiation frequency.

Proposed resolution part B, Interim Operating Procedures, calls for appropriate instructions and training for plant operators and also interim operating procedures to preclude closure of the RHR valve. Implementation of this proposed resolution part will remind the operators of the possibility of the valve closure and its consequence, and providing interim operating procedures and appropriate training will reduce the possibility of human error. It is assumed that this proposed resolution will reduce the human error by 50 percent. This estimate is based on the best engineering judgment available.

Assuming that better operating procedures and operator training will make the operator 50 percent more effective will reduce the operator failure from probability of .04 to .02.

It is necessary to mention that other suggested system modifications such as installation of loss of RHR flow or pressure alarms are beneficial in getting the operators attention and in aiding the operator to discover the cause of the transient, i.e. RHR valve closure. Such modifications are beneficial in dealing with the transient once it has occurred. They do not have any impact on preventing inadvertent closure of the RHR valve and therefore their contribution was not included in this analysis.

Using the above estimates in the event tree shown in Figure 1, will give us the adjusted-case core-melt frequency.

Therefore, the adjusted-case core-melt frequency is estimated to be 3.6E-07/py.

The public risk reduction analysis was done by assuming that the containment failure modes, likelihood, and release categories are similar to T_1MLU in Oconee RSSMAP (PWR-3, 5, and 7). Therefore, the base-case public risk is estimated to be 13 man-rem/py and the adjusted-case public risk is estimated to be 1 man-rem/py. Therefore, the reduction in public risk per plant year is estimated to be about 12 man-rem.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   RCS/RHR Suction Line Interlock on PWRs (99)

2. Affected Plants (N):

   3D Operating PWRs

   28 Planned PWRs

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

   27.7 years for the operating plants

   30.0 years for the planned plants

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

   PWR: (19,900 man-rem) (4.44E-06/py) = 8.84E-02 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
147	957	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 discussions held with PNL staff. 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 = (2.5E-02 R/hr)(3 man-wk/plant)(40 man-hr/man-wk)(75% worker efficiency)
   = 2.25 man-rem/plant

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

   ND = (58 plant)(2.25 man-rem/plant) = 130 man-rem

TABLE 2. (cont'd.)

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):

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

3.0 SAFETY ISSUE COSTS

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

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   RCS/RHR Suction Line Interlock on PWRs (99)

2. Affected Plants (N):

   30 operating PWRs

   28 planned PWRs

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

   27.7 years for operating PWRs

   30.0 years for planned PWRs

Industry Costs (Steps 4 through 12)

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

   PWR: ($1.65E+09)(4.44E-06/py) = $7.33E+03/py

TABLE 3. (cont'd.)

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

Best Estimate	Upper Bound	Lower Bound
\$1.22E+07	\$7.94E+07	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 are based on PNL staff estimates.

A. Review and Documentation of the Design Basis of RHR Suction Valve Interlocks.

This task is estimated to require 4 man-wk. The cost associated with this is (4 man-wk)(\$2270/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 loss of RHR flow alarm and pressure alarm 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-week 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 replacement power cost is excluded from this analysis.

D. Technical Specification 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.

TABLE 3. (cont'd.)

Therefore, the total cost of implementing this safety issue per plant is as follows:

TASK	COST ($)
A	$ 9.08E+03
B	$ 1.14E+04
C	$ 1.76E+04
D	$ 9.08E+03
Total	$ 4.72E+04/plant.

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

NI = ($4.72E+04/plant)(58 plants) = $2.74E+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 Io = 0.

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$2.74E+06	$4.11E+06	$1.37E+06

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Cost is estimated directly in step 14.

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

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.

TABLE 3. (cont'd.)

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

It is estimated that a total of 6 man-wk/plant is required. This estimate is broken into 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 SIR implementation is:

(6 man-wk/plant)($2270/man-wk) = $13,620/plant.

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

NC = ($13620/plant)(58 plants) = $7.90E+05

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 (S_N):

Best Estimate	Upper Bound	Lower Bound
$8.1E+05	$1.2E+06	$4.1E+05

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: 105, Interfacing Systems LOCA at Boiling Water Reactors

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Existing evidence indicates that the isolation boundary between the reactor coolant system and the low-pressure injection systems failed at least three times per 200 BWR years of operation. Most of these failures or partial failures can be attributed to errors that occur during maintenance. The analysis of low-pressure-system overpressurization, public risk, occupational radiation exposure (ORE) and costs associated with resolution of this issue are the focus of this evaluation.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 2.2E+04

OCCUPATIONAL DOSES:

SIR Implementation =	16
SIR Operation/Maintenance =	510
Total of Above =	530
Accident Avoidance =	65

COST RESULTS ($1E+06)

INDUSTRY COSTS:

SIR Implementation =	0.22
SIR Operation/Maintenance =	1.2
Total of Above =	1.4
Accident Avoidance =	5.4

NRC COSTS:

SIR Development =	0.095
SIR Implementation Support =	0.23
SIR Operation/Maintenance Review =	0.58
Total of Above =	0.91

INTERFACING SYSTEMS LOCA AT BOILING WATER REACTORS

ISSUE 105

1.0 SAFETY ISSUE DESCRIPTION

Recent BWR operating experience indicates that the isolation boundary (including related tests and maintenance requirements) between the reactor coolant system and low-pressure interfacing systems may not adequately protect against overpressurization of the interfacing low-pressure systems. Evidence exists which indicates at least three failures and five potential failures of the boundary between the reactor coolant system and low-pressure injection systems in approximately 200 BWR-years of operation. The three failure experiences described below have been utilized in an effort to formulate an approach for determining the significance of the risk associated with these failures.

1. The most recent failure event occurred at Brown's Ferry 1 in 1984. "While performing a semi-annual logic function test at Brown's Ferry Unit 1, the operators failed to electrically disarm the motor operated injection valve in the Loop 1 core spray system, and it opened upon receiving the actuation signal. By itself, this error would not have been a problem. However, the testable check valve, which is in series with the motor operated valve, was open due to previous maintenance errors. In December 1983 or earlier, during maintenance on the pilot solenoid valve for the testable check valve, a plunger with reversed air parts was apparently installed in the solenoid valve. This resulted in the check valve being held open until the event on August 14, 1984. Additional erroneous adjustments of valve position indicators gave improper indications in the control room. When the motor operated injection valve opened, the core spray system was subjected to reactor coolant system pressure and temperature causing the relief valve on the pump discharge to open. The maximum core spray system pressure could not be determined. The event was terminated by the operator who closed the motor operated valve that had been erroneously opened. The motor operated valve was open for about 13 minutes (Newberry 1984)." This event, in itself, did not affect operation of the plant; however, the plant was shut down so questions about the check valve and system could be resolved. Following analysis and inspection of the core spray system and repair of the check valve, the plant was restarted.

2. The next most recent failure event took place at Pilgrim in 1983. "This event involved an actual overpressurization of the low-pressure pump suction piping of the high pressure coolant injection (HPCI) system during a functional test of the HPCI system logic. The cause of this event was also traced to personnel errors. These errors included conducting more than one surveillance test at the same time and not insuring that the test prerequisites and initial test conditions for all steps in the test procedure were being met. The personnel errors led to the simultaneous opening of the two HPCI pump discharge valves. With both valves open, a downstream isolation check valve, which was stuck in a partially open position, permitted a sudden pressurization of the low-pressure HPCI pump suction piping (AEOD 1984)."

3. The earliest failure event occurred at Vermont Yankee in 1975. "At full power with both the LPCI/RHRS testable isolation check valve and a motor operated isolation valve unknowingly partly open when they were believed fully closed, operators cycled open a second motor operated isolation valve during a routine LPCI surveillance test. The LPCI line overpressurized and steam and water were discharged from the three relief valves on the line. The RHRS heat exchanger flange gasket also began leaking. (The gasket is only rated for the system design pressure even though the piping itself was probably over designed.) The reason for the check valve failure was not stated. The motor operated valve failure was due to an erroneous position indication after the valve was cycled in a test.[a].

In all of the cases described above, there was a degradation of the high-pressure/low-pressure system isolation valves due to personnel errors. An inadvertently opened valve significantly increases the likelihood of an interfacing loss-of-coolant accident (LOCA). Multiple valve failures, including those caused by operator error, cause a loss of both barriers between the reactor coolant system and the low-pressure systems which will result in overpressurization of the low-pressure system. Subsequent accident progress is dependent on several things which might include: the availability of remaining LPCI or LPCS trains and the success criteria for reflooding the core with only two LPCI pumps and the core spray system (as an example); the role and adequacy of other means of coolant makeup; the rate of depletion of suppression pool inventory; and the extent of adverse environmental impact on vital equipment in the reactor building (Newberry 1984). These considerations are beyond the scope of this analysis. However, these and additional concerns should be investigated to achieve a complete and realistic risk evaluation.

PROPOSED ISSUE RESOLUTION

No specific requirements have been proposed at this time, however, consideration of the following potential requirements has been suggested (Holahan, 1984):

1. Adequacy of valve testing (frequency, type and order of tests) and the consideration of conducting logic tests only when the plant is shutdown and at low pressure.

2. Safety classification of testable check valve actuator/controls (upgrading may require better maintenance practices).

3. Design adequacy (possible modifications including the removal or deactivation of the testable check valve actuator).

4. Isolation capability of motor-operated valves given a LOCA.

(a) Harris, J. D. (ORNL) and J. W. Minarick (SAIC). 1985. An Evaluation of BWR Over-Pressure Incidents in Low Pressure Systems. Preliminary Draft. Prepared for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Research.

5. Reevaluation of Event V criteria to see if they should also pertain to power-operated valves, or if it is important for testable check valves to have their air operators deactivated.

6. Accelerating ASME Section XI Inservice Testing reviews for interfacing LOCAs.

7. Reliability of valve position indicators.

For this evaluation it is assumed that the resolution includes minimal hardware changes (e.g., fool-proof fittings, color coding and labeling to eliminate problems of crossing air actuation lines) and procedural changes (e.g., testing procedures, order of tests, safety classification of equipment).

AFFECTED PLANTS

Operating BWRs which have RCS/RHR system interface configurations similar to Hatch Unit 2 have been identified and include: Duane Arnold, Brunswick 1 and 2, Cooper, Dresden 2 and 3, Hatch 1, Fitzpatrick, Monticello, Peach Bottom 2 and 3, Pilgrim, and Quad Cities 1 and 2 (AEOD 1984). This list does not include plants similar in design to Browns Ferry, a plant which also experienced a similar isolation boundary problem. Therefore, the list of affected plants utilized in this analysis will include additional BWR 3 and 4 class operating plants (i.e., Millstone, Browns Ferry 1, 2 and 3, Vermont Yankee). Therefore, the total number of potentially affected operating BWRs considered in this analysis is 20. It is assumed that corrections will be incorporated in all forward fit BWRs.

2.0 SAFETY ISSUE RISK AND DOSE

Public risk reduction estimates used in this analysis are based on the Interim Reliability Evaluation Program (IREP): Analysis of the Browns Ferry, Unit 1, Nuclear Plant (NRC 1982). Attachment 1 includes the assumptions used in deriving failure rate data such as that associated with a LOCA resulting from a system overpressurization. In addition, assumptions made in utilizing the available LOCA sequences within the Browns Ferry IREP are included. The general approach was to use available historical data for the failure of the isolation boundary in conjunction with estimates made for piping failure resulting from an overpressurization. This approach is different from that previously used in risk analyses. For instance, in other studies, large-break LOCA in the RHR injection piping has been estimated by considering the frequency for a large pipe rupture (1E-04/py), the frequency of a check valve in severe internal leak mode (3E-07/hour), and the test frequency for check valves (1 test every 3 months). The resultant failure frequency for a large break LOCA outside containment is thus, (1E-04)(3E-04)= 3E-08/py. Because this failure frequency is insignificant when compared to large break LOCAs inside containment, further analysis was not considered necessary according to NUREG/CR-2802 (NRC 1982).

In this evaluation an attempt was made to redefine the initiating event based on historical data available and to adapt the available LOCA functional event trees based on breaks inside containment. The results of this analysis are presented in Table 1.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Interfacing Systems LOCA at Boiling Water Reactors (105)

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

This issue includes 20 operating BWRs which were assumed to have similar RCS low-pressure system interfaces. It was also assumed that none of these plants have modified their isolation boundaries or have altered their maintenance programs in response to previous problems associated with failure of the isolation boundaries.

The average remaining life for the 20 plants considered is 24.8 years.

3. Plants Selected for Analysis:

Browns Ferry Unit 1 - Representative BWR (NRC 1982)

4. Parameters Affected by SIR:

The affected parameter for this issue is L_S, the frequency of a LOCA due to defeat of the isolation boundary and subsequent overpressurization of the low-pressure system. Only the frequency of the inboard and outboard valve failures is considered to be affected when establishing the LOCA frequency used as the initiating event for the associated dominant accident sequences. The main element affecting this frequency is operator error associated with proper positioning and maintenance of the two valves in question.

5. Base-Case Values for Affected Parameters:

The initiating event L_s is taken as the product of the valve failure frequency times the probability of LOCA in a situation where the valves have failed in an open position and the low-pressure system is overpressurized. Operator error is considered to be embedded in the frequency of inboard and outboard valve failures and is reflected in the historical data available. Three boundary failures per 200 BWR operating years will be taken as the base-case frequency for the valve failures. The probability of a LOCA following defeat of the isolation boundary is established in Attachment 1 and is given here as 1E-01. Therefore, L_s is (3/200py)(1E-01) = 1.5E-03/py.

TABLE 1. (cont'd.)

6. Affected Accident Sequences and Base-Case Frequencies:

Affected accident sequences were identified from candidate dominant accident sequences for Browns Ferry Nuclear Plant (US NRC 1982). LOCA sequences for a large liquid break are given for the suction-side of recirculation pumps. It should be noted again that these sequences are for a break inside of containment modified to include break isolation.

The dominant accident sequences and base-case frequencies are:

$L_SIR_BR_A$ (1.5E-03)(0.1)(1)(3.1E-02) = 4.65E-06/py (1)

$L_SIF_AG_B$ (1.5E-03)(0.1)(5.2E-02)((0.5)(2.1E-02)) = 8.19E-08/py (2)

$L_SIG_AG_B$ (1.5E-03)(0.1)(1)((0.5)(2.1E-02)) = 1.58E-06/py (3)

This assumes the unavailability of torus cooling in the first sequence and one train of LPCI in Sequence 2 and 3. These variations were included to show the difference in decay heat removal in the event of a break outside containment. If the break occurs outside containment, the coolant emitted from the break does not enter the torus as it does when the break is inside containment since there is no closed loop to return the coolant to the core from the torus. In the second and third sequences, it is considered that one loop of LPCI is lost without regard to the number of pumps in that loop. Thus, the unavailability value for one loop is 1 and for two loops it is reduced by one half.

7. Affected Release Categories and Base-Case Frequencies

The base-case, affected core melt frequency is estimated directly in Step 8 which follows.

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

The affected total sequence frequencies from information provided in Step 6 is:

$\bar{F}$ = 4.65E-06 + 8.19E-08 + 1.58E-06 = 6.31E-06/py

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

Using a dose factor associated with BWR release category 2 (to represent a leak outside containment) from Table D.1 of the Guidelines for Nuclear Power Plant Safety Issue Prioritization (Andrews, et al. 1983) the affected public risk is:

W = (6.31E-06/py)(7.1E+06 man-rem) = 44.8 man-rem/py

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

It is assumed that operator error associated with the failure of the inboard and outboard valves on a low-pressure system is a target for improvement. It is also assumed that the frequency of failure is adjusted to the values which are now considered to exist, that is the products of the values associated with failure frequency of the check valve (3E-07/hr (NRC 1982)) and failure (inadvertent opening) of the motor operated valve due to spurious signals (1E-08/hr) plus the rate of disk rupture (1E-07/hr)(AEOD 1984). The adjusted case value for failure of the two valves in series is:

((3E-07/hr)(1000hr)) ((1.1E-07/hr)(8760hr/py)) = 2.9E-07/py.

This includes a reduction in operator error from what was presented in the historical data. This estimate is assuming an "adequate" inspection every three months as required which translates to a mean dead time of 1000hr.

The above calculations result in an initiating event L_S of 2.9E-08/py where the probability of a LOCA given overpressurization remains the same (1E-01).

11. Affected Accident Sequences and Adjusted-Case Values:

The adjusted-case values for LOCA sequences identified above are:

$L_SIR_BR_A$ (2.9E-08)(0.1)(1)(3.1E-02) = 8.99E-11/py

$L_SIF_AG_B$ (2.9E-08)(0.1)(5.2E-02)((0.5)(2.1E-02)) = 1.58E-12/py

$L_SIG_AG_B$ (2.9E-08)(0.1)(1)((0.5)(2.1E-02)) = 3.05E-11/py

12. Affected Release Categories and Adjusted-Case Frequencies:

The adjusted-case, affected core-melt frequency is estimated directly in Step 13.

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

$\bar{F}$* = 8.99E-11/py + 1.58E-12/py + 3.05E-11 = 1.22E-10/py

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

W* = (1.22E-10/py) (7.1E+06 man-rem) = 8.66E-04 man-rem/py

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

$\Delta \bar{F}$ = 6.31E-06/py

TABLE 1. (cont'd.)

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

ΔW = 44.8 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
2.2E+04	6.7E+05	0

ATTACHMENT 1 (To Table 1)

To date there have been no LOCAs as a result of ruptured low-pressure piping associated with loss of the isolation boundary between the reactor coolant system and a low-pressure system such as LPCI or LPCS. The historical data provides information about three instances where failure of this boundary has occurred and in each instance, the isolation valves that were open have eventually been closed. Concern arises when considering an overpressurization which would result in a LOCA in the low-pressure system. In that case it must be considered that the break may not be isolated in which case reactor coolant would be lost outside of containment.

For LOCA functional event trees where breaks occur outside of containment, there are two basic functions available for mitigating the LOCA once it is determined that the break cannot be isolated. These functions include successful reactor shutdown and core cooling. Containment overpressure protection is not required since all heat, noncondensable gases and radioactivity will be transmitted outside of containment by the break. Core cooling is still needed during the injection and long-term decay heat removal phases. These phases along with the initiating event and break isolation are shown in the event tree in Figure A.1 (NRC 1982).

The Browns Ferry IREP concludes that the failure frequency for a large break LOCA in the RHR injection piping outside the primary containment is approximately 3E-08/py which is insignificant compared to a large discharge break inside containment and thus, the analysis of this accident sequence is eliminated. Because there is historical evidence which suggests that the failure frequency of the two isolation valves in series is orders of magnitude higher than that considered in the Browns Ferry IREP, this analysis will be based on historical evidence as well as some estimate of the probability of LOCA given an overpressurization of the low-pressure system and the probability of break isolation.

The following steps provide the information necessary to determine the initiating event, in this case a large break LOCA, L_S. The frequency of inboard and outboard valves failing is given as 3/200 BWR years. The probability of a LOCA following defeat of these two valves is developed below.

ESTIMATE OF THE PROBABILITY OF LOCA GIVEN OVERPRESSURIZATION

It is assumed for this analysis that if a crack exists in the low-pressure piping of concern and is significant enough to cause failure if exposed to reactor coolant system pressure (1100 psi), then a LOCA will occur. It is also assumed that an overpressure failure will occur so rapidly that neither operator action of closing the motor operated isolation valve, nor opening of the downstream, low-pressure-system, relief-valve will prevent the piping failure.

For this analysis it was assumed that piping was 304 stainless steel, had an OD range from 18 inches to 24 inches, had a yield strength of approximately 25 ksi and was either Schedule 40 or Schedule 80. A hoop stress was calculated to determine if an overpressure of 1100 psi (reactor coolant system pressure) could result in a hoop stress near the yield value. Under the assumptions given, the worst case was under 20 ksi (i.e., below yield stress).

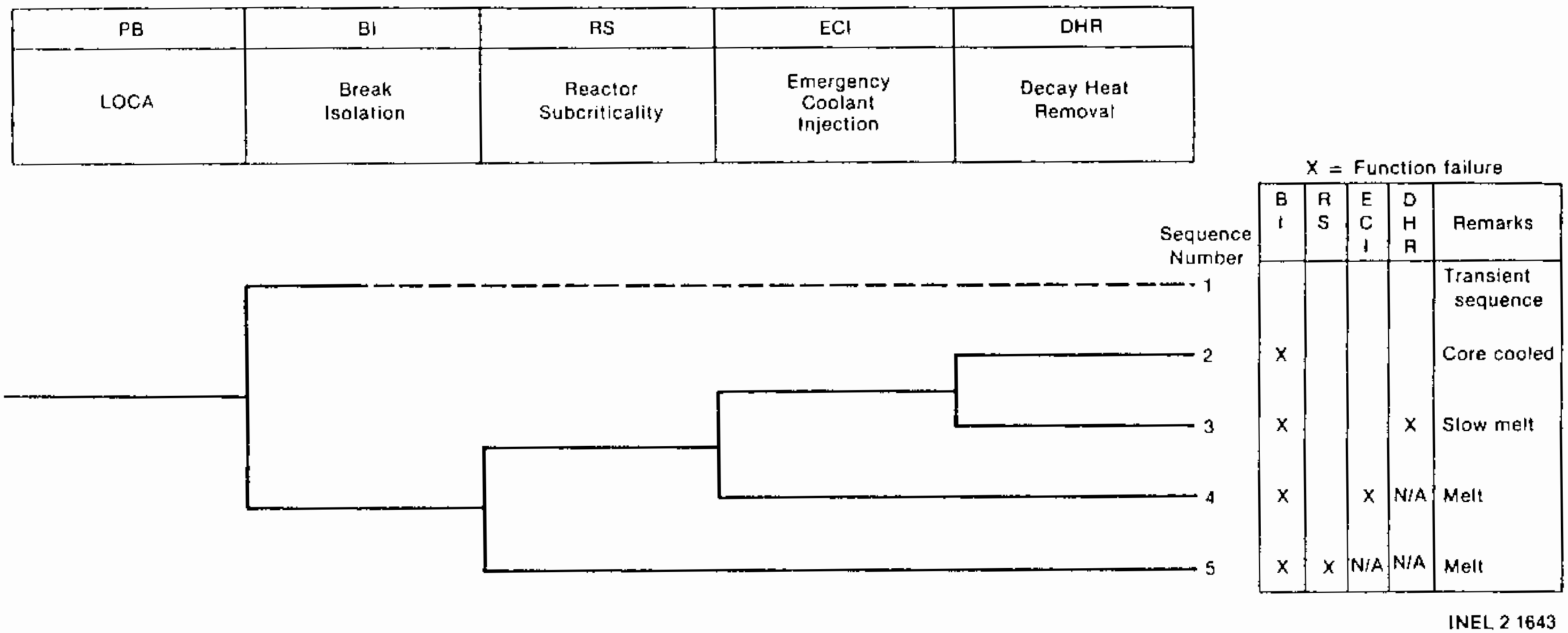


FIGURE A.1. LOCA functional event tree-break outside containment

It was assumed that the low-pressure piping could be subject to cracking mechanisms which would decrease the effective pipe thickness making it more susceptible to failure given the same overpressure. In developing this scenario we investigated intergranular stress corrosion cracking experience in BWR piping welds. Figure A.2 presents information on cumulative failure distributions for weld cracks versus days since startup. The failure distribution presents cracks that are significant enough to cause future failures. In estimating the days since startup for the 20 plants affected by this issue, we have a mean value of approximately 10 years or 3600 days with a range for individual plants of 2200 to 4700 days. This produces the probability of having an average of 6 significant cracks per 1000 welds. If one assumes no more than 50 welds per line, there is a probability of 3.0E-01 significant cracks per line. It is assumed that these include all cracks with a depth greater than or equal to 10 percent of the wall thickness. In our postulated case we need a crack greater than or equal to 60 percent of the wall thickness for the effective thickness to be reduced to the point that the stresses would exceed a stress between yield and ultimate (criteria typically considered for fracture). Therefore, not all of the significant cracks will be large enough to produce fracture given the overpressure. In this analysis we will consider that 30 percent of these cracks are large enough to produce a fracture. That is, we assume that 70 percent of the cracks are between 10 and 60 percent throughwall and 30 percent are cracks which might exceed 60 percent of the wall thickness. This is very conservative when assuming there is a 90 percent probability of detecting a flaw with a depth of 10 percent of the wall (Simonen 1984). Thus, the probability of a LOCA given the overpressure is now on the order of 1E-01. It is suggested that this probability needs to be addressed at a greater level of detail in an extended analysis.

PROBABILITY OF A LOCA

The probability of a LOCA is now the frequency of failure of the two isolation valves times the probability of a LOCA in the event of this failure, or 1.5E-03/py. It should be noted that the potential for the system failing due to dynamic loading is not considered. Although the probability of this type of failure is low in the RHR keepfill system, not all of the low-pressure systems are designed with this feature.

USE OF BROWNS FERRY IREP EVENT TREES

Because no estimates are given for events leading to large LOCA outside of containment, the large LOCA systematic event tree with a break inside containment is used and modified. The IREP event tree incorporates an element called short term containment integrity which is not necessary for this analysis since all heat, noncondensable gases and radioactivity will be transmitted outside of containment by the break. The IREP event tree (break inside containment) logic is modified to represent a break outside containment by assuming a probability of break isolation. The probability of break isolation is assumed to be dominated by operator failure to isolate. It is assumed that this probability is 0.1 given that the LOCA is outside containment and

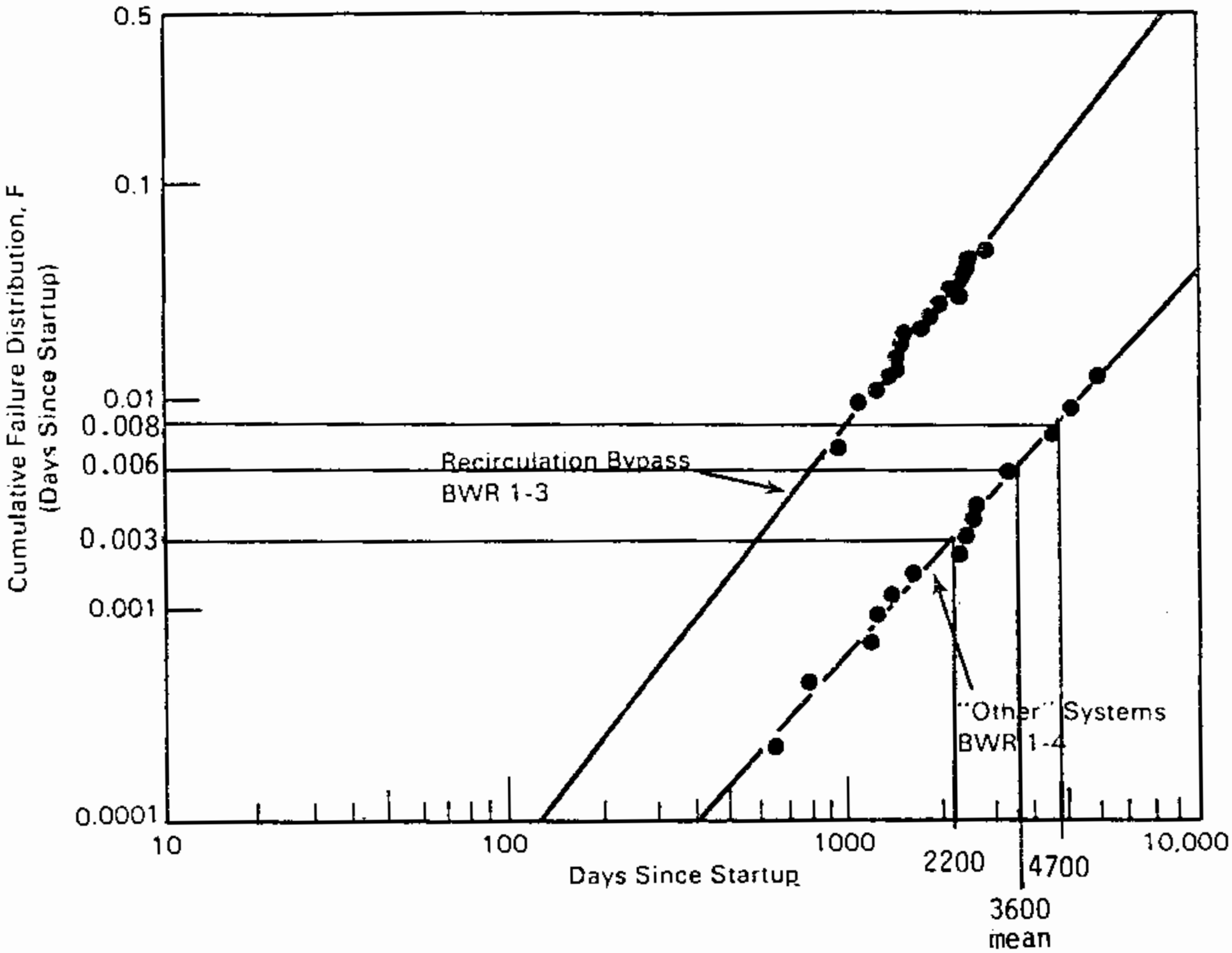
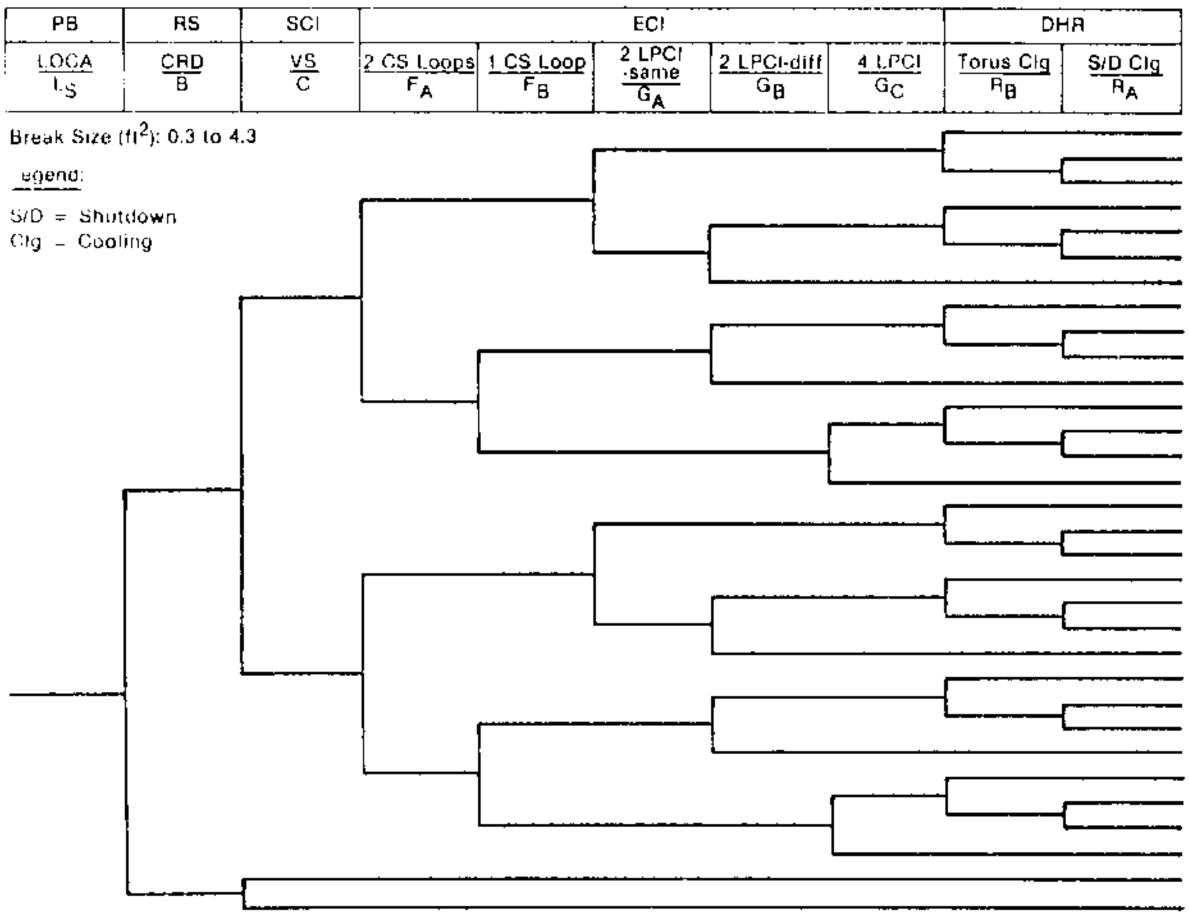


FIGURE A.2. Intergranular Stress Corrosion Cracking Experience in BWR Piping (Eason 1982) considering a high stress situation. This probability of failure to isolate is designated as parameter "I" in the following sequences. The initiating event L_S is changed and reflects the calculations performed above. The dominant sequences affected are included below and are illustrated in Figure A.3 (without the break isolation modification) (NRC 1982):

$L_S I R_B R_A$
$L_S I F_A G_B$
$L_S I G_A G_B$

The frequencies for these sequences are developed in Table 1.



X = Function failure

Sequence Designator	RS	SCI	ECI	DHR	Remarks
					Core cooled
					Core cooled
$L_S R_B R_A$				X	Slow melt
					Core cooled
					Core cooled
$L_S G_A R_B R_A$				X	Slow melt
$L_S G_A G_B$			X	N/A	Melt
					Core cooled
					Core cooled
$L_S F_A R_B R_A$				X	Slow melt
$L_S F_A G_B$			X	N/A	Melt
					Core cooled
					Core cooled
$L_S F_A F_B R_B R_A$				X	Slow melt
$L_S F_A F_B G_C$			X	N/A	Melt
		X			Core cooled
		X			Core cooled
$L_S C R_B R_A$		X		X	Slow melt
		X			Core cooled
		X			Core cooled
$L_S C G_A R_B R_A$		X		X	Slow melt
$L_S C G_A G_B$		X	X	N/A	Melt
		X			Core cooled
		X			Core cooled
$L_S C F_A R_B R_A$		X		X	Slow melt
$L_S C F_A G_B$		X	X	N/A	Melt
		X			Core cooled
		X			Core cooled
$L_S C F_A F_B R_B R_A$		X		X	Slow melt
$L_S C F_A F_B G_C$		X	X	N/A	Melt
$L_S B$	X		N/A	N/A	Melt
$L_S B_C$	X	X	N/A	N/A	Melt

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FIGURE A.3. LOCA systemic event tree for large liquid break, suction-side of recirculation pumps (L_S)

TABLE 2. Occupational Dose Work Sheet

It is anticipated that additional occupational radiation exposure will result if plants fulfill current requirements, instigate a more rigorous or altered inspection program (e.g., follow outlined test and post maintenance procedures, conduct one surveillance test at a time, perform leak tests after the flow or operability test), and make minor hardware changes (e.g., air supply lines physically coded to preclude interchanging lines as is the case with different threads or different diameter connectors, color coding supply lines, labeling). At this time major system hardware changes are not anticipated. The consideration of an alternative test method for the testable check valve which would include passing shutdown cooling flow through the valve during each cold shutdown, and thereby allow for deactivation of the air-actuator, is a potential change which is not considered here. Consideration will be given to this potential fix when it is obvious that shutdown cooling flow will allow for full-stroking of the valve.

1. Title and Identification Number of Safety Issue:

   Interfacing Systems LOCA at Boiling Water Reactors, 105

2. Affected Plants (N):

   20 operating BWRs

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

   The average remaining life for each of the 20 BWRs is 25.7 years.

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

   $\Delta(\bar{F}D_R)$ = (6.31E-06/py)(19,900 man-rem) = 1.26E-01 man-rem/py

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
65	390	0

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

   Additional utility labor in radiation zones is anticipated for the installation of minor hardware changes (e.g., color coding, altering connectors on air supply lines, etc.). Labor hours required for 2 men working 16 man-hours each is 32 man-hour/plant.

TABLE 2. (cont'd.)

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

Consideration is given to the fact that the testable check valve is located inside containment and the MOV is located just outside containment. An average dose rate for work performed is assumed to be 25 mR/hr.

D = (32 man-hr/plant)(.025 R/hr) = 0.80 man-rem/plant

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

ND = (0.80 man-rem/plant)(20 plants) = 16 man-rem

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

Evidence suggests that some plants are not performing tests or post-maintenance procedures accurately. Under the recommended procedures these plants would most likely require additional labor time in radiation zones to complete currently established requirements. Other plants may be completing tests and post-maintenance procedures as outlined. At this latter group of plants minimal additional time in radiation zones will be required. Any changes in procedures (e.g., performing leak tests after operability tests or upgrading the classification of the air actuator and pilot solenoid on the testable check valve to safety-related) would require additional inspection time. It is assumed that on the average an additional 40 man-hr/py will be required. This assumes a three month test interval as outlined in ASME Section XI, IWV-3000 for both the testable check valve and MOV and two maintenance staff utilized in testing and post maintenance procedures.

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

An average dose rate for work performed is again assumed to be 25 mR/hr.

D = (40 man-hr/py)(0.025 R/hr) = 1.0 man-rem/py

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

NTD_o = (20 plants)(25.7 years)(1.0 man-rem/py) = 514 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
530	1600	180

3.0 SAFETY ISSUE COSTS

Results of industry and NRC cost calculations are included in this section with best estimates given for labor and equipment costs. The results of this analysis are included in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Interfacing Systems LOCA at Boiling Water Reactors (105)

2. Affected Plants (N):

   20 operating BWRs

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

	$\bar{T}$ (yr)
BWRs: operating	25.7

Industry Costs (Steps 4 through 12):

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

   $(\bar{F}A) = (6.31E\text{-}06/py)(\$1.65E{+}09) = \$1.04E{+}04/py$

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

Best Estimate	Error Bounds Upper	Lower
$5.4E+06	3.2E+07	0

6. Per-Plant Industry Resources for SIR Implementation:

   It is anticipated that 3 man-wks/plant will be required to assess current testing and post-maintenance procedures relative to any requirements or NRC staff recommendations which develop as a result of this issue. This might also include a review of maintenance records, development of recommendations or a training review program for maintenance and supervisory staff. Costs for materials to install fool-proof features (e.g., connectors) are estimated at $2500/plant.

   Additional utility labor is anticipated for the installation of minor hardware changes (e.g., color coding, altering connectors on air supply lines, etc.). Labor hours required for 2 men working 16 man-hrs each is 32 man-hrs/plant.

TABLE 3. (cont'd.)

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

$$I = (3 \text{ man-wk/plant})(\$2270/\text{man-wk}) + \$2500 + (32\text{man-hr/plant})(\text{man-wk}/40 \text{ man-hr})(\$2270/\text{man-wk}) = \$1.11\text{E+04/plant}$$

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

$$NI = (20 \text{ plants})(\$1.11\text{E+04/plant}) = \$2.22\text{E+05}$$

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

Additional labor beyond present maintenance and testing is estimated to be 40 man-hr/py (average over affected plants- see Table 2, Step 9).

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

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

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

$$N\bar{T}I_o = (20 \text{ plants})(25.7 \text{ yrs})(\$2270/\text{py}) = \$1.17\text{E+06}$$

12. Total Industry Cost (S_I):

Best Estimate	Upper	Lower
\$1.4E+06	\$2.0E+06	\$8.0E+05

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

Resources for SIR development will require a review of input from plants and a determination of future action. The time required for this effort is estimated to be 20 man-wks. Because there are many unanswered questions with regard to historical data, plants affected, probability of valve failure, probability of a LOCA given defeat of the isolation barrier and the probability of core-melt given a LOCA outside containment, it is anticipated that technical support will be needed in the development of this issue. It is anticipated that a minimal effort required to complete _just_ the probabilistic risk analysis would be approximately \$50K.

TABLE 3. (cont'd.)

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

C_D = (20 man-wk)($2270/man-wk) + ($50E+03) = $9.54E+04

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

It is anticipated that a review of each affected plant will be required to assure a plant-specific application of any requirements. The estimated labor for these reviews is 5 man-wks/plant.

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

C = (5 man-wks/plant)($2270/man-wk) = $1.14E+04/plant

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

NC = (20 plants)($1.14E+04/plant) = $2.28E+05

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

This issue involves a history of oversights related to testing and maintenance procedures that are currently in existence. If the SIR involves improving or altering procedures, it is anticipated that NRC staff will monitor results of incorporating procedural changes at affected plants. This should require an additional 0.5 man-wk/py averaged over the lifetime of the affected plants.

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

C_o = (0.5 man-wk/py)($2270/man-wk) = $1135/py

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

NTC_o = (20 plants)(25.7 yrs/plant)($1135/py) = $5.83E+05

21. Total NRC Cost (S_N):

Best Estimate	Error Bounds Upper	Error Bounds Lower
$9.1E+05	$1.2E+06	$5.9E+05

REFERENCES (For Issue 105)

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: B-22, LWR Fuel (Pellet/Cladding Interaction)

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

Plants experience fuel rod failures during normal operation, and rod failures are expected during accident sequences. There failures can result in dose releases which, in turn, can dictate plant shutdowns or reduced power operation to meet acceptable release rates. Improvements in fuel performance are anticipated by completing minor improvements in fuel designs and by altering planned power maneuvering. However, this requires fuel behavior predictability under various operating levels.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	0

OCCUPATIONAL DOSES:

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

COST RESULTS ($$10^6$)

INDUSTRY COSIS:

SIR Implementation =	1.7
SIR Operation/Maintenance =	-190
Total of Above =	-190
Accident Avoidance =	0

NRC COSTS:

SIR Development =	0.20
SIR Implementation Support =	0.32
SIR Operation/Maintenance Review =	0.95
Total of Above =	1.5

LWR FUEL (PELLET/CLADDING INTERACTION)

ISSUE B-22

1.0 SAFETY ISSUE DESCRIPTION

Individual reactor fuel rods sometimes fail during normal operation, and many rods are expected to fail during severe accidents, releasing activity to the surroundings and providing a source of releases from the plant. Failures during some accidents could be severe enough to fragment the cladding and disperse fuel pellets into the coolant, but regulations require that the coolable rod-like geometry must be maintained. Behavior characteristics such as rod bowing and densification also have a strong effect on plant-limiting conditions. Thus, fuel behavior during normal operation and postulated accidents must be predictable in order to set operating limits, to limit activity releases and to insure no more than an acceptable degradation of the fuel system. The object of the LWR fuels task is to assure that such predictions are reliable. (NRC 1978)

Several problems were identified in an effort to improve the predict ability of fuel performance. The 11 subtasks identified were categorized into three major tasks as follows:

1. Evaluation of Design Criteria
   a. Pellet/Cladding Interaction (PCI)
   b. Fuel Behavior During Design Basis Accidents
   c. Fuel Cladding Design Limits

2. Evaluation of Performance Predictions
   a. Fuel Rod Bowing
   b. Fuel Rod Performance Codes
   c. ECCS Materials Behavior
   d. Radioactive Fission Gas Release
   e. Fission Gas Release
   f. Behavior of Water-Logged Fuel

3. Evaluation of Fuel Operating Experience
   a. Surveillance of New Fuel Assembly Design
   b. Fuel Rod Failure Detection

Ten of the above 11 subtasks have been resolved, subsumed by other programs, or dropped. The remaining issue is Pellet/Cladding Interaction (PCI), and it has been been reclassified as a safety issue because of its relationship to the adequacy of fuel design criteria.[a]

(a) R. J. Mattson. "Comments on Prioritization of Licensing Improvement Issues." Feb. 2, 1983, Memorandum (Enclosure 2) to J. L. Funches, U.S. Nuclear Regulatory Commission, Washington, D.C.

The resolution of this issue involves the development of methods to reduce fuel failures and limit activity releases. This is possible if fuel behavior during normal operation and under postulated accidents is predictable and if operating limits are set in accordance with these predictions.

In an effort to control fuel failures, General Electric has made some design changes in fuel assemblies. However, it is felt that most of the efforts to resolve this issue will come in an attempt to predict or alter power excursions and reduce dose releases by tightening up on such things as steam generator leakages. Although control of power excursions may not be as feasible under accident conditions, such control is possible during planned power maneuvering. Potential failure of fuel assemblies in light water reactors may be limited in these situations.

2.0 SAFETY ISSUE RISK AND DOSE

In the situation where fuel failures exist, plants must maintain off-gas release rates within allowable limits. If the plant exceeds these limits or fails to clean up high activity in the coolant, a shutdown is required. In less severe cases, the plant will often stay within the allowable release limits by maintaining reduced levels of power until the next refueling outage. Because of the allowable release limits imposed on plant operation, the reduction in public risk resulting from this safety issue resolution (SIR) is expected to be negligible. In addition, the number of refueling outages is not expected to change significantly due to resolution of this issue. Therefore, it is also anticipated that the occupational dose changes will be negligible. Based on these considerations, the Public Risk Reduction Work Sheet and the Occupational Dose Work Sheet have been omitted from this issue analysis.

3.0 SAFETY ISSUE COSTS

When fuel failures are severe enough to require operation at reduced power levels for extended periods of time (e.g., until next scheduled refueling outage) or require shutdown to replace failed assemblies, monetary losses due to downtime or power reduction become a concern. Estimated values for these losses are included in the following table.

TABLE 1. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   LWR Fuel: Pellet/Cladding Interaction (B-22)

TABLE 1. (cont'd.)

2. Affected Plants (N):

All operating and planned PWRs and BWRs are affected by resolution of this safety issue.

		N
PWR:	Operating	47
	Planned	43
	Total	90
BWR:	Operating	24
	Planned	20
	Total	44
All		134

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

		$\bar{T}$ (yr)
PWR:	Operating	27.7
	Planned	30.0
	Total	28.8
BWR:	Operating	25.2
	Planned	30.0
	Total	27.4
All		28.3

Industry Costs (Steps 4 through 12)

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

No change in core-melt frequency is foreseen for this SIR. Thus, $\Delta H=0$.

6. Per-Plant Industry Resources for SIR Implementation:

The assumption for resolution of this issue is that there will be some minor fuel assembly design changes, and there will be alterations in power

TABLE 1. (cont'd.)

6. Per-Plant Industry Resources for SIR Implementation (cont'd.):

excursions during planned power maneuvering. Implementation of these changes will only require training of appropriate staff as outlined below.

Training personnel for minor fuel assembly design changes will require a minimal amount of staff labor. It is estimated that 2 man-weeks will be required to train all personnel handling fuel assemblies or supervising refueling techniques.

Changes in operations reflected by alterations in power excursions will require that all reactor operators and supervisors receive appropriate training. In this issue it is assumed that 5 shifts/plant are available and that one of these is a training shift. It will be assumed that the following personnel will receive 10 man-hours of initial training, followed by additional training on their respective training shifts.

Shift Supervisor (1/shift)
Unit Supervisor (1/shift)
Assistant Shift Supervisor (1/shift)
Reactor Operators (2/shift)
Shift Technical Advisor (1/shift)
Reactor Engineer (1)
Day Shift Supervisor (1)
Trainees (3)

This example yields a total of 35 plant personnel needing the initial training procedures. Assuming a 10-hour initial training program, a total of 350 man-hours or 8.75 man-weeks is required. Therefore, the total man-hour requirement for implementing the SIR is 10.75 man-weeks.

It is assumed that the training described above is only required for those plants currently in operation. Training for personnel in plants under construction and in the planning stage will be absorbed by initial reactor operations training.

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

I = (10.75 man-wk/plant)($2270/man-wk) = $2.44E+04/plant

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

NI = ($2.44E+04/plant)(71 plants) = $1.73E+06

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

No additional labor for operation/maintenance of the SIR is anticipated. However, a reduction in plant down-time is expected to result. This is discussed in the next step.

TABLE 1. (cont'd.)

10.-11. Steps Related to Industry Cost for SIR Operation and Maintenance:

When fuel failures occur, plants attempt to remain in operation until a scheduled refueling outage. In most cases, this is possible. However, this procedure often means that a reduced level of power must be maintained in order to remain within allowable release limits. In this issue, it is assumed that three plants/year exceed allowable release limits due to fuel failures and that conditions require completion of a Licensee Event Report (LER). It is also assumed that an additional three plants/year approach these allowable release limits due to fuel failures but that power maneuvering prevents exceeding allowable limits. It is difficult to predict whether these plants will shut down, remain operating at reduced levels for long periods of time, or remain operating at reduced levels for relatively short periods of time. In this issue, it is assumed that all of the plants will remain on line at the reduced level of 50% power for thirty days. This is assumed to be an average power level and an average period of time until a scheduled outage under these conditions.

The major potential expense under the above conditions is that of replacement power due to reduced levels of output. The value assumed for the purchase of replacement power during each outage day is $300,000 (Andrews et al. 1983). Therefore, under conditions of 50% output, it is assumed that the purchase of replacement power during each day is $150,000.

The cost per plant for replacement power at those plants operating at a 50% power level due to fuel failures is as follows:

($150,000/day)(30 days/plant) = $4.50E+06/plant

Although it is assumed that the cost per plant remains the same for plants experiencing fuel failures under resolved conditions, the number of plants experiencing these failures will be reduced. Upon resolution of this issue, it is assumed that the number of plants reporting fuel failures and reducing power output as a direct result of these failures will be reduced by 25%. Defective fuel and failures attributable to accident conditions prevent higher predictions for the reduction in fuel failures. Therefore, the replacement power cost savings due to the resolution of this issue is as follows:

(-0.25)($4.50E+06/plant = -$1.13E+06/plant (Negative sign indicates cost savings.)

This applies to 6 plants/yr. Therefore, the total industry cost for SIR operation and maintenance (savings due to decrease in down-time) is as follows:

TABLE 1. (cont'd.)

10.-11. Steps Related to Industry Cost for SIR Operation and Maintenance (contd):

$\bar{NTI}_0$ = (6 plants/yr)(28.3 yr) (-\$1.13E+06/py) = -\$1.91E+08

It is assumed that the average remaining life of all plants (28.3 yr) applies to the 6 plants/yr experiencing decreased downtime.

12. Total Industry Cost (SI):

Best Estimate	Upper Bound	Lower Bound
-\$1.9E+08	-\$9.4E+07	-\$2.8E+08

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

It is anticipated that additional technical support will be needed to establish more closely PCI failure predictions and their correlation to release rates. The cost is estimated directly in the next step.

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

For further model development to accomplish the above, it is anticipated that an additional \$200,000 will be needed. Thus, C_D = \$2.0E+05

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

NRC review of implementation procedures and training procedures at individual plants should require approximately 2 man-wk per operating plant.

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

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

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

NC = (\$4540/plant)(71 operating plants) = \$3.22E+05

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

Once the SIR is implemented, it is anticipated that review of results from power-maneuvering changes will be required for several years. In addition, an assessment of the improvement in any fuel assembly design changes will be required. This review would require perhaps 1 man-wk/py over the 5 years subsequent to any changes. It is assumed that the 71 operating plants will fall under the 5-year review, and that the

TABLE 1. (cont'd.)

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

additional 25 plants in the planning phase and scheduled for start-up within that time will be under review for an average of 2.5 of the 5 years.

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

C_o = (1 man-wk/py)($2270/man-wk = $2270/py (for a 5-year period only)

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

$N\bar{T}C_o$ = ($2270/py)[(71 plants)(5 yr) + (25 plants)(2.5 yr)] = $9.48E+05

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$1.5E+06	$2.0E+06	$9.6E+05

REFERENCES

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

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

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: B-50, Post-Operating Basis Earthquake Inspection

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

In the event of an Operating Basis Earthquake (OBE) the functional capability of a nuclear power plant must be determined. This implies inspection of all safety-related equipment and other equipment necessary for operation. At present, this inspection is required, but no inspection procedures exist. Resolution of this issue is intended to provide a complete plan for post-OBE inspection.

AFFECTED PLANTS[(a)]			
	PWR:	Operating = NA	Planned = NA
	BWR:	Operating = NA	Planned = NA

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	0
OCCUPATIONAL DOSES:	
SIR Implementation =	-150
SIR Operation/Maintenance =	0
Total of Above =	-150
Accident Avoidance =	0

COST RESULTS ($1E+06)

INDUSTRY COSTS:	
SIR Implementation =	-3.0
SIR Operation/Maintenance =	0
Total of Above =	-3.0
Accident Avoidance =	0
NRC COSTS:	
SIR Development =	0.045
SIR Implementation Support =	-0.58
SIR Operation/Maintenance Review =	0
Total of Above =	-0.54

(a) Only one generic LWR is assumed to be affected. There is no breakdown into PWRs or BWRs for this analysis.

POST-OPERATING BASIS EARTHQUAKE INSPECTION

ISSUE B-50

1.0 SAFETY ISSUE DESCRIPTION

Appendix A of CFR 10 Part 100 specifies that the operating basis earthquake (OBE) shall be defined by response spectra and that the maximum vibratory ground acceleration of the OBE shall be at least one-half of the maximum vibratory ground acceleration of the safe shutdown earthquake (SSE). Suitable instrumentation is required at the reactor site, so that the seismic response of the nuclear power plant components that are important to safety can be determined promptly in order to permit a comparison of such response with that used as a design basis. Some detailed guidance on the nature and extent of this seismic instrumentation is provided in Regulatory Guide 1.12, Rev. 1, "Instrumentation for Earthquakes," April 1984. Shutdown of the nuclear plant is required in the event that vibratory ground motion exceeds that of the OBE. Prior to resuming operation, the licensee is required to demonstrate that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public.

In attempting to assess the functional capability of the plant to resume operation, field inspection of the safety-related equipment should be implemented. This inspection is intended to assure that no functional damage has occurred to the plant equipment necessary for safe operation and to determine the plant's ability to safely withstand future seismic and nonseismic loadings.

PROPOSED ISSUE RESOLUTION

Although the necessity for inspection requirements is stated in 10 CFR 100, Appendix A, Section V a(2), and in Standard Review Plan 3.7.4.II.4, the details for such an inspection procedure are not provided. Resolution of this issue is intended to provide a complete plan for post-OBE inspection. This will include a comprehensive and systematic integrated checklist to identify which structures, systems and components must be inspected, including a description of the extent of the inspection to be performed. A decision tree approach has been suggested. The initial inspection should be designed to uncover potential problem areas. Subsequent investigations may or may not be necessary, based upon the results of conditional checks as the inspection progresses through the decision tree. Procedures may include a combination of inspection techniques, including visual inspection, in-situ testing, nondestructive examination and analytical confirmation. It is recommended that the plan be system or subsystem oriented. Individual components such as valves and pumps should undergo operability checks to demonstrate that they meet their functional requirements safely (NUREG-0471, NRC 1978).

2.0 SAFETY ISSUE RISK AND DOSE

"The 'Operating Basis Earthquake' is that earthquake which, considering the regional and local geology and seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant; it is that earthquake which produces the vibratory growth motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional." (10 CFR 100, 1981)

To date, there have been no OBEs at nuclear power plant (NPP) sites. Thus, post-OBE inspections have not been needed. This issue analysis assumes that, in the event of an OBE, plant shutdown occurs to allow for completion of all inspections to assure safe operation.

It is currently envisioned that in the event of an OBE, a detailed and time-consuming post-shutdown inspection of all critical safety components of a plant would be conducted by the utility. The safety issue resolution (SIR) involves development of a detailed but specific post-OBE inspection procedure designed to streamline the inspection process (thereby to reduce the inspection time) and ensure safe plant operation. Thus, the SIR is assumed to focus the post-OBE inspection on the critical safety components and to ensure that damaged critical plant components are not overlooked.

Although the probability of overlooking damaged critical plant components may be reduced through resolution of this issue, it is assumed that the probability of core melt will not be significantly reduced. However, resolution of this issue is expected to reduce occupational dose exposure and plant downtime due to the acceptance of a systematic approach to a post-OBE inspection. The Occupational Dose Work Sheet is included as Table 1.

TABLE 1. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Post-Operating Basis Earthquake Inspection (B-50)

2. Affected Plants (N):

   Although all plants can potentially be affected, prior history of OBEs indicates that this potential is extremely small (none has occurred at an operating plant; one has occurred at Humboldt Bay during a refueling outage). Therefore, only one plant will be considered affected over the lives of all plants, for the purpose of implementing the SIR in this analysis. This plant will be viewed as a generic LWR, rather than either a PWR or a BWR.

TABLE 1. (cont'd.)

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

Based on Appendix C of NUREG/CR-2800 (Andrews et al. 1983), the average remaining life of all plants is 28.3 yr. This value is assumed as $\bar{T}$ for the one affected plant.

4-5. <u>Steps Related to Occupational Dose Reduction Due to Accident Avoidance</u>:

Since the core-melt frequency is presumed to be unaffected, no occupational dose reduction will be realized from accident avoidance. Thus, $\Delta U = 0$.

6. <u>Per-Plant Utility Labor in Radiation Zones for SIR Implementation</u>:

Throughout this analysis, it is assumed that SIR implementation occurs at the time when the developed OBE inspection procedures are implemented following an OBE occurrence.

<u>Base Case</u>

Assuming that no inspection plan is available following an OBE event, it is estimated that two weeks would be required for a post-OBE inspection.[a] This assumes that an inspection team of 10 persons is available during the two weeks. Throughout this issue analysis, plant downtime will be related only to that time required to assure functional capability of plant facilities.

<u>Adjusted Case</u>

Assuming that, with a comprehensive and systematic inspection procedure specified in advance (adjusted case), the inspection process could be completed more expeditiously and require only 4 days, with the same number of licensee personnel with no special seismic knowledge of the plant.

<u>Total Labor</u>

The calculations below are based on an average 20-hour work day. This reflects the need for power in a situation where there will be massive devastation to the surrounding region.

(4 days/man)(20 man-hr/day)(10 men/plant)

- (14 days/man)(20 man-hr/day)(10 men/plant)

= -2000 man-hr/plant

(a) Personal communication with the NRC.

TABLE 1. (cont'd.)

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

An average dose rate of 75 mR/hr is assumed.[a]

D = (-2000 man-hr/plant)(0.075 R/hr) = -150 man-rem/plant

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

ND = (1 plant)(-150 man-rem/plant) = -150 man-rem

9-11. Steps Related to Occupational Dose Increase for SIR Operation and Maintenance:

Since resolution of this issue involves implementation of an inspection procedure at a specified time, no ongoing operation and maintenance program is anticipated. Thus, $D_O = 0$.

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
-150	-50	-450

3.0 SAFETY ISSUE COSTS

Resolution of this safety issue will provide a post-OBE inspection procedure adaptable to all PWRs and BWRs. Industry and NRC costs will be incurred in the development of the procedure, but once established, it is anticipated that inspection time following an OBE will be decreased. Cost analysis results are summarized in Table 2.

(a) From Table 12.2-22b, Amendment 7 of Virgil Summer PWR FSAR, August 1978.

TABLE 2. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

Post-Operating Basis Earthquake Inspection (B-50)

2. Affected Plants (N):

Development of a detailed post-OBE inspection procedure is assumed to require the efforts of all licensees (for both operating and planned plants). Thus, costs for development of this procedure are assumed to be incurred by all 134 plants. However, historical data indicate that the likelihood of an OBE occurring is extremely small (none has occurred so far at an operating plant; one has occurred at Humboldt Bay during a refueling outage). Thus, one plant is presumed to be affected. This one plant is viewed as a generic LWR, rather than as a PWR or a BWR.

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

The average remaining life of all plants is 28.3 yr. This value is assumed as $\bar{T}$ for the one affected plant.

Industry Costs (Steps 4 through 12)

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

Since the core-melt frequency is presumed to be unaffected, no cost savings will be realized by the industry from accident avoidance. Thus, $\Delta H = 0$.

6. Per-Plant Industry Resources for SIR Implementation:

Industry implementation of the SIR will involve the following three items:

1. Development of a detailed post-OBE inspection procedure (possibly requiring technical assistance from non-utility sources)--an effort shared by all 134 plants.

2. Reduction in labor time during the post-OBE inspection (reduced labor time results from streamlined inspection procedure).

3. Reduction in scheduled down-time for the post-OBE inspection (reduced down-time results from streamlined inspection procedure).

Resources for the latter two items are discussed below. The cost to develop the detailed post-OBE inspection procedure is estimated directly in Step 8.

TABLE 2. (cont'd.)

Reduced Labor

The detailed inspection procedure would be implemented only by those plants experiencing an OBE event. In this case, we assume only one affected plant over the lifetime of all plants. From Step 6 of the Occupational Dose Work Sheet, the total labor time saved due to implementation is estimated to be 2000 man-hr/plant.

Reduced Down-Time

Due to the difficulty in predicting the ramifcations of an OBE on down-time, only that down-time directly related to the inspection time necessary to assure the functional capability of equipment is considered here. In Step 6 of the Occupational Dose Work Sheet, a reduction in inspection time of 10 days (from 2 weeks to 4 days) was estimated. This is taken as the reduction in scheduled down-time. This reduction assumes only minor equipment replacement or repair.

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

Procedures Development: See Step 8.

Reduced Labor: (-2000 man-hr/plant)(1 man-wk/40 man-hr)

($2270/man-wk) = -$1.14E+05/plant

Reduced Down Time: (-10 days/plant)($3.0E+05/day) =
-$3.0E+06/plant

(Negative signs indicate cost savings.)

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

It is assumed that the effort involved in developing the inspection procedure will be borne by the industry. The procedure will be submitted to the NRC for approval and incorporation into the Regulatory Guides. A possibility currently being discussed would be to have the American National Standards Institute develop this procedure. It is anticipated that technical assistance required to develop the inspection procedure would be approximately one man-year. Therefore,

NI = (1 man-yr)($1.0E+05/man-yr) + (1 plant) [(-$1.14E+05/plant)

+ (-$3.0E+06/plant] = -$3.01E+06

TABLE 2. (cont'd.)

9-11. Steps Related to Industry Cost and SIR Operation and Maintenance:

This issue does not address plant operation and maintenance beyond the implementation of the detailed inspection procedures developed, thus, $I_O = 0$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
-\$3.0E+06	-\$1.5E+06	-\$4.5E+06

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

It is anticipated that costs for developing this SIR will be carried by industry. However the resolution would then be submitted to the NRC for review. It is anticipated that such a review would take approximately 20 man-wk.

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

C_D = (20 man-wk)(\$2270/man-wk) = \$4.54E+04

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

Once again, it is assumed that SIR implementation is limited to the inspection procedure as detailed in the resolution. It is difficult to anticipate what resources would be required beyond this direct application.

Base Case

Assuming that the NRC would be involved in reviewing inspection procedures as well as supporting the implementation, it is anticipated that during the 2-week base case inspection procedure and the 2 weeks subsequent to startup, 3 full-time NRC staff members would be required.

(160 man-hr/man)(3 men) = 480 man-hr/plant

Adjusted Case

Assuming that review of the procedure has been completed, it is anticipated that only 2 full-time NRC staff members would be required during the 4-day inspection and 2 weeks subsequent to startup.

(4 days + 10 days)(8 hours/day)(2 men) = 224 man-hr/plant

TABLE 2. (cont'd.)

15. Per-Plant NRC Labor for Support of SIR Implementation: (cont'd.)

Total Labor

224 man-hr/plant - 480 man-hr/plant = -256 man-hr/plant

(Negative sign indicates labor time reduction.)

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

Assuming that an OBE occurs,

(-256 man-hr/plant)($2270/man-hr) = -$5.81E+05/plant

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

NC = (1 plant)(-$5.81E+05/plant) = -$5.81E+05

18-20. Per-Plant NRC Labor for Review of SIR Operation and Maintenance:

Since this issue addresses the development and implementation of an inspection plan only, costs for review of SIR operation and maintenance are not foreseen. Thus, $C_O = 0$.

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
-$5.4E+05	-$2.4E+05	-$8.3E+05

REFERENCES (For Issue B-50)

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ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: B-59, (N-1) Loop Operation in BWRs and PWRs

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION

When a PWR reactor coolant pump or a BWR recirculation pump becomes inoperative, the flow provided through the remaining loops (i.e., N-1, where N is the total number of loops) is sufficient for steady state operation at less than full power. However, the NRC staff has disallowed this mode of operation for most plants primarily because of insufficient analysis of ECCS response. The proposed resolution is to develop a set of acceptance criteria and review guidelines for the authorization requests made by utilities for (N-1) loop operation on an extended time basis.

AFFECTED PLANTS		
	PWR: 45	Planned = 0
	BWR: 22	Planned = 0

RISK/DOSE RESULTS (man-rem)

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

COST RESULTS ($1E+06)

INDUSTRY COSTS:	
SIR Implementation =	64
SIR Operation/Maintenance =	-18
Total of Above =	46
Accident Avoidance =	0
NRC COSTS:	
SIR Development =	0.12
SIR Implementation Support =	0.15
SIR Operation/Maintenance Review =	4.3
Total of Above =	4.6

(N-1) LOOP OPERATION IN BWRs AND PWRs

ISSUE NUMBER B-59

1.0 SAFETY ISSUE DESCRIPTION

The majority of presently operating BWRs and PWRs are designed to operate with less than full reactor coolant flow. If a PWR reactor coolant pump or a BWR recirculation pump becomes inoperative, the flow provided through the remaining (N-1) loops is sufficient for steady state operation at a power level less than full power. Although the FSARs for the licensed BWRs and PWRs present (N-1) loop calculations showing allowable power and protective system trip set-points, the NRC staff has disallowed this mode of operation for most plants primarily because of insufficient Emergency Core Cooling System (ECCS) analyses. Some Babcock and Wilcox (B&W) PWRs are authorized for long-term operation with one reactor coolant pump out of service since they have submitted and the NRC staff has approved the necessary ECCS, steady state, and transient calculations. It should be noted that all but one of the operating and planned B&W PWRs have two pumps in each of two loops. In addition, the NRC approved the Westinghouse PWR at the Beaver Valley Power Station No. 1 (BV-1) for (N-1) loop operation in 1984. The Technical Specifications of the remaining BWR and PWR licensees require shutdown within a fairly short time if one of the reactor coolant loops becomes inoperable.

PROPOSED SAFETY ISSUE RESOLUTION

The purpose of this task action plan item is to develop a set of acceptance criteria and review guidelines for the expeditious resolution of (N-1) loop authorization requests. This set of criteria and guidelines most likely will include accident scenarios (both LOCAs and non-LOCAs) to be analyzed by the licensees, computer models acceptable to NRC for these analyses and acceptable input parameters in terms of reactor operating conditions (such as allowance for uncertainties in power level and fluid measurement). A staff report will be prepared summarizing the NRC criteria for each Nuclear Steam Supply System (NSSS) design.

It is recognized that establishing criteria and guidelines will not affect risk unless plants apply for (N-1) loop operation approval and then exercise the option to run with (N-1) loop when the situation occurs. Therefore the rest of this analysis is based on plants applying for approval and exercising the (N-1) loop operation option.

2.0 SAFETY ISSUE RISK AND DOSE

Allowing (N-1) loop operation gives utility operators more flexibility in deciding whether to shut down a plant or let it operate at a reduced power level. In this safety issue, (N-1) loop operation is restricted to operation that results from a single reactor coolant or recirculation pump failure. Since fixing an out-of-service pump is usually a major task, it is not expected that the pumps will be repaired while the plant is on-line. By continuing (N-1) loop operation, the repair work may be delayed until the next scheduled refueling time if the period is not too far in the future. Prolonged (N-1) loop operation is not economical.

The risk introduced by Option 1 (shutting the plant down, repairing the pump, and starting the plant back up) is compared to the risk of Option 2 (operating the plant with only (N-1) loops in operation). There are three parts to the risk introduced by Option 1: 1) risk related to the occurrence of a transient during the controlled power descent to a hot shutdown situation for repair of the pump, 2) very low level of risk associated with the plant being in hot shutdown, and 3) risk of starting the plant back up and running at 100 percent power. There are two parts to the risk of Option 2: 1) risk related to the occurrence of a transient during the controlled power descent to (N-1) loop operation (i.e., reduced power operation) and 2) risk of running at reduced power.

Part 2 of Option 1 has not been addressed in other issues where downtime was a part of the analysis of risk. To be consistent with the other issues, the risk associated with the plant being in hot shutdown for pump repair is not addressed in this issue.

The risk related to parts 1 and 3 of Option 1 versus the risk related to parts 1 and 2 of Option 2 can not be quanitified without a significant analysis effort because of the complexity of concerns to be evaluated. Also a qualitative comparison of risk is not feasible because of the complexity of concerns and the fact that no overriding advantages or disadvantages are obvious. Therefore due to the uncertainty that Option 1 results in more or less risk than Option 2, zero risk reduction is assumed for the risk change for implementation of this issue.

For occupational exposure, the repair work will be required at some time during either a forced or scheduled outage. The radiation exposure would thus be equal under either case, indicating that no increases or decreases in occupational exposure would result from allowing operation with (N-1) loops.

Therefore, the safety issue resolution (SIR) as described in Section 1.0 will not affect either public risk or occupational doses.

Since any potential risk reduction is perceived to be negligible, no Public Risk Reduction Work Sheet has been prepared. No occupational dose will accrue since the SIR only involves some additional analyses (i.e., for (N-1) loop operation). Thus, no Occupational Dose Work Sheet has been prepared.

3.0 SAFETY ISSUE COSTS

To estimate the cost to industry, it is assumed that the work performed by Duquesne Light for Beaver Valley Power Station No. BV-1 is applicable (Dunn 1978).

BV-1 analyzed an (N-1) loop large break LOCA and 12 non-LOCAs. Accidents involving the partial loss of forced reactor coolant flow, startup of an inactive reactor coolant loop, single reactor coolant pump locked rotor, and complete loss of forced reactor coolant flow were analyzed in the original FSAR. Therefore, they were not reanalyzed. This leads to 13 transient scenarios to be analyzed. Since BV-1 has been approved for the (N-1) loop operation and the Millstone Unit 3 plant is under review by NRC, it is expected that more plants (although perhaps not all of them) will make a request for approval of (N-1) loop operation. For this prioritization analysis, we assume that half of the 90 PWRs and half of the 44 BWRs will submit the (N-1) loop analysis for the NRC approval.

Using an estimated resource requirement of 10 man-wk and 15 computer hours for each of the 13 cases yields a total of 130 man-wk and 195 computer hours. Another 30 man-wk per plant are allowed for modifying and upgrading procedures and/or systems and familiarizing operations staff with upgrades. Using the industry rate of \$2270/man-wk and an estimated computer cost of \$3000/hr,[a] the total cost is \$9.5E+05 per plant. This same cost per unit is assumed for both PWRs and BWRs.

For this issue, it is assumed that there is no incremental increase or decrease in labor for operation and maintenance due (N-1) loop operation.

Clark and Barrow (1979) and Olsen (1981) reported that the main contributor to pump failures is pump seal failure. Clark and Barrow (1979) indicated that for Oconee 1, 88 percent of pump failure events are due to pump seal problems and 99 percent of pump maintenance time is spent on seal fixes. Since the non-seal failures contribute only 1 percent of the total maintenance time in the Oconee case, they are neglected in this analysis and pump seal failure probability is used as the probability of losing one pump and operating under (N-1) loop conditions.

Although the failure frequency of PWR pump seals that contribute significantly to core-melt frequency is only 0.02 per plant-year (Kolb 1982), seal failures that result in the loss of one loop are estimated to occur at a rate of 0.36/py for both PWRs and BWRs (see Safety Issue Prioritization for MPA G-1, PNL). It is clear that prolonged (N-1) loop operation would not be economical if it occurs early in a cycle.

Cost of Forced Outages

Current practice requires plant shutdown given a loop failure. Assuming that pump repair will require a 10-day outage and that the cost of replacement power is \$300,000 per full power day (FPD), the economic penalty for a 10-day outage will be \$3E+06.

It is assumed that the outage could occur randomly any time during an assumed 330-day normal operating cycle. The economic penalties must then be adjusted to reflect the fact that failures could occur within 10 days of a

(a) Computer usage costs to the government are 10 times less. See Issue I.C.I.(4).

normal outage. With a loop failure frequency of 0.36/py, the probability of failure occurrence where more than 10 days exists prior to a normal outage must then be adjusted by the factor (330-10)/330. Based on this probability of failure occurrence, the estimated economic penalty for a failure and a forced outage during this period will be F1, where

$$
\begin{aligned}
F1 &= [(330-10)/330](0.36/\text{yr})(10 \text{ days})(\$300{,}000/\text{day}) \\
&= \$1.047E{+}06/\text{yr}.
\end{aligned}
$$

The probability of occurrence within 10 days of a scheduled outage must then be adjusted by the factor (10/330). Note also that the outage can overlap with the scheduled outage, with an assumed average of 5 days of outage occurring within the 10-day period of interest. Based on this probability of occurrence, the estimated economic penalty for loop failure within 10 days of the scheduled outage, F2, is then

$$
\begin{aligned}
F2 &= (10/330)(0.36/\text{yr})(5 \text{ days})(\$300{,}000/\text{day}) \\
&= \$1.64E{+}04/\text{py}.
\end{aligned}
$$

The total economic penalty estimated for the current policy of forced outage, F, is then

$$
\begin{aligned}
F &= F1 + F2 = \$1.047E{+}06 + \$1.64E{+}04 \\
&= \$1.06E{+}06
\end{aligned}
$$

Cost of Option to Continue Operation

If the utilities are given the option of continuing operation with one loop out, they must decide whether to continue operating at reduced power until the scheduled outage period, or shut down immediately for repairs. There will be an economic break even point where the cost of replacement power, assuming a 10-day outage, would equal the cost of replacement power due to operation at reduced power over a longer period of time; beyond that point, an economic penalty will be incurred.

The economic penalty for operation at reduced power will then be

$$(X)(\$3E{+}5/\text{day})(1/N),$$

where

X = days of operation at reduced power
N = number of loops in the plant.

Comparing this to the cost of a 10-day outage [(10 days)($3E+05/day)], it can be seen that the economic break even point occurs at X = 10N days. For a four, three, and two-loop plant this occurs at 40, 30, and 20 days from the normally scheduled outage, respectively. Again, this means that if the failure occurs more days ahead of the scheduled outage than this, it would be economical to shut down the plant for repairs regardless of the NRC policy concerning the owner's options.

Where operation continues, the estimated economic penalty, C1, for outages occurring within 10 days of the scheduled outage is:

$$C1 = (10/330)(0.36/yr)(\$3E+05/day)(1/N)(5\ days).$$

The above assumes an average 5-day outage for random failures occurring within this 10-day period. The estimated economic penalty, C2, for outages occurring at greater than 10 days but within the period where continued operation is economical is then:

$$C2 = [(10N-10)/330](0.36/yr)(\$3E+05/day)(1/N)(10N\ days).$$

The estimated economic penalty, C3, for outages occurring at greater than 10N days (i.e., shutdown still recommended but for economic reasons) is then:

$$C3 = [(330-10N)/330](0.36/yr)(\$3E+05/day)(10\ days).$$

The total annual cost of replacement power using the new regulatory policy, C = C1+C2+C3, is summarized in the table below. In addition, the economic benefit predicted for implementing this policy (forced outage minus continued operation, or F-C) is given.

Predicted Annual Cost Due to Pump Failures, $/py

No. of Loops	(F-C)	C	C1	C2	C3
5	1.28E+04	1.0506E+06	3.273E+03	1.309E+05	9.164E+05
4	1.20E+04	1.0514E+06	4.091E+03	9.818E+04	9.491E+05
3	1.07E+04	1.0527E+06	5.455E+03	6.545E+04	9.818E+05
2	7.50E+03	1.0559E+06	8.182E+03	3.273E+04	1.015E+06

The predicted economic benefit ranges from $1.28E+04/py for a five-loop plant to $7.50E+03/py for a two-loop plant. Multiplying the predicted annual cost savings (F-C) by the respective number of plants and then summing these cost savings and dividing by the number of affected plants gives an average annual cost savings per plant-year. This calculation is represented below:

Estimated Average Annual Cost Savings Per Plant-Year, $/py

No. of Loops	No. of Plants, P	(F-C)	P * (F-C)
5	2	1.28E+04	2.560E+04
4	42	1.20E+04	5.040E+05
3	16	1.07E+04	1.712E+05
2	73	7.50E+03	5.475E+05
	133		1.248E+06

Therefore 1.248E+06/133 = 9.4E+03. Thus the average annual cost savings is $9.4E+03/py and this value is used in the following calculations.

The cost to NRC of developing a set of acceptance criteria and review guidelines for the issue will be limited to that for the BWRs. It is assumed that for PWRs these guidelines should have been available through the review and approval of the BV-1 analysis. For the prioritization purpose, 10 man-wk of NRC staff labor and $100,000 for contractor support are allowed. NRC labor to support SIR implementation should be minimal, about 1 man-wk/plant. To review SIR analyses and operation for each reload fuel cycle, 1 man-wk/py of NRC labor is assumed.

The results of the industry and NRC cost analyses are summarized in Table 1.

TABLE 1. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

(N-1) loop operation in BWRs and PWRs (B-59).

2. Affected Plants (N):

67 plants (45 PWRs and 22 BWRs).

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

	$\bar{T}$ (yr)
PWRs:	28.8
BWRs:	27.4
Average	28.3

Industry Costs (Steps 4 through 12)

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

Since there is no change in the core-melt frequency as a result of SIR, no cost savings is realized for accident avoidance, i.e., H = 0.

6. Per-Plant Industry Resources for SIR Implementation:

Labor = 160 man-wk/plant

Computer time = 195 hr/plant

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

Labor = (160 man-wk/plant)($2270/man-wk) = $3.63E+05/plant

Computer time - (195 hr/plant)($3000/hr) = $5.85E+05/plant

I = $9.48E+05/plant

TABLE 1. (cont'd.)

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

$$NI = (67 \text{ plants})(\$9.48E+05/\text{plant}) = \$6.4E+07$$

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

No incremental labor increase or decrease assumed.

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

Savings in avoidance of downtime (see text) = -\$9.4E+03/py

(Negative sign indicates savings)

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

$$N\bar{T}I_o = (67 \text{ plants})(28.3 \text{ yr})(-\$9.4E+03/\text{py}) = -\$1.8E+07$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$4.6E+07	\$7.9E+07	\$1.3E+07

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC Staff Labor = 10 man-wk

Contractor Support (costs estimated directly in next step)

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

Labor = (10 man-wk)(\$2270/man-wk) = \$2.27E+04

Contractor Support = \$1.0E+05

C_D = \$1.2E+05

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

1 man-wk/plant

TABLE 1. (cont'd.)

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):

$NC = (67 \text{ plants})(\$2270/\text{plant})$

$= \$1.5E+05$

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

Labor = 1 man-wk/py

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

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

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

$N\bar{T}C_o = (67 \text{ plants})(28.3 \text{ yr})(\$2270/\text{py})$

$= \$4.3E+06$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$4.6E+06	$6.8E+06	$2.4E+06

REFERENCES (To Issue B59)

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.

Clark, R. H. and W. E. Barrow. 1979. Limiting Factor Analysis of High Availability Nuclear Plants. EPRI-NP-1138, Electric Power Research Institute, Palo Alto, California.

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Makay, E. and M. L. Adams. 1979. Operation and Design Evaluation of Main Coolant Pumps for PWR and BWR Service. EPRI-NP-1194, Electric Power Research Institute, Palo Alto, California.

Olsen, E. A. J. 1981. Nuclear Unit Operating Experience - 1978 and 1979 Update. EPRI-NP-2092, Electric Power Research Institute, Palo Alto, California.

ISSUE SUMMARY WORK SHEET

ISSUE NO./TITLE: B-65, Iodine Spike

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

Following a temperature/pressure transient, a spike can occur in which the iodine concentration in the reactor coolant rises to 10-20 times its equilibrium level. A coincident LOCA could result in an offsite radioiodine release. Model development and confirmation are needed to predict potential offsite releases from such spikes. These releases can then be minimized by setting limiting conditions of operation specifically addressing the iodine spiking problem.

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

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION = 12

OCCUPATIONAL DOSES

SIR Implementation =	71
SIR Operation/Maintenance =	300
Total of Above =	370
Accident Avoidance =	0

COST RESULTS ($1E+6)

INDUSTRY COSTS:

SIR Implementation =	0.65
SIR Operation/Maintenance =	1.4
Total of Above =	2.0
Accident Avoidance =	0

NRC COSTS:

SIR Development =	0.50
SIR Implementation Support =	0
SIR Operation/Maintenance Review =	0.86
Total of Above =	1.4

IODINE SPIKE

ISSUE B-65

1.0 SAFETY ISSUE DESCRIPTION

This safety issue is similar in nature to Issue 74, "Iodine Coolant Activity Limiting Conditions for Operation." The goal of that issue is to establish uniform iodine concentration limits for PWRs and BWRs. The basis for this limiting concentration is a maximum acceptable public exposure calculated for a "worst site" meteorology being applied to all plants. The design basis accidents were represented by a postulated steam generator tube rupture (SGTR) in a PWR or steam line break in a BWR. Similarly, the allowable secondary activity in a PWR was limited by a postulated secondary coolant steam line break. As a result, the following equilibrium iodine concentration limits were proposed: 1.0E-06 and 0.1E-06 Ci/g of I-131 in the primary and secondary coolant of a PWR, respectively, and 0.2E-06 Ci/g of coolant for BWRs.

It was recognized in Issue 74 that transient-induced spiking of iodine concentrations would occur in the coolant. As a result, the limiting conditions postulated for Issue 74 would allow for the continued operation of a plant with elevated iodine levels for a period not to exceed 5 percent of its annual operating time.

Safety Issue 74 deals primarily with equilibrium iodine concentration levels observed in the coolant. However, the concentrations can increase significantly from equilibrium levels following a transient in the cooling system. Safety Issue 65 would focus on this iodine-spiking phenomena. In order to couple offsite dose limits with coolant iodine concentration limits as in Issue 74, the behavior of iodine release from fuel for various conditions and exposure histories during postulated transients must be understood. Once fully modeled, new limiting conditions of operation (LCOs) would likely be instigated.

PROPOSED ISSUE RESOLUTION

The proposed resolution to Issue B-65 is the development and confirmation of a model for the iodine-spiking phenomenon. Procurement of data from operating plants and the development of a fuel release model for predicting the magnitude of the spikes will provide an understanding of this phenomenon which is not presently available. Improved knowledge of this topic will allow setting of the coolant activity limits at realistic levels. In addition, this safety issue resolution (SIR) will provide the basis for more realistic accident calculations.

AFFECTED PLANTS

New limiting conditions on coolant activity concentrations would apply to all operating and planned light water reactors.

2.0 SAFETY ISSUE RISK AND DOSE

The analyses for public risk reduction and occupational dose are discussed in this section.

PUBLIC RISK REDUCTION

This issue affects public risk only through non-core-melt accidents. The analysis is presented in Table 1 and Attachment 1.

OCCUPATIONAL DOSE

From currently available data (Pasedag 1977), it is judged that the four-hour sampling interval for iodine activity surveillance after a transient, as proposed in NRC LCOs, will probably miss some spiking peaks.[a] A two-hour sampling interval should provide adequate information for peak spike activity calculations.

Because the SIR requires the development of a model for iodine spiking which is currently unavailable for any plants, it is assumed that all plants are affected.

To estimate the utility labor in radiation zones for SIR operation and maintenance, the requirement for additional sampling of the primary coolant after each power transient is estimated. There have been 80 occurrences of iodine spikes in the past three years. Among them are 70 PWR events and 10 BWR events. Based on Appendix C of NUREG/CR-2800 (Andrews et al. 1983), there are 135 reactor-years for PWRs and 72 reactor-years for BWRs. The calculated frequencies for iodine spikes become 0.52/py at PWRs and 0.14/py at BWRs. The difference between these frequencies and the total non-core-melt frequencies (1.3E-03/py for PWRs and 1.4E-03/py for BWRs) results because the non-core-melt events involve only those with iodine activities released to the environment, whereas the frequencies of 0.52/py and 0.14/py are for total spike occurrences whether released or not.

The total sampling period after each transient is estimated to be three hours. This is based on the data given by Pasedag (1977). Current LCOs require one sampling per 4-hour period, i.e., eight samplings in a 33-hour period. If we use a 2-hour sampling interval, 16 samplings in a 33-hour period are needed. Therefore, the additional sampling requirement is eight samplings.

Analysis results are summarized in Table 2.

(a) Personal communication with F. Akstulewiez, Nuclear Regulatory Commission, 1983.

TABLE 1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

Iodine Spike (B-65)

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

All Plants

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

3. Plants Selected for Analysis:

Base-case PWR calculations are derived from information on an SGTR event at Prairie Island 1. Base-case BWR calculations are scaled from the PWR results using original values from NUREG/CR-2800 (Andrews et al. 1983) given for Grand Gulf 1.[a]

4-8. Steps Related to Affected Parameters, Accident Sequences, Release Categories, Core-Melt Frequency, and Their Base-Case Values:

Core-melt frequency is unaffected by Issue B-65. Therefore, F = 0 for both PWRs and BWRs. The base-case, affected public risk is estimated from non-core-melt accidents (an SGTR for a PWR and a small break LOCA for a BWR) and given directly in the next step.[a]

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

W_{PWR} = 0.0143 man-rem/py[a] W_{BWR} = 0.0185 man-rem/py[a]

10-13. Steps Related to Adjusted-Core Values for Affected Parameters, Accident Sequences, and Release Categories and Core-Melt Frequency:

As mentioned above, core-melt frequency is unaffected. Therefore, F* = 0 for both PWRs and BWRs. The adjusted-case, affected public risk is estimated from non-core-melt accidents and given directly in the next step.[a]

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

W^*_{PWR} = 0.0114 man-rem/py[a] W^*_{BWR} = 0.0148 man-rem/py[a]

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

$\Delta\bar{F}$ = 0 for both PWRs and BWRs

(a) See Attachment 1.

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

$$\Delta W_{PWR} = 0.0143-0.0114 = 0.0029 \text{ man-rem/py}$$

$$\Delta W_{BWR} = 0.0185-0.0148 = 0.0037 \text{ man-rem/py}$$

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

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
12	1800	0

ATTACHMENT 1 (To Table 1)

The procedures outlined in NUREG/CR-2800 (Andrews et al. 1983) have to be modified for this SIR because the iodine spike becomes an issue only in non-core-melt accidents.

As mentioned in Section 1.0, the limiting transient on iodine activity release for PWRs is the SGTR accident. Since iodine spikes are usually associated with large power level changes, they are very likely to occur during an SGTR transient where a rapid power decrease is expected. A June 1983 report by the NRC Task Force on SGTR estimated the total probability for non-core-melt SGTR events to be 1.3E-03/py.[a] From the SGTR report, the NRC staff estimated the total I-131 release for the non-core-melt SGTR events to be 53,600 Ci.

From the information given in NUREG-0651 (Marsh 1980) for the Prairie Island 1 SGTR event, a release of 210 μCi of I-131 results in a population dose of 4.3E-06 rem to the thyroid. Using this conversion factor, it follows that the total I-131 dose to the thyroid from a non-core-melt SGTR is (53,600 Ci)(4.3E-06 rem/2.10E-04 Ci) = 1100 rem.

To convert this to whole-body dose (man-rem), as utilized in NUREG/CR-2800, the conversion developed in Issue II.A.1.3, "Maintain Supplies of (Potassium Iodide) Thyroid Blocking Agent," is used. There, two assumptions are made:

1. Health effects from thyroid dose are 95 percent curable with no long-term effects.

2. Whole-body dose is given five times the weighting of thyroid dose in protective action guides (NRC 1980).

Thus, thyroid dose should be reduced by a factor (1/0.05)(5) = 100 to give an equivalent whole-body dose (man-rem) for consistency with other issues.

Based on the above, the total whole-body dose from a non-core-melt SGTR is (1100 thyroid-rem)/(100 thyroid-rem/man-rem) = 11 man-rem. The total expected dose from coincident iodine spiking and a non-core-melt SGTR becomes (1.3E-03/py)(11 man-rem) = 0.0143 man-rem/py. This is assumed to be the base-case, affected public risk for a PWR.

From NUREG/CR-2800, the total public risk for a BWR is 1.2 times that for a PWR, based on the original release category frequencies given for Oconee 3 and Grand Gulf 1 and the whole-body dose conversion factors in Appendix D. However, at a BWR, an SGTR cannot be the initiator of the type of release experienced at Prairie Island 1. Rather, it is assumed that a small-break LOCA (frequency = 1.4E-03/py) is the analogous initiator. Therefore, the total expected dose from coincident iodine spiking and a non-core-melt small-break LOCA becomes (1.2)(0.0143 man-rem/py)(1.4E-03/py)/(1.3E-03/py) = 0.0185 man-rem/py. This is assumed to be the base-case, affected public risk for a BWR.

(a) Personal communication with W. Milstead, U.S. Nuclear Regulatory Commission, 1983.

In the analysis by the NRC Task Force on SGTR, a spiking factor of 20 was used. To our knowledge, the equilibrium I-131 concentrations on which this factor of 20 is applied are 1.0 and 0.1 μCi/g in the primary and secondary coolant, respectively, for PWRs, and 0.2 μCi/g of coolant for BWRs. Actual plant data (Denton 1974) have shown that the PWR primary I-131 concentrations are in the 0.01 to 0.1 μCi/g level and that the BWRs operate with I-131 concentrations of approximately 0.01 μCi/g of coolant. These concentrations are much lower than the above LCO levels. On the other hand, plant data also indicate that the iodine spike factors range from 7 to 1030 (Pasedag 1977).

Plants with high initial equilibrium concentrations tend to show low spikes in the transient, while systems with low initial equilibrium concentrations show greater spikes in the transient. Plants may experience postulated transients such as an SGTR or main steam line break (MSLB) with potential release of I-131 higher than would be calculated if based on the NRC-proposed LCOs. However, the above effect minimizes high release. For example, assuming an actual spiking factor of 500 (around the average of the range from 7 to 1030) and a realistic PWR primary I-131 concentration level of 0.05 μCi/g (around the average of the range from 0.01 to 0.1 μCi/g), the spike level is (500)(0.05 μCi/g) = 25 μCi/g. The proposed LCO gives (1 μCi/g)(20) = 20 μCi/g for a spike occurring in a PWR operating at the maximum allowable equilibrium level. The corresponding total expected dose for coincident iodine spiking and non-core-melt SGTR is (0.0143 man-rem/py)(20 μCi/g)/(25 μCi/g) = 0.0114 man-rem/py. This is taken as the adjusted-case, affected public risk for a PWR.

For BWRs, a similar calculation using a spiking factor of 500 and a typical measured concentration level of 0.01 μCi/g results in a spike level of (500)(0.01 μCi/g) = 5 μCi/g. The proposed LCO gives a spike level of (0.2 μCi/g)(20) = 4 μCi/g. The corresponding total expected dose for coincident iodine spiking and a non-core-melt small-break LOCA is (0.0185 man-rem/py)(4 μCi/g)/(5 μCi/g) = 0.0148 man-rem/py. This is taken as the adjusted-case, affected public risk for a BWR.

TABLE 2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Iodine Spike (B-65)

2. Affected Plants (N):

All Plants		N
PWRs:	Operating	47
	Planned	43
	Total	90
BWRs:	Operating	24
	Planned	20
	Total	44

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

		$\bar{T}$ (yr)
PWRs:	Operating	27.7
	Planned	30
	Total	29.8
BWRs:	Operating	25.2
	Planned	30
	Total	27.4

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

Since core-melt frequency is unaffected, U = 0.

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

It is estimated that equipment adjustment would be made for the additional sampling requirements in the affected plants. One man-week (40 man-h) for adjustment is estimated. This applies to both PWRs and BWRs, operating plants only.

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

It is assumed that radiation fields of 25 mR/h exist for the equipment adjustment.

D = (40 man-h/plant)(0.025 rem/h) = 1 man-rem/plant

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

ND = (71 plants)(1.0 man-rem/plant) = 71 man-rem

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

Additional sampling of the primary coolant after each power transient is needed to obtain detailed information on the iodine spike. As estimated earlier in the text, the spike frequencies are 0.52/py at PWRs and 0.14/py at BWRs. Eight extra samplings are required per spiking incident. It is assumed that the sampling and analysis require 2 man-h/sample, 50% each for the sampling (in a radiation zone) and the analysis (non-radiation zone labor). Thus, the labor required in radiation zones becomes

PWR: (8 samples/spike)(1 man-h/sample)(0.52 spike/py)
= 4.16 man-h/py

BWR: (8 samples/spike)(1 man-h/sample)(0.14 spike/py)
= 1.12 man-h/py

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

$$(D_o)_{PWR} = (4.16 \text{ man-h/py})(0.025 \text{ R/h}) = 0.104 \text{ man-rem/py}$$

$$(D_o)_{BWR} = (1.12 \text{ man-h/py})(0.025 \text{ R/h}) = 0.0280 \text{ man-rem/py}$$

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

$$N\bar{T}D_o = 90(28.8 \text{ yr})(0.104 \text{ man-rem/py}) + 44(27.4 \text{ yr})(0.0280 \text{ man-rm/py}) = 303 \text{ man-rem}$$

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
370	1100	120

3.0 SAFETY ISSUE COSTS

It is assumed that the costs to industry are due to the increased frequency of iodine sampling after each transient. No new equipment for sampling and analysis is required. The base-case equipment requirements are set in Safety Issue 74 ("Iodine Coolant Activity Limiting Conditions for Operation") and are not considered in this issue.

Efforts required by the NRC to develop and confirm a model for the iodine-spiking phenomenon could be significant, because very little is known about the physics associated with the phenomenon. Two man-years are estimated for the SIR development. Contractor support is estimated at $300,000.

Minimal NRC staff labor is anticipated for review of SIR operation and maintenance. A value of 0.1 man-week/py is used for this analysis.

Analysis results are summarized in Table 3.

TABLE 3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Iodine Spike (B-65)

2. Affected Plants (N):

All Plants		N
PWRs:	Operating	47
	Planned	43
	Total	90
BWRs:	Operating	24
	Planned	20
	Total	44

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

		$\bar{T}$ (yr)
PWRs:	Operating	27.7
	Planned	30
	Total	29.8
BWRs:	Operating	25.2
	Planned	30
	Total	27.4

TABLE 3. (cont'd.)

Industry Costs (Steps 4 through 12)

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

Since core-melt frequency is unaffected, $H = 0$

6. Per-Plant Industry Resources for SIR Implementation:

Four man-weeks/plant are assumed for modification of surveillance equipment (one man-week of which is assumed to be in a radiation zone). No additional sampling and analysis equipment is needed. This applies only to operating plants since it is assumed that modified equipment will be incorporated in the initial design of planned plants.

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

I = (4 man-wk/plant)($2270/man-wk) = $9080/plant

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

NI = ($9080/plant)(71 plants) = $6.45E+05

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

Based on Step 9 of Table 2, the labor is estimated to be as follows:

PWR: (8 samples/spike)(2 man-h/sample)(0.52 spike/py)
= 8.32 man-h/py

BWR: (8 samples/spike)(2 man-h/sample)(0.14 spike/py)
= 2.24 man-h/py

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

$(I_o)_{PWR}$ = (8.32 man-h/py)(1 man-wk)/40 man-h)($2270/man-wk)
= $472/py

$(I_o)_{BWR}$ = (2.24 man-h/py)(1 man-wk/40 man-h)($2270/man-wk)
= $127/py

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

$N\bar{T}I_o$ = 90(28.8 yr)($472/py) + 44(27.4 yr)($127/py)
= $1.38E+06

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$2.0E+06	$2.8E+06	$1.3E+06

NRC Costs (Steps 13 through 21)

13. NRC Resources for SIR Development:

NRC Staff Labor = 2 man-yr
Contractor Support (cost estimated directly in next step)

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

$$\begin{aligned} \text{Labor} &= (2 \text{ man-yr})(\$1.0E{+}05/\text{man-yr}) = \$2.0E{+}05 \\ \text{Contractor Support} &= \underline{\$3.0E{+}05} \\ C_D &= \$5.0E{+}05 \end{aligned}$$

15-17. Steps Related to NRC Cost for Support of SIR Implementation:

The SIR asks for new spike model development. No need for site-specific support is seen. Thus, C = 0.

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

0.1 man-wk/py

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

C_o = (0.1 man-wk/py)($2270/man-wk) = $227/py

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

$$N\bar{T}C_o = 90(28.8 \text{ yr})(\$227/\text{py}) + 44(27.7 \text{ yr})(\$227/\text{py}) = \$8.62E{+}05$$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$1.4E+06	$1.9E+06	$8.6E+05

REFERENCES (for Issue B-65)

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OFFSITE

W. B. Andrews
Science Application International Corporation
101 Convention Center Drive
Suite 407
Las Vegas, NV 89109

R. H. V. Gallucci
C. E. Power Systems
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, CT 06095-0500

ONSITE

49 Pacific Northwest Laboratory

A. S. Tabatabai (25)
B. A. Fecht
W. E. Bickford
S. H. Bian
P. M. Daling
E. J. Eschbach
C. H. Allen
G. J. Konzek
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UNC
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13. ABSTRACT (200 words or less)

This is the fifth in a series of reports to document the use of a methodology developed by the Pacific Northwest Laboratory to calculate, for prioritization purposes, the risk, dose and cost impacts of implementing resolutions to reactor safety issues (NUREG/CR-2800, Andrews et al. 1983). This report contains results of issue-specific analyses for 23 issues. Each issue was considered within the constraints of available information as of winter 1986, and two staff-weeks of labor. The results are referenced, as one consideration in setting priorities for reactor safety issues, in NUREG-0933, A Prioritization of Generic Safety Issues.

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