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Regulatory Analysis for Generic Issue 23: Reactor Coolant Pump Seal Failure

Draft Report for Comment

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

S. K. Shaukat, J. E. Jackson, D. F. Thatcher

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S. K. Shaukat, J. E. Jackson, D. F. Thatcher

Division of Safety Issue Resolution
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ABSTRACT

This report presents the regulatory/backfit analysis for Generic Issue 23 (GI-23), "Reactor Coolant Pump Seal Failure." A backfit analysis in accordance with 10 CFR 50.109 is presented in Appendix E. The proposed resolution includes quality assurance provisions for reactor coolant pump seals, instrumentation and procedures for monitoring seal performance, and provisions for seal cooling during off-normal plant conditions involving loss of all seal cooling such as station blackout. Research, technical data, and other analyses supporting the resolution of this issue are summarized in the technical findings report (NUREG/CR-4948) and cost/benefit report (NUREG/CR-5167).

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EXECUTIVE SUMMARY

This report provides a regulatory analysis for the Nuclear Regulatory Commission's (NRC's) proposed resolution of Generic Issue 23 (GI-23), "Reactor Coolant Pump Seal Failure." Technical findings related to the resolution of this generic issue are presented in NUREG/CR-4948. Those technical findings summarize potential safety concerns of seal failures in pressurized water reactors (PWRs) only. Boiling water reactors (BWRs) are not included as they exhibit significantly lower leak rates from seal failures because of lower system pressures and the presence of isolation valves in the reactor recirculation loops. BWRs also typically use a steam-driven reactor core isolation cooling (RCIC) system and larger high-pressure coolant injection (HPCI) systems and feedwater makeup capabilities. A cost/benefit analysis and other information supporting the proposed resolution is presented in NUREG/CR-5167.

Reactor coolant pump (RCP) seal failure refers to the loss of integrity of the primary coolant system pressure boundary through failure of the RCP seals designed to minimize reactor coolant leakage along the RCP shafts. Since leak rates resulting from the loss of RCP seal integrity may exceed the capability of the reactor coolant makeup system, the failure of RCP seals is a potential initiating event for small-break loss-of-coolant accidents (LOCAs).

Analysis and evaluation of GI-23 is separated into failures during normal operation and during off-normal conditions involving loss of seal cooling. The primary objective of the proposed resolution of GI-23 is to reduce the risk of severe accidents associated with RCP seal failure by reducing the probability of seal failure, thus making RCP seal failure a relatively small contributor to total core damage frequency. The following items are included in the proposed resolution:

1. Treat the RCP seal assembly as an item performing a safety-related function similar to other components of the reactor coolant pressure boundary, applying quality assurance requirements consistent with Appendix B to 10 CFR 50 and applicable General Design Criteria of Appendix A,

2. Provide RCP-manufacturer-recommended instrumentation and instructions for monitoring RCP seal performance and detecting incipient RCP seal failures, and

3. Provide RCP seal cooling during off-normal plant conditions involving loss of all seal cooling such as station blackout (SBO).

Items 1 and 2 are directed toward reducing RCP seal failures during normal operation whereas Item 3 serves to eliminate the relatively high seal failure probabilities predicted during off-normal conditions. Implementation of all three items is calculated to result in a 60% reduction in the probability of seal failure. That is, core damage frequencies due to RCP seal failures are estimated to be reduced from 2.79 E-05 to 1.12 E-05 per reactor-year for PWRs.

Since both Item 1 and Item 2 address RCP seal failures during normal operating conditions, benefits realized from implementing one item may overlap benefits from the other item. Cost/benefit analyses of Items 1 and 2 therefore consider both independent and combined implementation. Consequence assumptions from NUREG/CR-2800 were used to determine the benefits gained through reduction in seal failure during normal operating conditions. The best-estimate cost/benefit ratios ($/person-rem) for Item 1, Item 2, and Item 1 & 2 combined are -392, 394, and 6 respectively. A negative number indicates that the total costs resulted in a net savings.

For Item 3, the core damage frequency due to RCP seal leakage under SBO was determined using a pump seal failure model developed by Atomic Energy of Canada Limited in conjunction with SBO frequencies and durations from NUREG-1032. Separate calculations (Ref. 3) were done by Brookhaven National Laboratory for benefits gained through the provision of seal cooling during other off-normal conditions (such as loss of component cooling water (CCW) and loss of service water (SW) without station blackout). Consequence assumptions from NUREG-1109 and NUREG-1150 were used to determine the benefits (values) gained through reduction of RCP seal failures during off-normal conditions. The best-estimate cost/benefit ratio ($/person-rem) for Item 3 is 958.

Comparison of the best-estimate cost/benefit ratios for all the three items against a guideline cost/benefit ratio of $1000/person-rem shows that all the items are cost effective. It should be noted that, if the $1000/person-rem given in Ref. 6 in 1983 dollars is corrected for inflation at the rate of 5% per year, a guideline of $1400/person-rem would result at the end of 1990. Comparison with this number would make all the items more cost effective. The regulatory analysis used cost/benefit calculations as well as other factors to develop a proposed resolution for GI-23.

The analysis and determination that the proposed resolution concerning RCP seal failure complies with the backfit rule of 10 CFR 50.109 are presented in Appendix E.

The evaluation of Item 3 involved a seal model that estimated the probability of failure of individual RCP seal stages as a function of time and the resultant probability of core uncovery as a function of time. This model is complicated, and there is a large uncertainty in the results. The cost/benefit analysis was based on a system that used an air-cooled diesel generator to power existing components to provide RCP seal cooling and other plant functions (e.g., reactor coolant makeup or high-pressure injection) during off-normal conditions. Such a single-train backup cooling system is estimated to have a 95% probability of preventing RCP seal failure during off-normal conditions. This engineering approach avoids dealing with the uncertainties associated with the complicated seal model. Other less-expensive generically applicable backup means of cooling the RCP seals may also be possible. One such approach that uses the fire water system is described in Appendix C; its cost is estimated to be about one-third of that of the system used to estimate the cost/benefit ratio for item 3.

Other means for addressing the RCP performance during off-normal conditions were considered but not recommended. For example, one approach would be to test seals to verify that excessive leakage does not occur when the seals are not cooled. However, the tests performed to date have had various inadequacies. Furthermore, a large number of successful repetitive tests would be needed to statistically demonstrate a reliability comparable to that of a backup cooling system.

Information comparing core damage frequencies from GI-23 studies with those of the three PWRs of NUREG-1150 is included in Appendix D. Evaluation of the potential for core damage due to LOCAs induced by pump seal failure in BWRs is presented in Appendix B. Appendix A provides a summary of pump seal failure experience data.

PREFACE

The NRC staff has prepared this draft Regulatory Analysis to evaluate a proposed resolution that could serve as an acceptable approach for addressing the concerns of Generic Issue GI-23, "Reactor Coolant Pump Seal Failure."

This draft regulatory analysis is being issued for public comment to obtain additional information prior to making the final decision on the resolution of GI-23.

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ACRONYMS

AECL	Atomic Energy of Canada Limited
ASME	American Society of Mechanical Engineers
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CCW	Component Cooling Water
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CVCS	Chemical and Volume Control System
ECCS	Emergency Core Cooling System
EEDB	Energy Economic Data Base
ESW	Essential Service Water
GDC	General Design Criteria
GI	Generic Issue
HPCI	High Pressure Coolant Injection
HPI	High Pressure Injection
IPE	Individual Plant Examination
LOCA	Loss-of-Coolant Accident
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PCV	Pressure Control Valve
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
QA	Quality Assurance
RAB	Reactor Auxiliary Building
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
SBO	Station Blackout
TCV	Temperature Control Valve
TMI	Three Mile Island
USI	Unresolved Safety Issue

1. STATEMENT OF THE PROBLEM

"Reactor coolant pump seal failure" refers to degradation of the seals that limit primary coolant leakage along the shafts of the reactor coolant pumps (RCPs). Generic Issue 23 (GI-23) addresses the consequences of such seal failures in the RCPs of pressurized water reactors (PWRs). Although operating experience suggests that the number of pump seal failures experienced by PWRs and boiling water reactors (BWRs) is roughly the same (Ref. 1), the technical investigations carried out for GI-23 primarily consider the concerns of seal failures in PWR plants. Appendix A provides a summary of pump seal failure experience data. BWR plants exhibit significantly lower leak rates from seal failures than do PWR plants, primarily because of the lower system pressure in BWRs. The use of reactor core isolation cooling (RCIC) systems, larger high-pressure coolant injection (HPCI) systems and feedwater makeup capabilities, and the presence of isolation valves in the reactor recirculation loops in BWRs reduces the potential for significant leak rates as a result of pump seal failures. With past seal failure data (Refs. 1, 2, 3, and 4) for PWRs demonstrating relatively high seal failure rates and the possibility of resulting leak rates well into the range of small-break loss-of-coolant accidents (LOCAs), the consequences of seal failures in PWRs are potentially significant. The major portion of this report therefore deals with PWRs only. However, a discussion of the potential for core damage in BWRs due to LOCAs induced by pump seal failure is included in Appendix B.

RCP primary and secondary seals limit the leakage of reactor coolant along the pump shaft and thereby into the containment. These seals, forming part of the reactor coolant pressure boundary, require cooling[1] during normal operation, even while the reactor is in hot standby or hot shutdown. Without such cooling, both primary and secondary seals are susceptible to increased leakage once temperatures exceed design limits[2]. A detailed discussion of RCP seal design and operation is provided in Section 2 of Reference 5.

Analysis of reported seal failures (References 1, 2, and 4) indicates that leakage past pump seals may be of sufficient magnitude to constitute a small-break LOCA. To resolve this issue, the known root causes of seal failure were studied, and from this

[1]Seal cooling is normally accomplished through one or both of the following design features: (1) seal injection and (2) a thermal barrier heat exchanger. For Westinghouse plants, the seal injection water is supplied from the chemical and volume control system (CVCS) with a portion flowing through the pump seals and the remainder flowing past the thermal barrier heat exchanger and into the reactor coolant system. The thermal barrier heat exchanger reduces the temperature of any reactor coolant leakage along the RCP pump shaft upstream of the RCP seals. This heat exchanger is cooled by the CCW system.

[2]RCP seals are generally designed to operate at less than 160°F. The normal reactor coolant operating temperature is approximately 550°F.

knowledge, six alternatives were formulated to reduce the probability of seal failure. These alternatives were then subjected to an evaluation to determine their relative merits and to select the proposed resolution. Three alternatives were rejected and are discussed in Section 3. The three remaining alternatives, discussed in Section 2, were subjected to cost/benefit analyses in accordance with Reference 6 and are included as the proposed resolution.

The analysis and evaluation of GI-23 were separated into initiating failures during normal operation and consequential failures as a result of loss of cooling water during off-normal conditions. For failures associated with normal operation, the primary concern is whether or not sufficient reactor coolant inventory to cool the core can be maintained until the system is brought to a safe shutdown condition. This can be accomplished using normal reactor coolant makeup unless the leak rate from the failed seal(s) exceeds the capacity of the reactor coolant makeup system (resulting in a small-break LOCA), in which case, emergency core cooling system (ECCS) operation may be required. Failure mechanisms associated with normal operation include those involving operator error and normal wear and those resulting from improper maintenance (e.g., damage during handling and installation, improper installation or adjustment, or the introduction of foreign materials damaging to the seal faces).

For off-normal conditions, particularly station blackout with associated loss of seal injection cooling or loss of CCW/SW or both, the major concerns involve seal failures due to adverse temperature effects on secondary seal elastomer materials as well as performance instabilities at the primary seal faces related to coolant flashing and two-phase flow. Current RCP secondary seal materials are susceptible to accelerated degradation when seal cooling is lost and seal temperatures approach normal reactor coolant system operating temperatures. Seal failures resulting in a small-break LOCA under station blackout or loss of CCW conditions could lead to core damage due to the concurrent loss of normal and emergency reactor coolant makeup capabilities.

In the worst case, the potential exists under two-phase flow conditions for the hydraulic balance on the primary seal faces to be upset, causing the faces to fail in the open position, i.e., the position of maximum leakage. Seal failures of this severity could result in leak rates that exceed the capability of the makeup system. Primary system leaks exceeding normal plant makeup capability are considered LOCAs that can lead to core damage unless the coolant inventory of the reactor coolant system is maintained using the ECCS. If station blackout conditions continue, it will not be possible to maintain coolant inventory because of the inability to power the motor-driven charging and ECCS pumps. Extended loss of CCW/SW could have a similar effect because of the dependency of these pumps on CCW/SW for cooling.

Based on data collected from seal failures that have occurred during normal operation at commercial nuclear plants (Ref. 1), seal performance has been such that many seals have failed or required maintenance before regularly scheduled outages. Reference 1 indicates that about 60% of recorded PWR RCP seal failures can be attributed to one or more of the following: improper maintenance, harsh environment, transients, contamination, corrosion, operator error, defective parts, or lack of proper instrumentation while the remaining 40% are attributed to other causes. WASH-1400 (Ref. 7) reported that breaks in the primary coolant system pressure boundary with an equivalent diameter of between 0.5 and 2.0 inches represent a significant contribution to total core damage probability. WASH-1400 assumed a small pipe break frequency of 10^{-3} events/year. Examination of actual plant data has shown (Ref. 1) that comparable flow rates representative of these equivalent break diameters have resulted from RCP seal failures at a frequency of approximately 10^{-2} events/year for each plant.

Results of a more recent PWR data survey (a 1988 study by Brookhaven National laboratory) indicate that seal failure rates may have decreased appreciably since 1984. However this survey was limited to data reported in the Nuclear Plant Reliability Data System from January 1984 to October 1987. The resulting improvements were not uniform with respect to NSSS vendor/pump manufacturer. This nonuniformity indicates that some but not all licensees have implemented portions of the improvements being proposed by the GI-23 resolution to reduce seal failures under normal operation.

Probabilistic risk assessment (PRA) analyses indicate that the overall probability of core damage due to postulated small breaks could be dominated by events such as RCP seal failures (Ref. 1). Reference 1 cites the impact of mechanical- and maintenance-induced failures of RCP seals on plant safety. It is concluded in Reference 1 that, for some PWRs, the annual core damage frequency due to mechanical- and maintenance-induced RCP seal failures may be as high as 19% of the total core damage frequency for all causes.

Table 1-1 shows that the estimated core damage frequencies (CDF) per reactor-year induced by RCP seal failure before implementing GI-23 represent significant public risk. The CDF value for normal operating conditions representative of PWRs was derived from Reference 1. CDF for off-normal conditions is composed of two parts, station blackout (CDF_{SBO}) and loss of CCW independent of station blackout (CDF_{CCW}). The value for off-normal CDF_{SBO} is taken from an AECL model (detailed in Appendix A of Reference 8). The value of CDF_{CCW} is taken from a sensitivity study (Ref. 3) performed on one plant. When compared to PRAs of several plants, this value is considered to be generally low. Use of a more representative value would increase the benefit obtained from the proposed resolution.

Table 1-1

Core Damage Frequency Induced by RCP Seal Failure

Operating Conditions	CDF/reactor-year induced by RCP seal failure
Normal	1.63 E-05
Off-Normal	1.16 E-05*
Total	2.79 E-05

*For off-normal conditions, $CDF = CDF_{SBO} + CDF_{CCW}$

where: $CDF_{SBO} = 5.6$ E-06
$CDF_{CCW} = 6$ E-06

Note: Consideration of additional CDF induced by service water failure would tend to increase the benefit. This contribution is being evaluated as part of GI-130.

2. OBJECTIVES OF THE PROPOSED RESOLUTION

The proposed resolution for GI-23 applies to PWRs only. The general objective of the proposed resolution is to reduce the risk of accidents associated with RCP seal failure by making RCP seal failure a relatively small contributor to total core damage frequency. Six alternatives directed at the known root causes of RCP seal failures during both normal operation and abnormal conditions were investigated for the resolution of GI-23. Three alternatives were rejected as discussed in Section 3, and the remaining three are included as the proposed resolution for this generic issue. These resolution items and the specific improvements addressed by each are outlined in Table 2-1.

Table 2-1
Resolution Items and Improvements

Item	Action	Improvements
1	Treat the RCP seal assembly as an item performing a safety-related function similar to other components of the reactor coolant pressure boundary, applying quality assurance requirements consistent with Appendix B to 10 CFR 50 and applicable General Design Criteria of Appendix A.	. Quality control over seal materials and fabrication . Quality control over installation and maintenance through the use of process specifications . Quality control over plant operations through the use of operating procedures designed to avoid damage during pump startup and shutdown
2	Provide RCP-manufacturer-recommended instrumentation and instructions for monitoring RCP seal performance and detecting incipient RCP seal failures.	. Improved monitoring capability identifies degraded seal performance early enough to take corrective action to mitigate seal failures
3	Provide RCP seal cooling during off-normal plant conditions involving loss of all seal cooling such as station blackout.	. Maintain RCP seal temperatures within system design conditions . Avoid two-phase flow through the seals

Items 1 and 2 are directed toward reducing RCP seal failures and optimizing seal performance during normal operation whereas Item 3 serves to eliminate the high seal failure probabilities predicted (Ref. 8) during off-normal conditions involving loss of seal cooling. The proposed resolution is based on current RCP seal technology and performance characteristics.

2.1 Item 1

Under the provisions of Item 1, the RCP seal assembly would be treated as an item performing a safety-related function similar to other components of the reactor coolant pressure boundary as defined in 10 CFR 50.2, and the requirements of General Design Criterion (GDC) 30 of Appendix A to 10 CFR 50 will be implemented in full. GDC 30 states:

> "...Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage."

Historically, GDC 30 has not been fully applied to RCP seals. Current design requirements for the reactor coolant pressure boundary are contained in Section III of the ASME code as endorsed by 10 CFR 50.55a. Portions of the RCP seals are made of nonmetallic materials and thus the RCP assembly was not considered to be covered by the ASME standards. Under the provisions of Item 1, the highest quality standards available would be applied to the RCP seal assembly.

The adoption of Item 1 will mean a tighter system of quality control in the following areas:

1. Increased control over materials and fabrication methods used for the manufacture of RCP seals to improve the seal performance.

2. Increased control over the installation and maintenance of the RCP seals to help maintain seal integrity. This control will be realized through the use of detailed procedures for RCP seal installation and maintenance.

3. Increased procedural control over the operation of the RCPs to be consistent with manufacturer's specifications, particularly during startup and shutdown when the seals are most susceptible to damage.

2.2 Item 2

Item 2 refers to instrumentation recommended by the RCP manufacturer. Installation and use of this instrumentation in plants that do not already have it will help detect incipient RCP seal failures in time to take corrective action to mitigate the effects of the leakage. This forewarning will be afforded through additional instrumentation, alarms, and monitoring procedures providing the operator with more accurate information concerning leakage through the RCP seals.

2.3 Item 3

Unlike Items 1 and 2, which deal with reducing the probability of RCP seal failure during normal operating conditions, Item 3 addresses the inherent coupling of RCP seal failure with off-normal conditions (such as loss of CCW/SW or station blackout) during which thermal barrier cooling and/or injection flow to the seals are lost. As noted in Reference 9, the probability of seal failure increases rapidly as the seal leakage temperature increases beyond prescribed operating limits. This temperature increase is an expected consequence whenever normal seal cooling mechanisms have been rendered ineffective. Without seal cooling, leakage increases and the seals have an increased probability of failure. Seal failure models (Ref. 8) used in the cost/benefit analysis of Item 3 are based on independent testing and evaluation of certain design characteristics and materials of the Westinghouse RCP seal design by Atomic Energy of Canada Limited (AECL). This design was selected because (1) it represents the majority (53 of 76) of commercial PWRs and (2) most seal testing and analysis have involved Westinghouse pump seals. Because basic design similarities exist in all PWR RCP seals, however, the results of the Item 3 cost/benefit analysis are generally applicable to all commercial PWRs in the United States. (This is discussed further in Reference 8.)

To preclude temperature-related failures of the RCP seals, Item 3 includes provisions for RCP seal cooling during off-normal conditions. Cost/benefit analysis assumes meeting the intent of Item 3 by (1) installing an "alternate ac source" (as defined in 10 CFR 50.2) to provide at least one mode of seal cooling (seal injection or thermal barrier cooling) to the RCP seals and (2) performing plant modifications to allow backup cooling of the makeup pump from an existing plant water system for those plants that have a potential vulnerability to loss of seal cooling from conditions other than SBO (loss of CCW/SW). Further details are provided in Appendix C of Reference 8.

The installed equipment must be capable of maintaining seal temperatures within specified design limits for the duration of the off-normal conditions.

3. EVALUATION

3.1 Proposed Resolution

Three items are included in the proposed resolution discussed in Section 2. Each item was evaluated based on benefits and present worth of costs, impacts on other requirements, and constraints. The attributes used to calculate cost/benefit ratios are public health benefit, all industry costs (for 76 PWRs), all NRC costs, and averted onsite property damage and occupational exposure. A summary of cost and benefit is presented in Table 3-1. More detailed cost/benefit information for each item in the proposed resolution, including high, low, and best estimates, is presented in Reference 8.

Table 3-1

Summary of Cost ($\$10^6$) and Benefit (for 76 PWRs)

DESCRIPTION	ITEM 1	ITEM 2	ITEMS 1&2 COMBINED	ITEM 3
INDUSTRY COST:				
IMPLEMENTATION	$ 3.80	$20.27	$24.07	$64.46
OPERATION	-$13.17*	-$ 6.47*	-$22.35*	$ 1.01
NRC COST:				
DEVELOPMENT	$ 0.37	$ 0.37	$ 0.74	$ 0.37
IMPLEMENTATION	$ 0.037	$ 0.037	$ 0.074	$ 0.037
OPERATION	$ 0.335	$ 0.335	$ 0.67	$ 0.335
AVERTED PROPERTY COST (onsite damage)	-$ 1.49	-$ 1.91	-$ 2.90	-$ 5.60
PUBLIC HEALTH BENEFIT (person-rem)	26,201	33,479	50,946	62,814
OCCUPATIONAL EXPOSURE (Reduction in person-rem; negative means increase)				
OPERATIONAL	-532	-1,569	179	NEGLIGIBLE
ACCIDENTAL	117	150	228	440
COST/BENEFIT** ($/person-rem)	-392	394	6	958

* A negative industry cost represents savings achieved by fewer seal replacements per regularly scheduled refueling outages.

$$** \text{ Cost/Benefit} = \frac{\text{(industry cost + NRC cost + onsite property cost)}}{\text{(benefit + occupational exposure reduction)}}$$

3.1.1 Analysis of Items 1 and 2

Item 1 is estimated to reduce the frequency of seal failure/reactor-year by about 18 percent from 1.28E-02 to 1.05E-02. This would result in a reduction of CDF/reactor-year from 1.63E-05 to 1.34E-05. Similarly, Item 2 is estimated to reduce the frequency of seal failure/reactor-year by about 23 percent from 1.28E-02 to 1.00E-02. This would result in a reduction of CDF/reactor-year from 1.63E-05 to 1.26E-05.

Since both Item 1 and Item 2 address RCP seal failures during normal operating conditions, benefits realized from implementing one item may overlap benefits from the other item. Therefore, the effect of implementing Items 1 and 2 combined is also presented. Implementing Items 1 and 2 combined is estimated to reduce the frequency of seal failure/reactor-year by about 35 percent from 1.28E-02 to 8.3E-03. This would result in a reduction of CDF/reactor-year from 1.63E-05 to 1.06E-05.

Cost/benefit analyses of Items 1 and 2 also consider both independent and combined implementation. Consequence assumptions from NUREG/CR-2800 were used to determine the benefits gained through reduction in seal failure during normal operating conditions. The best-estimate cost/benefit ratios ($/person-rem) for Item 1, Item 2, and Items 1 & 2 combined are -392, 394, and 6 respectively. A negative number indicates that the total costs resulted in a net savings.

The best-estimate cost/benefit ratios for Items 1 and 2 are favorable based on the $1,000/person-rem decision guideline of Reference 6. The increased costs of quality assurance (Item 1) or instrumentation (Item 2) are more than offset by the cost decrease due to fewer seal replacements per regularly scheduled refueling outages. Because all of these savings were assumed to occur during normal outages, no additional cost savings associated with replacement power were included. However, if Items 1 and 2 resulted in avoiding some forced outages due to seal failure, added savings could be included and Items 1 and 2 would be more cost effective.

3.1.2 Analysis of Item 3

The effect of implementing Item 3 on frequency of seal failure is too complex to describe as a simple number. A brief summary of the complex event tree used to evaluate off-normal conditions is presented in Section 4.1.2. The details including RCP seal failure rate estimates are provided in Reference 5 and Appendix B of Reference 8. Item 3 is estimated to reduce the CDF/reactor-year by about 95 percent from 1.16E-05 to 6.00E-07. The best-estimate cost/benefit ratio ($/person-rem) for Item 3 is 958.

The cost/benefit ratio for Item 3 includes the same factors as does that for Items 1 and 2. Again, negative numbers indicate that the total costs resulted in a net savings. The best-estimate cost/benefit ratio for Item 3 is favorable based on the $1,000/person-rem decision guideline of Reference 6. There is also considerable uncertainty in the cost/benefit analysis of Item 3 due mainly to plant-specific uncertainties involving station blackout probabilities and release consequences. The uncertainties in seal failure model constitute only about 25% of the total uncertainties. Reference 8 includes more information on uncertainties. The justification for including Item 3 in the proposed resolution includes factors in addition to cost/benefit considerations. This is further discussed in Section 4. Section 4.2 and Appendix C describe a lower cost alternative.

3.1.3 Core Damage Frequency (CDF) Induced By RCP Seal Failure

Implementation of the proposed items will cause a reduction in seal failure thus reducing the CDF per reactor-year induced by RCP seal failure. The total CDF due to normal and off-normal conditions would be reduced by 60% if all three proposed items are implemented. The results of implementing these items are shown in Table 3-2.

Table 3-2

Core Damage Frequency Reduction due to GI-23 Implementation

ITEMS	OPERATING CONDITIONS	CDF INDUCED BY RCP SEAL FAILURE BEFORE GI-23 IMPLEMENTATION	REDUCTION IN CDF DUE TO SEAL FAIL.		CDF INDUCED BY RCP SEAL FAILURE AFTER GI-23 IMPLEMENTATION
			PERCENT	Delta CDF	
1 & 2 COMBINED	NORMAL	1.63 E-05	35%	5.70 E-06	1.06 E-05
3	OFF-NORMAL	1.16 E-05	95%	1.10 E-05	6.00 E-07
1 & 2 COMBINED AND 3	TOTAL	2.79 E-05	60%	1.67 E-05	1.12 E-05

3.2 Rejected Alternatives

Three alternatives to the proposed resolution were considered and rejected. Preliminary evaluations showed that these alternatives involve too many uncertainties or are otherwise ineffective or incomplete in resolving GI-23 relative to those items included in the proposed resolution.

3.2.1 Alternative 1

Alternative 1 involves replacement of all secondary seals with seals fabricated from improved high-temperature elastomers (equivalent to those proposed by Westinghouse in Reference 2). This material improvement relates to Item 3 of the proposed resolution in that its primary objective is to reduce the failure probability and improve the performance of the RCP seals during off-normal conditions involving loss of seal cooling. This alternative also involves a partial implementation of Item 1 in that limited quality assurance (QA) requirements are applied to secondary seal materials whereas the full QA requirements are applied to RCP seal assembly in Item 1 of the proposed resolution. Inconsistency in the application of QA requirements is the primary cause for rejecting this alternative. All components of the RCP seals that form part of the reactor coolant pressure boundary should be subject to full QA requirements not just secondary seal materials. Relative to Item 3, the improved secondary seal materials are beneficial only under high-temperature conditions. Therefore, if Item 3 is implemented and seal cooling is either maintained or not required for acceptable seal performance, this material improvement is unnecessary. This alternative, while providing improved high-temperature performance of the secondary seals, does not preclude seal failures (especially face seal failures) under off-normal conditions involving a loss of RCP seal cooling and two-phase flow through the seals. Analysis of the improved secondary seals proposed by Westinghouse using the seal failure model developed by AECL showed that the probability of seal failure is not fully eliminated under station blackout conditions because there are other seal failure modes, e.g., seal faces popping open. Reference 8 provides a description of the complex event tree used to develop estimates of the probability of seal failure and core uncovery as a function of time. This tree shows that there are a number of RCP seal failure paths that will continue to exist even with "perfect" secondary seals.

3.2.2 Alternative 2

Alternative 2 involves the design and installation of an emergency backup seal on each RCP to preclude excessive seal leakage during off-normal conditions. The design of this seal would take advantage of and depend on fixed shaft position (no rotation) and limit RCP seal leakage to less than 3 gallons per minute (gpm). Activation of the emergency seals in the event of off-normal conditions (such as station blackout or loss of CCW) would eliminate the need for continued seal injection or thermal barrier cooling as proposed in Item 3. This alternative is directed at eliminating the risks associated with high seal-failure probabilities predicted whenever seal cooling is lost. Reference 8 contains a discussion of temperature-related failure probabilities.

Although this alternative would be most effective in reducing the consequences of RCP seal failure during both normal operation and station blackout conditions, it was rejected because of the high degree of uncertainty associated with development and installation costs. The scope and extent of modifications to existing RCP shafts and seal assemblies required to accommodate the emergency seal are indeterminate at this time, and the potential problems associated with inadvertent actuation need to be examined. The development of such a seal should be encouraged, and any proposed designs should be given serious consideration.

3.2.3 Alternative 3

Alternative 3 involves full-scale testing of RCP seals to verify their acceptable performance under worst-case plant conditions for loss of all seal cooling such as SBO or loss of CCW. Tests developed to meet certain prescribed pressure, temperature, and other conditions with loss of seal cooling may provide some judgment on seal performance under off-normal conditions. However, the staff believes that the limited number of seal tests conducted by the industry has not adequately represented all the conditions that can occur during the loss-of-seal-cooling events and hence did not adequately address the safety concerns regarding RCP seals. The following arguments support this conclusion:

1. <u>50-hour SBO test of the St. Lucie production seal cartridge (Byron-Jackson seal)</u>

   Although leakage remained within normal limits, the vapor seal rotating ring cracked, O-rings were permanently compressed, and U-cups were permanently hardened with extrusion. Test conditions did not allow for shaft motion. Temperature and pressure did not allow the saturation at the seal inlet that would be seen under actual SBO conditions. The seal cartridge was new (unused) and the test was of the cartridge only without a pump and did not follow actual SBO reactor coolant conditions.

2. <u>30-minute loss-of-seal-cooling test on San Onofre RCP while the pump was operating (Byron-Jackson Seal)</u>

   Pressure fluctuations were observed for the second- and third-stage seals. Vapor seal leakage indicated seal cartridge degradation, O-rings were permanently deformed, and U-cups were extruded up to 1/16 inch axially with seal faces showing signs of wear and heat checking. Although controlled leakage of 2 gpm and vapor seal leakage of 0.5 gpm were the maximum recorded, results from this type of test should not be extrapolated to longer times or to nonrotating loss-of-seal-cooling events such as SBO. It is not clear that loss of seal cooling with the pump running is as severe as loss of cooling with a nonrotating pump, as in SBO.

3. <u>30-minute loss-of-seal-cooling test of operating boiler recirculation pump with 4½-inch-diameter shaft (Bingham International Test for San Onofre)</u>

   The test was not run with an RCP seal but with a smaller (4½-inch- diameter shaft) boiler recirculation pump. The second- and third-stage seals exhibited bistable behavior (see NUREG/CR-4821 and item 2 above).

4. <u>Secondary seal material tests (O-rings, channel seals, and U-cups)</u>

   Secondary seals currently used in RCPs failed under SBO conditions (see NUREG/CR-4077, NUREG/CR-4821, NUREG/CR-4948, and Board Notification BN-83-139 and BN-84-123).

5. SBO test of 7-inch-diameter seal assembly typical of Westinghouse pump seals used in European nuclear power plants

   There is no guarantee that, under SBO conditions, the 7-inch seal would behave like the 8-inch seal typically used in the U.S. The 7-inch seal is significantly different in design from the 8-inch seal with differences in O-rings and channel seal materials, seal ring thicknesses, mounting and support configurations, flow restriction downstream of the gap between seal rings, and the balance ratio of the second-stage seal. The test seal was in "as-new" condition when tested whereas NRC research has demonstrated a potential for a second-stage seal to pop open if the seal faces have scratches or wear marks. Modeling of important leakoff systems with orifice plates may have provided excessive flow resistance and choked the flow artificially thus limiting the leakage. The test procedure was a compromise between test objectives and the facility capabilities; therefore, the actual SBO sequence was not accurately duplicated. The test was a seal test, not a pump-seal test. Pump shaft growth could potentially drag seal faces open, yet no consideration was given to shaft movement under thermal expansion, either to introduce or to monitor it (NUREG/CR-4948, -4821, -4906P, and -4907P).

6. Multiple in-plant loss-of-seal-cooling events

   These events were of short duration (mostly hot functional testing) and of undocumented reactor coolant system conditions. They generally did not run long enough to cause hydraulic instability. Many events were of 10 minutes or less.

7. Byron Jackson 9000 seal test

   A report of the testing details has not been submitted for NRC review, and the number of plants actually using this seal model has not been identified.

Depending on the confidence level desired, between 14 and 59 successful tests would be needed to statistically demonstrate a 95 percent probability that the RCP seals will not fail upon loss of cooling, which would be comparable to the reliability of the backup cooling system. These arguments represent the basis for the staff's conclusion that this alternative is not a viable one. Moreover, a calculation of cost for this alternative is highly uncertain. This alternative was therefore rejected.

3.3 Evaluation of No Regulatory Action

This is simply to take no regulatory actions. This was used as the baseline for cost/benefit evaluations (Section 3.2.1). A no-action resolution was rejected because of the significant core damage probability associated with RCP seal LOCAs (Table 1-1).

3.4 Impacts on Other Requirements

3.4.1 USI A-44, Station Blackout

Station blackout and RCP seal failure affect each other because of the relationship between the probability of RCP seal failure and the duration of station blackout as described in Reference 2. The resolution of USI A-44 (detailed in NUREG-1109 and NUREG/CR-3226) assumes that RCP seal leakage does not exceed 20 gpm per pump during station blackout. Item 3 specifically addresses this dependency and provides a means of mitigating the effects of a station blackout on RCP seals, ensuring continued availability of seal cooling for the duration of the station blackout. Without the implementation of Item 3, there is a significant probability that leak rates substantially greater than 20 gpm (Ref. 2) will occur within the first half hour of a station blackout. (This is further discussed in Reference 8.) These higher leak rates would invalidate the 20-gpm seal leakage assumption and subsequent reactor coolant inventories upon which the NUREG-1109 resolution of USI A-44 is based. Therefore, without the implementation of Item 3, the ability of individual plants to cope with station blackouts of specified durations would need to be reevaluated. It should be noted that Reference 8 includes an analysis of the probability of core damage due to seal leakage for station blackout durations of 4 hours and longer. The analysis concluded that, for a station blackout duration of 4 hours, the probability of core damage due to RCP seal failure is 5.6E-06. The goal of the SBO rule (10 CFR 50.63) is to limit the total core damage frequency to 1E-05. The proposed actions for GI-23 are aimed at providing better assurance of seal integrity under SBO conditions so that the overall goal of the SBO rule is achieved.

3.4.2 GI-65, Component Cooling Water System Failure

The probability of core damage due to component cooling water system failures (GI-65) is included in the scope of GI-23 as it specifically relates to RCP seal LOCAs initiated by CCW failures. The effect of CCW system failure on the probability of RCP seal failure is dependent upon the duration of the unavailability of the CCW system. For some plants, a loss of RCP thermal barrier cooling for a period of 1 to 2 hours results in a significant probability for failure of the RCP seals. The number of plants that rely solely on thermal barrier cooling for the RCP seals is small, but the adoption of Item 3 will ensure that adequate cooling can be provided to RCP seals. Loss of CCW poses an additional risk even in those plants with seal injection cooling where CCW is generally required for lube oil, bearings, and environmental (pump room fan coolers) cooling of reactor coolant makeup and safety injection pumps. Exactly how long these pumps can be operated without CCW is plant specific. The adoption of Item 3, however, virtually eliminates the risk of an RCP seal LOCA coincident with a CCW system failure. Therefore, the primary concerns associated with CCW system failures previously addressed under GI-65 would be resolved by implementing Item 3 in the proposed resolution of GI-23.

3.4.3 GI-130, Essential Service Water (ESW) System Failures at Multiplant Sites

The design of the ESW system is highly plant specific with regard to equipment, crosstie capability, and operational and functional requirements. The ESW system typically supports most, if not all, of the front-line safety systems required for safe shutdown of the plant. The ESW system in most plants also provides cooling water to the CCW system, which, in turn, cools the RCP seals.

The resolution of GI-130 includes an alternative requiring an addition of ac-independent charging pumps for the RCP seals. It should be recognized, though, that GI-130 is concerned with only 7 plant sites (14 plant units) while GI-23 applies to 76 PWRs. Requiring an addition of ac-independent charging pumps for the RCP seals may prove to be an action satisfying both GI-130 and GI-23 but for only 14 plants. During the evaluation of GI-130, it was recognized that all plants appear to be somewhat susceptible to loss of ESW, and a new generic issue, GI-153, "Loss of ESW in LWRs," has been established regarding the vulnerability of ESW at all LWRs. If the resolution of GI-23 is adopted, i.e., if an alternative power source is installed to provide seal injection or thermal barrier cooling to the RCP seals or both, a large portion of the risk associated with GI-130 will have been considered. Therefore, the resolution of GI-130 must be careful not to "double count" the benefits.

3.4.4 Three Mile Island (TMI) Actions II.K.2.16 & II.K.3.25

Following the TMI accident of 1979, the Commission generically questioned the potential for a serious accident involving the failure of the reactor coolant pump seals upon a loss-of-offsite-power event. This led to the establishment of TMI Action Items II.K.2.16 and II.K.3.25 in NUREG-0737. TMI Action Items II.K.2.16 (for Babcock & Wilcox plants) and II.K.3.25 (for Combustion-Engineering, General Electric, and Westinghouse plants) require licensees to evaluate the integrity of their reactor coolant/recirculation pump seals for a period of 2 hours following a loss-of-offsite-power event. All PWR plants except Calvert Cliffs, Units 1 & 2, Haddam Neck, and Arkansas Nuclear One, Units 1 & 2 have limited the potential for seal failure by automatically loading pumps that provide seal cooling onto the emergency power bus. This design was found acceptable. The remaining five units have committed to manually load seal cooling based on plant-specific considerations.

Therefore, plants should have considered maintaining seal cooling for a loss of offsite power that is consistent with the GI-23 resolution of maintaining seal cooling during off-normal events.

3.4.5 Individual Plant Examination (IPE)

The Commission's Severe Accident Policy, 50 FR 32128 (August 8, 1985), calls for all existing plants to perform a plant-specific search for vulnerabilities. Such searches, referred to as Individual Plant Examinations (IPEs), involve a systematic plant review (which could be a PRA-type analysis). The reactor coolant pump seal loss-of-coolant accident is considered as a potential special event that a licensee may choose to address in an IPE submittal (NUREG-1335). However, the prediction of seal failure involves a complicated and controversial model, and any PRA results would be heavily dependent on such a model. An engineering solution (e.g., providing additional cooling) is therefore a preferable approach.

4. DECISION RATIONALE

The likelihood of core damage accidents from RCP seal failures has been evaluated using both engineering evaluations and cost/benefit analyses. The engineering evaluation includes the interpretation of the intent of 10 CFR 50.55a(c) and the study of major engineering considerations regarding the probability of RCP seal failure and the duration of possible loss-of-seal-cooling events. The cost/benefit studies focused on the timing and consequences of various accident sequences identifying root cause and dominant factors for core damage from RCP seal failures. These studies indicate that RCP seal failures can be a significant contributor to the overall plant risk.

4.1 Engineering Evaluation

4.1.1 Normal Operation (Items 1 & 2)

The intent of General Design Criteria in Appendix A to 10 CFR 50 is to provide a high-quality pressure boundary. Criterion 14 states that the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. 10 CFR 50.55a(c) requires that components that are part of the reactor coolant pressure boundary meet the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code. This includes all piping, valves, etc. However, Section III of the ASME Boiler and Pressure Vessel Code has specific exemptions for seal components under NB-3411.2 and NB-2121(b), not because the seals were considered unimportant, but because of jurisdictional considerations.

Many causes for RCP seal failure during normal operation have been demonstrated through actual in-plant occurrences. Failures have resulted from maintenance errors, vibration, corrosion, plant transients, contamination, abnormal pressure staging, overheating of the seal cavity, system disturbances, operator error, improper venting, lack of instrumentation, defective parts, and other causes. The resulting seal leakage has varied from very low rates up to 500 gpm. When such failures occur, there is no way to isolate the seal. Plant shutdown and depressurization would be necessary to control the leak.

Although some RCP seal failure leakage rates during normal operation are less than the makeup capability of the plant, operator error can increase the leak rates or exhaust the makeup supply. Operator error can turn a manageable event into a LOCA. Operator error was involved in the largest RCP seal failure leak rate that has occurred to date.

These causes of RCP seal leakage can be appreciably reduced by adopting Items 1 and 2. It should be noted that Item 2 is closely associated with Item 1 in that seal monitoring is an important aspect of operational quality assurance. Therefore, improved instrumentation and procedures for monitoring seal performance are inherently included in the increased quality assurance provisions of Item 1, and implementation of both items will probably result in some overlapping benefit. Therefore, both Item 1 and Item 2 in the proposed resolution are recommended for adoption, and consideration is given in the cost/benefit analysis for overlapping benefits for Items 1 and 2 combined.

4.1.2 Loss of All Seal Cooling (Item 3)

The relationship between loss of seal cooling and seal failure is not a precise engineering determination. A method of quantifying this relationship described in Reference 8 involved the development of a complex event tree. The event tree considered the likelihood of failures in individual seal stages as a function of time and displays the consequences of various failure combinations quite well. However, the quantification of this event tree requires knowledge of the time-dependent failure rate of each seal stage. Little hard data exist, so the quantification was based on the judgment of seal experts with the following considerations:

1. Hydraulic instability may occur leading to the seal faces "popping open" if there is a sufficient loss of inlet subcooling or seal stage back pressure.

2. Although experimental measurements of the frictional forces exerted by degraded O-rings and channel seals were quite low during scaled component testing, the probability of RCP seal binding failures occurring during a real loss-of-cooling event is hard to assess since the seal face plate closing forces have not been measured under these conditions.

3. Extensive testing by NRC and Westinghouse of scaled O-rings and channel seal materials indicates that improved materials are available for 0-rings that would not fail under loss of cooling conditions and that, if channel seals failed as predicted, the backup O-rings would still perform their sealing function. However, limited full-scale O-ring tests seem to support the theory that batch testing of the actual materials used in manufacturing each particular set of O-rings may be necessary in order to ensure their high-temperature characteristics. Additional uncertainty exists regarding the performance of secondary seal materials since there have been few events or tests of these components under actual loss-of-cooling conditions.

The complexity of the proposed seal failure event tree used by both AECL and Westinghouse to describe the potential failure modes of the Westinghouse seal illustrates the need for caution in relying on any decision based solely on this type of analysis. The failure event tree has one success path that leads to a leakage rate of 21 gpm/pump and 15 possible failure paths that lead to leakages ranging from 47 gpm/pump to 480 gpm/pump. Each of these paths involves the use of several assigned probabilities based on engineering judgment and speculations on the behavior of seals during a loss-of-cooling event based on the items listed above.

Another aspect of the problem is that the longer the loss of seal cooling persists, the greater is the likelihood that the seal will fail and the core will be uncovered. Moreover, once a seal loses cooling and does fail, reestablishing cooling will not correct the seal failure. Major judgments in the duration of possible loss of seal cooling are:

1. The SBO rule places plants into groupings based on the length of time they are expected to have to cope with SBO. There is some probability that an SBO will actually last longer than this time.

2. The loss of seal cooling can be due to the loss of CCW/SW, which may lead to longer times than those assumed for SBO.

Providing an engineering solution to the loss-of-cooling seal failure problem, i.e., providing an alternative source of cooling that would be available during all other postulated loss-of-cooling events such as SBO would resolve the unknowns associated with all the above considerations. This also would ensure compatibility with the resolution of USI A-44. Therefore, Item 3 is also recommended for adoption.

4.2 Cost/Benefit Considerations

The cost/benefit considerations demonstrated that the proposed resolution causes a reduction in the core damage frequency per reactor-year and a reduction in the associated risk of off-site radioactive releases due to RCP seal failures. Implementation of Items 1 and 2 combined will reduce seal failure by an estimated 35% during normal operating conditions, and implementation of Item 3 will reduce seal failure during off-normal conditions by an estimated 95%. Implementation of all three items is estimated to reduce the current core damage frequency per reactor-year due to RCP seal failure by 60%. The risk reduction to the public for Items 1 and 2 combined takes into account the overlapping benefit that results from combined implementation of Items 1 and 2. For a summary of cost and benefits for 76 PWRs, see Table 3-1. The risk reduction to the public for 76 operating PWRs over an estimated average remaining lifetime of 25 years[1] is estimated to be about 114,000 person-rem (50,946 person-rem plus 62,814 person-rem). This supports the conclusion that the proposed resolution provides a substantial increase in the overall protection of the public health and safety.

The industry implementation cost for Items 1 & 2 combined and Item 3 is estimated at $88.5 million ($1,165,000 per plant). The industry operating cost (present value) for the remaining life of the plant for the three items is estimated at a savings of $21.3 million ($281,000 per plant). The net industry cost for implementation and operation is therefore estimated to be $67.2 million (present value). However, the expected reduction in RCP seal failure as a result of the resolution is estimated to result in a substantial savings in the area of onsite property costs with $2.9 million for Items 1 and 2 combined and $5.6 million for Item 3. (More information about the cost and benefit of GI-23 is given in Reference 8.)

[1]Although not included in this regulatory analysis, anticipated license renewal (life extension) for many of these plants would further extend the remaining years of operation resulting in even greater public health benefits.

The best-estimate cost/benefit ratios ($/person-rem) for Items 1 & 2 combined is 6, while for Item 3 it is 958. Reference 6 suggests a cost/benefit of less than $1000/person-rem as a guideline for the adoption of resolution. Considering these guidelines, Items 1, 2, and 3 are justified based on cost/benefit analysis. Reference 6 stipulates that the cost/benefit guideline of $1000/person-rem is given in 1983 dollars and should be corrected for inflation. Based on a 5% per year inflation rate, the projected guideline is approximately $1400/person-rem at the end of 1990.

For an alternative look at cost/benefit considerations, a different and less-expensive fix is presented in Appendix C. It is a totally independent seal cooling arrangement, generically applicable for an SBO event, loss of CCW, or loss of SW. This fix is shown to be cost effective. In fact, accounting only for the benefit achieved from preventing seal failures during an SBO event, the cost/benefit ratio is about $700/person-rem.

4.3 Consideration of NUREG-1150 Results

A comparison of core damage frequencies from GI-23 with those from NUREG- 1150 is presented in Appendix D.

4.4 Decision

Based on the above-mentioned considerations, it is recommended that all three items in the proposed resolution should be adopted. The analysis and determination that the proposed resolution concerning RCP seal failure complies with the backfit rule of 10 CFR 50.109 are presented in Appendix E.

5. IMPLEMENTATION

5.1 Plan for Implementation

5.1.1 Existing Plants

The staff recommends that all PWR licensees evaluate their facilities in accordance with the recommendations of the pertinent regulatory guide (DG-1008, "Reactor Coolant Pump Seals") when it is issued in final form and make submittals to the NRC according to the generic letter that will be issued after final resolution of GI-23.

5.1.2 Future Plants

For future PWR plants, the Standard Review Plan will be updated by the addition of a reference to the regulatory guide (DG-1008, "Reactor Coolant Pump Seals") when it is issued in final form.

5.2 Schedule

The steps and schedule listed in Table 5-1 summarize the proposed implementation schedule for the GI-23 resolution.

Table 5-1
Implementation Schedule

Activity	Time Allowed
Licensee's submittal of summary, description, and schedule for implementation	6 months after final resolution of GI-23 is published in the Federal Register.
Implementation complete	Prior to startup after the second refueling outage following the submittal

Other schedules were considered; however, the staff believes the schedule presented in Table 5-1 is achievable without unnecessary financial burden on licensees for plant shutdown. The schedule allows reasonable time for changes in hardware to achieve the desired reduction in public health risks associated with RCP seal failures. Shorter or less flexible schedules would be unnecessarily burdensome and longer schedules would delay plant improvements.

5.3 Relationship to Ongoing Requirements

One NRC program that is currently being implemented is directly related to GI-23. That program, USI A-44, "Station Blackout," is discussed in Section 3.4.1. An RCP seal leakage assumption was included in the resolution of USI A-44 to allow the implementation of USI A-44 to proceed. The GI-23 resolution acknowledges that some plants may be resolving USI A-44 with an alternate ac power source. As a result, the GI-23 resolution allows utilities to use this alternate ac to provide power to a cooling source for the seals.

6. REFERENCES

1. Azarm, M.A., Boccio J.L., and Mitra S., "The Impact of Mechanical- and Maintenance-Induced Failures of Main Reactor Coolant Pump Seals on Plant Safety," NUREG/CR-4400, BNL-NUREG-51928, Brookhaven National Laboratory, Upton, New York, December 1985.

2. Campen, C.H., and Tauche, W.D., "Westinghouse Owners Group Report; Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," WCAP-10541, Rev. 2, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, November 1986.

3. Mitra, S., Baradaran, R., and Youngblood R., "Evaluation of Core Damage Sequences Initiated by Loss of Reactor Coolant Pump Seal Cooling," NUREG/CR-4643, BNL-NUREG-52003, Brookhaven National Laboratory, Upton, New York, August 1986.

4. Boardman, T., et al., "Leak Rate Analysis of Westinghouse Reactor Coolant Pump," NUREG/CR-4294, Energy Technology Engineering Center, Rockwell International Corporation, Canoga Park, California, July 1985.

5. Ruger, C.J., Luckas, W.J.,Jr., "Technical Findings Related to Generic Issue 23, Reactor Coolant Pump Seal Failure," NUREG/CR-4948, BNL-NUREG-52144, Brookhaven National Laboratory, Upton, New York, March 1989.

6. Heaberlin, S.W., et al., "A Handbook for Value-Impact Assessment," NUREG/CR-3568,PNL-4646, Pacific Northwest Laboratory, Richland, Washington, December 1983.

7. "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, United States Atomic Energy Commission, Washington, D.C., December 1975.

8. Heiselmann, H.W., Neve, R.G., "Cost/Benefit Analysis for Resolution of Generic Issue 23," NUREG/CR-5167, Scientech, Inc., Idaho Falls, Idaho, 1990.

9. Kittmer, C.A., et al., "Reactor Coolant Pump Shaft Seal Behavior During Station Blackout," NUREG/CR-4077, EGG-2365, Idaho National Engineering Laboratory (EG&G), Idaho Falls, Idaho, April 1985.

7 Bibliography

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

NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, D.C., June 1988.

NUREG-1109, "Regulatory Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout," U.S. Nuclear Regulatory Commission, Washington, D.C., June 1988.

NUREG-1150, Vol. 1, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, D.C., June 1989.

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

NUREG/CR-3840, "Cost Analysis for potential Modifications to Enhance the Ability of Nuclear Plant to Endure Station Blackout," Science and Engineering, Inc., McLean, Virginia, July 1984.

NUREG/CR-4544, "Reactor Coolant Pump Seal Related Instrumentation and Operator Response: An Evaluation of Adequacy to Anticipate Potential Seal Failures," BNL-NUREG-51964, Brookhaven National Laboratory, Upton, New York, December 1986.

NUREG/CR-4821, "Reactor Coolant Pump Shaft Seal Stability During Station Blackout," EGG-2492, Idaho National Engineering Laboratory (EG&G), Idaho Falls, Idaho, May 1987.

NUREG/CR-4906P, "Review of the Westinghouse Owners Group Report WCAP-10541, Revision 2, Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," AECL Report CI-S-263, Chalk River Nuclear Laboratories, Chalk River, Ontario, January 1988.

NUREG/CR-4907P, "Report on the EDF-Montereau Full Scale Test of RCP Seals Under Station Blackout Conditions," AECL Report CI-S-258, Chalk River Nuclear Laboratories, Chalk River, Ontario, July 1985.

NUREG/CR-4985, "Indian Point 2 Reactor Coolant Pump Seal Evaluations," Brookhaven National Laboratory, Upton, New York, August 1987.

DOE/ET/34019-1, "Development of a Reactor Coolant Pump Monitoring and Diagnostic System," BAW-1721, Babcock and Wilcox, Lynchburg, Va., February 1982.

Appendix A

Pump Seal Failure Experience Data

A survey of the data on reactor coolant pump (RCP) seal failures for existing nuclear power plants in the U.S. (reported in NUREG/CR-4400) was performed by Brookhaven National Laboratory (BNL) using the following sources:

1. Nuclear Safety Information Center (NSIC) Files
2. EG&G Licensee Event Report (LER) Summaries (NUREG/CR-1205)
3. Nuclear Power Experience (NPE)
4. Nuclear Plant Reliability Data System (NPRDS)
5. Data Collected for Prioritization of GI-23 (Memo, Riggs to Adensam, December 19, 1980, and EPRI-NP-351)

The data collected covered the period from July 1969 through May 1984 and included only mechanical- or maintenance-induced seal failures during plant operation.

For the reasons presented in Appendix B, the pump seal failures discussed below are primarily for PWRs. However, data collected for both PWRs and BWRs showed that seal failure rates were essentially identical for both types of plants. A more recent study by the NRC Office for Analysis and Evaluation of Operational Data (AEOD) discussed later in this appendix came to the same conclusion.

A total of 173 RCP seal failures in PWRs were obtained from the survey -- 46 for Westinghouse (W) plants with W pumps, 31 for Combustion Engineering (CE) plants with Byron Jackson (BJ) pumps, 28 for Babcock and Wilcox (B&W) plants with BJ pumps, 9 for B&W plants with older two-stage Bingham pumps and 4 for B&W plants with the newer three-stage Bingham pumps. Considering seal failures of all magnitudes, the RCP seal failure rate was calculated to be 26.0 failures per million hours. Seal failures that resulted in a leak comparable to a small LOCA occurred at a rate of 1.3×10^{-2}/reactor-year.

A more recent limited data survey using only NPRDS data from January 1984 to October 1987 was performed by BNL to determine if seal failure rates had improved since 1984. The results indicated that W plants with W pumps showed some improvement in seal failure rate (about a 60% reduction). B&W plants with either BJ or Bingham seals experienced a more significant improvement, about an order of magnitude, while CE plants with BJ pumps had about the same seal failure rate for both periods.

The recent AEOD study reviewed the pump seal failure data in the NPRDS and the Sequence Coding and Search System (SCSS) for the period between January 1985 and March 1990. The failure rate for RCP seals was 11.9 failures per million hours. Comparison of the AEOD results for the past five years with the data reported in NUREG/CR-4400 for the earlier time period indicates that the seal failure rate has decreased by roughly a factor of 2.

Seal failures during normal operation continue to occur both in PWRs and BWRs as shown in Table A-1 and Table A-2 respectively, which list some recent events. Although none of these failures resulted in the large leakage rates seen in some of the earlier events, seal failure still remains a generic problem.

Table A-1

RECENT REACTOR COOLANT PUMP SEAL FAILURE EVENTS IN PWRs

EVENT DATE	PLANT	NSSS	SEAL VENDOR	DESCRIPTION OF FAILURE EVENTS
AUG 1, 1988	ANO-2	CE	BJ	1st & 2nd SEAL STAGES FAILED. LEAKAGE 40 gpm.
SEP 15, 1988	TMI-1	B&W	W	DAMAGED O-RING, 1st SEAL STAGE FAILED. LEAKAGE 9 gpm.
NOV 4, 1988	SEQUOYAH	W	W	DESCRIPTION NOT AVAILABLE.
NOV 9, 1988	WATERFORD 3	CE	BJ N9000	SEALS FAILED AFTER STARTUP. REPLACED PUMPS WITH OLD BJ SEAL.
DEC 5, 1988	PALISADES	CE	BJ	DESCRIPTION NOT AVAILABLE.
DEC 15, 1988	TMI-1	B&W	W	DAMAGED O-RING. SEAL LEAKAGE 8 gpm. W UNABLE TO ADVISE SOLUTION.
DEC 21, 1988	MAINE YANKEE	CE	BJ SU	DEGRADED SEAL PERFORMANCE. SEALS REPLACED WITH BJ N9000 SEAL.
MAR 3, 1989	PALO VERDE 3	CE	KSB	SEALS DAMAGED. LEAKAGE 2 gpm.
MAR 29, 1989	SAN ONOFRE 1	W	W	LEAKAGE 33 gpm. RCS PRESSURE LOWERED TO ATMOSPHERIC. LEAKAGE 22 gpm.
JUN 16, 1989	KEWAUNEE	W	W	O-RING DEGRADED, 1st STAGE SEAL FAILED. REPLACED SEAL.
JUN 19, 1989	INDIAN POINT 2	W	W	1st STAGE SEAL FAILED. LEAKAGE 14 gpm. SEAL REPLACED DURING OUTAGE.
NOV 7, 1989	MAINE YANKEE	CE	BJ SU	N9000 WAS EARLIER REPLACED BY SU TYPE, WHICH FAILED. PLANT SHUT DOWN.
JAN 20, 1990	WATERFORD 3	CE	BJ N9000	SEAL LEAK FOUND DURING SHUTDOWN WHILE CHECKING REACTOR COOLANT SYSTEM LEAKS.
JUL 2, 1990	ST. LUCIE 1	CE	BJ	1st & 2nd SEAL STAGES DETERIORATED. LEAKAGE 3 gpm.
AUG 24, 1990	FORT CALHOUN	CE	BJ	INLET PRESSURE ON 2nd STAGE STEADILY DECREASED. PLANT SHUT DOWN. SEAL REPLACED

Table A-2

RECENT RECIRCULATION PUMP SEAL FAILURE EVENTS IN BWRs

EVENT DATE	PLANT	NSSS	SEAL VENDOR	DESCRIPTION OF FAILURE EVENTS
MAY 23, 1988	NINE MILE PT.2	GE	BJ	SEAL LEAKAGE 5 gpm. PLANT SHUTDOWN, SEAL REPLACED.
MAY 29, 1989	MILLSTONE 1	CE	BJ	BOTH SEALS ON RECIRC. PUMP FAILED. LEAKAGE 45 gpm.
JUN 1, 1989	CLINTON 1	GE	BINGHAM	BOTH SEALS ON RECIRC. PUMP FAILED DURING STARTUP. LEAKAGE 63 gpm.
JUN 23, 1989	QUAD CITIES 1	GE	BJ	750 psi PRESSURE DIFFERENTIAL ACROSS OUTER SEAL CAUSING LEAKAGE.
AUG 8, 1989	RIVER BEND	GE	BJ	SEAL LEAKAGE 4 gpm, PLANT SHUTDOWN. REPLACED SEAL.
FEB 6, 1990	OYSTER CREEK	GE	BJ	BOTH SEALS ON RECIRC. PUMP FAILED. LEAKAGE 5 gpm.

APPENDIX B

EVALUATION OF POTENTIAL FOR CORE DAMAGE DUE TO LOCAs INDUCED BY RECIRCULATION PUMP SEAL FAILURE IN BWRs

B.1 INTRODUCTION

This appendix evaluates the vulnerability of BWRs to core uncovery accidents resulting from leak rates beyond the capacity of the available makeup water due to recirculation pump seal failure. The purpose of the evaluation is to provide information supporting the decision that BWR plants need not be considered in the generic resolution of GI-23.

B.2 GENERAL DESCRIPTION OF RECIRCULATION PUMP SEALS

Leakage around the rotating shaft of a BWR recirculation pump is controlled by a dual mechanical shaft seal assembly. Two similar pump configurations are in use. Both designs use two-stage balanced-endface-type mechanical shaft seals. Control pressure breakdown (seal staging) orifices in the seal assembly regulate the normal pressure drop across each seal to about half the full primary system pressure, although each seal can individually seal against full system pressure.

Two systems provide cooling to the seals of each pump: the reactor building closed cooling water (RBCCW) system and the seal purge system. Some plants do not have a seal purge system, but either system is capable of supplying sufficient seal cooling in normal operation. The RBCCW water is circulated through the secondary side of a heat exchanger that cools the primary water before it enters the lower seal cavity. The seal purge system injects clean cool water from the control rod drive (CRD) system into the lower seal cavity, maintaining the cavity at a pressure sufficient to prevent contaminated primary water from reaching the seals.

There are two paths for the seal cooling water passing through the seal stages to exit the pump: through the seal staging line at about 1 gpm, and across the second seal face through the seal leakage line at about 50 oz/hr. Both lines drain into the drywell equipment sump. In the event of total seal failure, some (approximately 35% of the total leakage) water/steam would escape past an outer restriction bushing into the drywell volume, eventually collecting in the drywell floor drain sump. These sumps are pumped to the liquid radwaste system in normal operation but overflow, when full, to the suppression chamber via downcomers if the pumps are unavailable.

B.3 RECIRCULATION PUMP SEAL FAILURE SCENARIOS

Recirculation pump seal failure scenarios can be divided into two categories: those resulting from mechanical- or maintenance-induced failures and those resulting from a loss of all seal cooling. Because of the dependence of seal cooling on ac power supplies, seal failures are linked to the reliability of onsite and offsite ac electric power supplies. Consequently, Generic Issue 23 considers the effects of station blackout (SBO) on seal performance to the extent they are not addressed in USI A-44, "Station Blackout".

B.3.1 Mechanical- and Maintenance-Induced Seal Failure Scenarios

Operating experience indicates that PWRs and BWRs experience approximately the same number of seal failures during normal operation due to mechanical failure of components of the seal assembly and improper maintenance, installation, and pump startup and shutdown procedures (References 1 and 2). An assessment of the scenarios of seal failures that led to excessive leak rates indicated that only two events in BWRs had the potential for a small LOCA (Reference 2). The smallest LOCA considered for BWRs consisted of a leak rate beyond the capacity of the purge water injected from the CRD system (generally between 50 and 60 gpm). Both of these events were attributed to basic design deficiencies that have been eliminated by corrective actions.

In addition to this experience data, BWR plants can be expected to exhibit significantly lower leak rates from seal failures than do PWR plants because of the lower system pressure in BWRs and the presence of isolation valves in the reactor recirculation loops. Estimates of the maximum leakage from a failed BWR recirculation pump seal assembly are discussed in a later section.

The primary means of limiting the risk significance of seal-failure-induced LOCAs in BWRs with ac power available is the ability to isolate the recirculation lines. In addition to the presence of isolation valves and the lower leakage potential of BWRs, the safety impact of mechanical- and maintenance-induced seal failures is further reduced by the large makeup capabilities of BWR systems. For seal leakage scenarios without station blackout, BWRs have available a multitude of high-capacity injection systems, including high-pressure coolant injection (HPCI) or high-pressure core spray (HPCS), reactor core isolation cooling (RCIC), low-pressure coolant injection (LPCI), and feedwater. Therefore, the lower leak rates, the presence of recirculation loop isolation valves, and the makeup capabilities of the normal and emergency vessel water level control systems should be sufficient to remove these plants from consideration for seal-induced LOCA risk during normal operation.

B.3.2 Seal Failure Scenarios Resulting From a Loss of All Seal Cooling

Loss of seal cooling can be attributed to loss of all ac power or simultaneous loss of the RBCCW and seal purge systems due to other causes. Since each of these seal cooling systems is functionally independent (independent piping, pump, water source, power, control, and instrumentation) loss of cooling sequences without blackout are unlikely for plants having both systems. Some operating BWRs do not have a seal purge system (Reference 3); however, seal failure scenarios resulting from a loss of RBCCW in these plants with ac power available should not pose a safety concern because of the numerous normal and emergency makeup systems available in BWRs.

Station blackout will result in a loss of all seal cooling and unavailability of all ac-powered mitigating systems. Because of the increased seal failure probability and the loss of ac-powered makeup systems under SBO conditions, the significance of recirculation pump seal failures during SBO is discussed in more detail.

B.3.3 Detailed Description of Seal Failure Modes Under Station Blackout Conditions

Under normal conditions, with the primary reactor system at or near rated temperature and pressure, the RBCCW and seal purge systems maintain the seal temperatures at approximately 120°F (Reference 3). If all seal cooling is lost, the pump seals will heat up. Vendor tests have indicated that seals begin to deteriorate when seal temperatures exceed 250°F. This temperature is reached approximately 7 minutes after loss of seal cooling if the staging flow is maintained (Reference 3). Under these conditions, the temperature surrounding each stage will approach system temperature (approximately 530°F) with the pressure drop still shared between the two stages. If the staging flow is valved closed at the bleedoff, the throughflow of hot water (1 gpm) is cut off, but the second seal stage is exposed to full system pressure (approximately 1040 psia). This will delay the heating of the second-stage seal and somewhat reduce the probability of reaching flashing conditions and the associated instability or "popping open" seal failure mode. For this reason it is recommended in Reference 4 that the preferred option is to valve the staging flow closed.

Whichever option is taken, the elastomer axial seals may fail because of high temperature for two reasons. The polymer seals could partially extrude and produce increased frictional drag forces, which could cause the seal faces to be pulled apart by the local shaft movement caused by its thermal expansion. This is more likely in BWRs than in PWRs since the frictional forces are a larger percentage of the hydraulic forces because of the lower system pressure in BWRs.

The second reason for high leakage due to high-temperature elastomer seal failure could be their complete extrusion and exposure of the annular clearance. This is probably less likely in BWRs because of the lower system pressure.

As pointed out in Reference 4, if the staging flow is maintained, a cascading of elastomer failures from one stage to the next is fairly certain because of the similarity of the stages. If the staging flow is cut off, the seal heating and stage failure might be delayed, but if only one stage failed, high leakage would result.

Another potential seal failure mode under SBO conditions results from the likelihood that the flashing of water to steam between the seal faces will give additional opening force, upset the seal balance, and cause the faces to pop completely open. The factors affecting the susceptibility of a seal to this type of instability are fully discussed in Reference 4. Since BWRs operate at a lower system pressure than PWRs and are closer to saturation temperature than PWRs, it appears that BWR seals will be susceptible to "popping open." Even if the staging flow is valved closed, reducing the influx of hot water, there is no guarantee that the seal faces will not "pop open." With no staging flow, the second-stage seal will be subject to full system pressure and consequently a higher leakage, allowing more hot fluid to reach the seal, with the hotter conditions resulting in higher leakage. This worsening situation may itself cause the seals to reach system temperature.

Therefore, it appears that the likelihood of seal failure under SBO conditions is even greater for BWRs than for PWRs primarily because of "popping open" caused by conditions closer to saturation.

B.4 ASSESSMENT OF MAGNITUDE OF BWR SEAL LEAKAGE

Having concluded that BWR recirculation pump seals are likely to fail during a station blackout, it is of interest to estimate the worst-case, upper-bound leak rate that can be expected for completely failed seal systems. Two analyses (References 4 and 5) are useful in making this estimate.

Reference 5 presents an analysis of the fluid loss through a grossly degraded seal system. Both seal stages are assumed to exhibit failure that encompasses warpage, fractures, and grooving of the seal faces. These failure modes were modeled in a simplified manner by assuming that each seal face is separated by a 0.010-inch gap. This is a much larger sealing gap than is present during normal operations, but still represents some flow resistance. The failure mode of "popping open" was not considered. This failure mode would have resulted in the seal faces being completely open (essentially no flow resistance).

Using the 0.010-inch gap representation of seal degradation, a two-phase flow calculation was performed (Reference 5) along a complex annular fluid leakage path through the seal system. The calculations were performed for two similar configurations representing the two pump manufacturer designs presently in use in BWRs. The exact initial conditions used are unclear, but appear to be approximately 530°F/1040 psia. The water enters the flow passage as a single-phase fluid, drops below its saturation pressure because of frictional and expansion/contraction losses with some flashing to steam, and exits as an expanding two-phase fluid.

The leakage rates calculated in Reference 5 are less than 8 lbm/sec per pump. This would require a makeup rate of less than 60 gpm per pump from ambient water sources (less than 70 gpm per pump for feedwater system makeup water at 420°F with no blackout). Approximately 67% of this leakage will be piped to the drywell equipment sump through the controlled bleedoff and leakoff lines. The remaining 33% of the leakage will escape around the pump shaft to the drywell, eventually collecting in the drywell floor drain sump.

As mentioned above, these calculations assume degraded seals that remain in the seal cavity under spring loading used to maintain contact between faces. However, as discussed in Section 3.3, BWR seals are likely to experience two-phase instability ("popping open") failure under SBO conditions. For this failure mode, the seals would remain intact but their faces would be fully open. Binding failures, which are also probable, would be essentially equivalent to total seal failure. Therefore, it is possible that the leak rate could be greater than 60 gpm per pump for these failure modes associated with SBO conditions. The leakage path with the seal faces fully open would essentially have two fewer flow restrictions than in the calculations performed in Reference 5.

To estimate the effects of this less restrictive flow passage, a calculation discussed in Reference 4 was employed. This calculation uses the same methodology used in Reference 5 to compute the leakage through a similar two-stage hydrodynamic seal assembly assuming the seal faces to be fully open.

Figure B-1, taken from Reference 4, shows the effect of system pressure on leak rate. The upper curve represents single-phase water at 200°F and shows a square root variation of leak rate with system pressure. The lower curve represents two-phase water at 530°F and shows an almost linear variation of leak rate with pressure in this pressure range. Note that Figure 22 of Reference 4 has an incorrect square root symbol associated with the lower curve. The two-phase leakage is lower and more strongly dependent on pressure because of the reduced choking speed in the presence of steam and the reduced area available for liquid flow because of pockets of steam. The exit void fraction is also shown in the figure.

To apply these results to recirculation pump seals in BWRs, the 530°F curve is linearly extrapolated down to a system pressure of 1040 psia. This results in a leak rate of about 12.5 lbm/sec per pump, which corresponds to a makeup rate of a little over 90 gpm per pump of ambient makeup water. This is about 50% higher than the results of Reference 5, which does not consider fully open seals.

It is therefore likely that the maximum leakage due to recirculation pump seal failures during a station blackout would be somewhat less than 100 gpm per pump or 200 gpm total for a typical two-loop BWR assuming that the seals in both pumps fail. Note that two older BWRs of the BWR-2 class (Nine Mile Point 1 and Oyster Creek) have five recirculation pumps. However, each of these pumps is smaller (1000 HP/pump vs 8000 HP/pump), and the total leakage is assumed bounded by the 200 gpm. This assumption should be further verified by a plant-specific analysis.

B.5 RISK SIGNIFICANCE OF BWR RECIRCULATION PUMP SEAL FAILURES DURING STATION BLACKOUT

For most BWRs, pump seal failure with a maximum leakage of 200 gpm would not contribute measurably to the probability of core damage in a station blackout event. These BWRs depend on primary coolant loss as the normal mode of core cooling during station blackout. The cooling loss occurs through the safety relief valves, and makeup is provided by the steam-driven RCIC pumps. RCIC and other independently powered emergency vessel water control systems (HPCI/HPCS) will easily compensate for the 200-gpm inventory loss.

The flow rates and capacities of the emergency makeup systems are plant specific. Based on Reference 4 and selected individual plant FSARs, the approximate flow available from the RCIC system is 500-800 gpm with an additional 2000-5000 gpm available from the HPCI/HPCS system. Both these systems draw water from the condensate storage tank (CST), which has a minimum capacity of about 135,000 to

150,000 gal. In addition, these systems can also draw from the suppression pool, which has a typical capacity of about 900,000 gal. Furthermore, the seal leakage is collected in the two drywell sumps, which will overflow to the suppression pool during a station blackout. Therefore, if the emergency makeup systems are connected to draw from the suppression pool, there is essentially a closed-loop path between the seal leakage and the makeup water source. This would provide mitigation of seal-failure-induced LOCAs for station blackouts as long as dc power was available from the station batteries.

The likelihood of the unavailability of both independently powered vessel water control systems (RCIC, HPCI/HPCS) during a station blackout is very low; furthermore, failure of these emergency systems during a station blackout leads to core damage even without additional leakage caused by seal failure. These multiple system failures are already considered in PRA analyses of station blackout events and the additional leakage caused by seal failure would not significantly alter the course of the accident or increase the probability of core damage.

A few older BWRs rely on maintaining primary system integrity to provide core cooling during SBO. These designs do not have the usual independently powered emergency makeup systems (RCIC, HPCI/HPCS) but utilize an isolation or emergency condenser that condenses steam from the core and returns it to the primary system. Reference 6 indicates that pump-seal-induced LOCAs could be important to risk in this type of plant design since there is no means to maintain primary coolant inventory during station blackout.

Reference 7 indicates that there are six operating BWRs utilizing isolation condensers. Two of these (Dresden 2 and 3) have HPCI systems that do not depend on ac power. Therefore, it does not appear that these two plants would be vulnerable to core damage from pump-seal-induced LOCAs during station blackout.

One plant (Millstone 1) has a gas-turbine-generator-driven feedwater coolant injection (FWCI) system capable of delivering 3600 gpm of water from a 460,000-gallon-capacity CST. However, this gas turbine generator is considered an emergency ac power source and is therefore unavailable during station blackout. Thus this plant appears vulnerable to core damage due to pump-seal-induced LOCAs during station blackout because no makeup systems are available for mitigation.

Two additional plants (Oyster Creek and Nine Mile Point 1) also appear susceptible to core damage from a pump-seal-induced LOCA during station blackout. Although the value of the bounding seal leakage is unclear because of the five smaller pumps (1000 HP each) in the design, it appears that no ac-independent makeup systems are available to mitigate such an event.

The final plant (Big Rock Point) is a special case because of its small size (60 MW) and unique design features. This plant has two recirculation pumps, each with a 400-HP motor. Therefore, the upper-bound seal leakage can be expected to be much less than the 100 gpm per pump estimated for the typically larger pump (8000 HP) designs. In addition to the expected lower pump seal leakage, this plant design includes a unique steam separation drum that contains a reservoir of water capable of supplying makeup water to the primary system by means of gravity feed in the event of a pump-seal-induced LOCA during station blackout. During normal operation, this drum contains 26,700 lb of water at 500°F. This is a little more than 4000 gal. Decay heat is removed via a natural circulation emergency condenser.

Based on a very rough assumption that the seal leakage is about 10% of the total 200 gpm of a typical plant and since the pumps are only 5% as large, the steam drum could supply makeup for a little over three hours. Since it is unclear if this is sufficient time to cool down this small plant and since the leak rate is really unknown, a more detailed plant-specific analysis would appear to be required to determine if this plant is at significant risk from a seal-induced LOCA during station blackout.

B.6 CONCLUSION

For normal operations, BWRs would not be vulnerable to core damage due to a recirculation-pump-seal-induced LOCA. BWRs are predicted to have seal leak rates lower than PWRs as indicated by historical data and analytical calculations. They have isolation valves in the recirculation loops and large-capacity normal and emergency vessel water level control systems to compensate for seal leakage.

For station blackout conditions, the maximum seal leakage is estimated to be less than 100 gpm per pump assuming complete opening of both seal stages due to two-phase flow instability or binding failure. For most BWRs, the resultant total inventory loss rate of 200 gpm would not contribute to the probability of core damage for a station blackout event. These BWR designs include independently driven (steam or dedicated diesel) emergency makeup systems (RCIC, HPCI/HPCS) that can easily compensate for the inventory loss.

However, four older BWRs (Millstone 1, Oyster Creek, Nine Mile Point 1, and Big Rock Point) appear to be potentially vulnerable to a pump-seal-induced LOCA during station blackout. Two of these BWRs (Oyster Creek and Nine Mile Point 1) have smaller recirculation pumps (1000 HP) than do typical BWRs (8000 HP). However, they have five recirculation loops and associated pumps. Although the total seal leakage during a station blackout event may be slightly less than the 200 gpm for a typical BWR, these designs do not have any systems not powered by ac that are capable of supplying makeup water.

One BWR (Big Rock Point) is a very small plant using two 400-HP recirculation pumps. In addition, its unique design includes a steam drum with a reservoir containing about 4000 gal. of water, but no other makeup systems that are not powered by ac. It is difficult, from available information, to estimate the magnitude of seal leakage for these smaller pumps or the adequacy of the rather limited water capacity of the steam drum to supply makeup for a long enough duration to either cool down the reactor or survive an extended blackout. Therefore, it is suggested that this BWR be considered vulnerable to pump seal LOCA during blackout unless a more detailed plant-specific analysis can prove otherwise.

This evaluation supports the conclusion that BWRs (except for four older plants) need not be considered for further action as part of Generic Issue 23. The four older BWRs discussed above appear potentially vulnerable to core damage due to pump seal failures during a station blackout and should therefore be further evaluated for loss of all seal cooling due to station blackout.

B.7 REFERENCES

1. Ruger, C.J., and Luckas, W.J., Jr., "Technical Findings Related to Generic Issue 23: Reactor Coolant Pump Seal Failure," NUREG/CR-4948, BNL-NUREG-52144, Brookhaven National Laboratory, Upton, New York, March 1989.

2. Azarm, M.A., Boccio, J.L., and Mitra, S., "The Impact of Mechanical- and Maintenance-Induced Failures of Main Reactor Coolant Pump Seals on Plant Safety," NUREG/CR-4400, BNL-NUREG-51928, Brookhaven National Laboratory, Upton, New York, December 1985.

3. "BWR Owner's Group Evaluation of NUREG-0737 Item # K.3.25 - Effect of Loss of Alternating Current Power on Pump Seals," BWR Owner's Group, May 1981.

4. Kittmer, C.A., et al., "Reactor Coolant Pump Shaft Seal Behavior During Station Blackout," NUREG/CR-4077, EGG-2365, Idaho National Engineering Laboratory (EG&G), Idaho Falls, Idaho, April 1985.

5. "Recirculation Pump Shaft Seal Leakage Analysis," NEDO-24083, Licensing Topical Report, General Electric Co., November 1978.

6. Kolaczkowski, A.M., and Payne, A.C., Jr., "Station Blackout Accident Analysis," NUREG/CR-3226, SAND82-2450, Sandia National Laboratories, Albuquerque, New Mexico, May 1983.

7. "Additional Information Required for NRC Staff-Generic Report on Boiling Water Reactors," NEDO-24708, General Electric Co., August 1979.

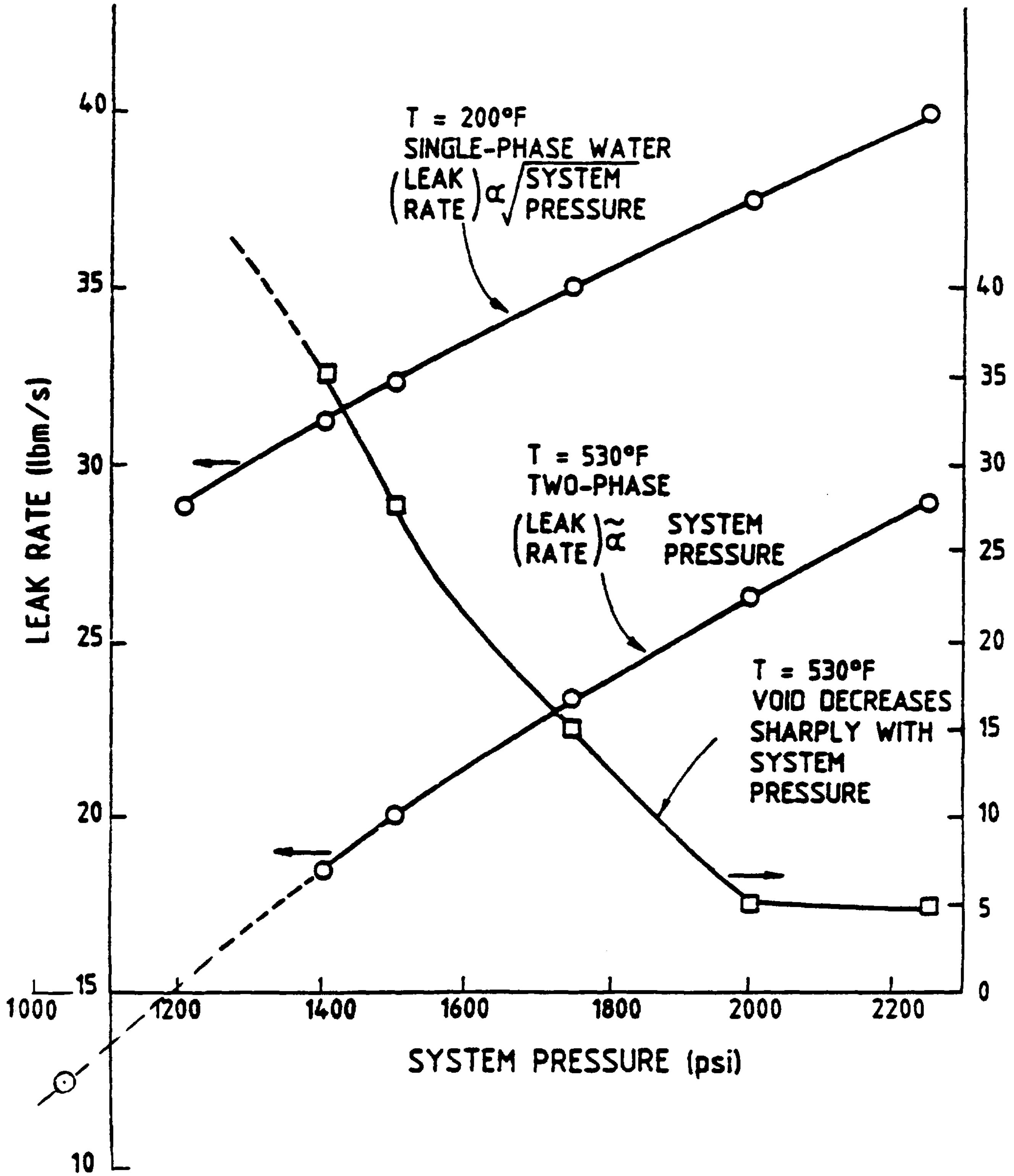

Figure B-1 Leak Rate and Exit Void Fraction Versus System Pressure for Two System Temperatures (200°F and 530°F) (NUREG/CR-4077)

Appendix C

COST/BENEFIT FOR A TOTALLY INDEPENDENT SEAL COOLING ARRANGEMENT

Introduction

This appendix describes a simple and comparatively inexpensive generically applicable means of cooling the reactor coolant pump seals of PWRs following a loss of all normal seal cooling, including a station blackout, a loss of component cooling water (CCW), or a loss of service water (SW). There are ordinarily two means of cooling the seals: (1) by continuous seal injection of borated reactor-coolant-grade water from the charging or reactor coolant system (RCS) makeup pumps and (2) by continuous cooling of the reactor coolant pump thermal barrier heat exchanger around the shaft seals by means of the component or closed cooling water system.

The arrangement described and analyzed below is based on the need to provide RCP seal cooling for an assumed period of 8 hours while the RCS is maintained in the hot standby mode.

A simplified cost/benefit analysis of the suggested design was performed. This analysis considers only the benefits derived from the prevention of seal failure events related to station blackout.

Proposed Method

In view of the fact that all PWRs have the thermal barriers described in (2) but C-E plants, with the exception of Maine Yankee and Palo Verde do not have the seal injection described in (1), it was decided to explore a means to provide independent cooling to the RCP thermal barriers rather than an alternative means of seal injection. In this way, nearly all plants could be covered by the same general solution.

This approach has other advantages in that pure borated water is not required, nor is it necessary to achieve charging pump discharge pressures greater than 2200 psig.

Also, since the RCS is in the hot standby mode, the only requirement for RCS makeup flow would be to compensate for RCP seal leakoff, not RCS shrinkage. The higher leakage hydrostatic seal designs have a 3-gpm leakoff per pump in normal operation. With loss of seal injection and only thermal barrier seal cooling, this leakoff increases to about 5 gpm. Assuming a 5-gpm leakoff per pump, this becomes 1200 gallons per hour leakoff for four pumps. In 8 hours, the total losses are less than 10,000 gallons. This lost inventory is substantially less than the 25 gpm assumed lost during a station blackout.

Description

The majority of nuclear plants possess at least one existing diesel-driven fire pump capable of pumping 1500 gpm at 100 psig discharge and ambient temperature. Furthermore, the fire protection system loop supply header ordinarily runs throughout the plant and through the reactor auxiliary building (RAB).

The proposed solution is to provide a cross-connection between the fire protection loop supply header and the CCW system somewhere in the RAB near the CCW supply line header to the RCPs. The connection would be made through a pressure control valve and a normally closed dc-powered valve, FP-1 (Figure C-1). Pressure would be reduced to the normal pressure of the CCW system, approximately 60 psig.

Modifications may be required to allow remote manual isolation of the unnecessary loads on the CCW system such as to the spent fuel pool or the RCP motors. This may involve converting several CCW valves to dc power or using an air or nitrogen accumulator to close the valves. These are represented by the symbol I in Figure C-1.

Since there is normally no fire protection system return header, a new line of approximately 3 to 4 inches would be required to run from the CCW system in the RAB to the fire water storage tank or equivalent. In this way, the tank volume would be maintained constant. This line would also contain a temperature control valve to maintain the proper temperature of the flow exiting the thermal barriers at approximately 125°F, a typical return temperature for the CCW system.

The total required flow rate through the RCP thermal barriers is assumed to be approximately 50 gpm per pump. For 4 pumps, the flow rate is then 200 gpm. This is well within the 1500-gpm capacity of one fire pump.

Should failure of the RCP seals or thermal barrier occur while this system is in operation, provisions already exist within the CCW system to automatically isolate the seals inside containment.

The water returned to the fire water storage tank should be cooled by ambient losses to the environment, particularly for uninsulated tanks with only freeze protection. For a flow rate of 200 gpm, 96,000 gallons would be circulated back to the tank in 8 hours. Assuming no heat losses, if 100,000 gallons were removed from the 250,000 gallons originally in the tank at 90°F and replaced with 100,000 gallons of water at 125°F, the final water temperature would be 104°F.

For plants without a diesel-driven fire pump, a possible solution is to install a dedicated pump and diesel or other driver of sufficient capacity just to cool the RCP thermal barriers, i.e., 200 gpm (approximately 80-HP driver). This pump may need an enclosure housing. The water source could be from the ultimate heat sink such as a lake or river, the fire water storage tank, or any other suitable storage tank.

For comparison purposes, an 80-HP pump is only a fraction of the size of the 1100KW (1500 HP) diesel generator that would be required to provide electric power to one charging pump, an alternative option considered in previous studies of this problem.

Costs

This cost estimate includes allowances for the hardware, its installation, and associated engineering and QA requirements. The estimate relies exclusively on generic cost data and thus should be viewed as an approximation. Furthermore, because of uncertainties regarding the environment in which the work would be performed and the technical specificity of the hardware (e.g., material, grade) the estimate incorporates a number of assumptions that could adversely affect its accuracy.

The following documents were relied upon for the data and methodology used in developing these cost estimates:

1. NUREG/CR-4627, Revision 1, "Generic Cost Estimates. Abstracts from Generic Studies for Use in preparing Regulatory Impact Analyses," February 1989.

2. "Energy Economic Data Base (EEDB) Program," (Pegasus and Concise computer printouts), United Engineers and Constructors for DOE, 1987.

3. NUREG/CR-5160, "Guidelines for the Use of the EEDB at the Sub-Component and Subsystem Level," May 1988.

Other impacts are likely to be incurred as a result of this proposed modification. These include technical specification changes, procedure rewrites, training, and periodic maintenance and inspection of the additional hardware. These costs are not addressed in this preliminary cost analysis.

Based on the suggested design (Figure C-1 and Table C-1), all PWRs will require new piping, valves, and circuits (including sensors and switches).

Piping

The design calls for 10 feet of 3/4-inch (diameter), 200 feet of 3-inch, and 0 to 500 feet of 4-inch piping. Carbon steel piping is specified.

NUREG/CR-5160 specifies labor, site material, and manufactured pipe costs by diameter on a linear foot basis. The estimate assumes schedule 40 carbon steel, non-nuclear-safety-grade piping. If 500 feet of 4-inch piping is included, the total cost of piping and associated hangers is estimated to be about $183,000. Alternatively, if this larger pipe line can be avoided, the total pipe cost is reduced to about $37,000. These estimates adopt the appropriate values in Tables 4.3 and 4.5 of NUREG/CR-5160 with the following additions and adjustments:

a. Assumes hourly labor rate of $39.60 from NUREG/CR-4627. This includes an overhead allowance of 59%.

b. Person-hours reported in Table 4.3 are for Greenfield Construction (new plant). Labor productivity adjustment factors were applied to reflect work done in a backfit environment (NUREG/CR-4627):

access and handling	-	0.1	
congestion and interference	-	0	
radiation	-	0	
manageability	-	0.2	(assumes non-outage- related activity)

c. A cost allowance of 25% was added to account for engineering and quality assurance (QA) (see NUREG/CR-4627).

Valves

The design identifies the need for 12 new carbon steel valves (various sizes and types) and modifications to 12 existing valves. The generic cost methodology estimates valve costs based only on valve size and material. In reality, other factors will have strong influences on the unit costs. These factors include such aspects as valve type (i.e., check valve, gate valve) and features (i.e. manual versus motor operated, type of seals, and whether or not it bears an "N" stamp). The cost estimates do not capture any of these factors and should be viewed as rough approximations based on "average" or typical valve characteristics. The expectation is that, for simpler valves such as check valves or those that are manually operated, the estimates probably overestimate costs. For more complex valves such as gate valves with motor operators, the data probably underestimate costs. Since a wide array of valves are incorporated in the modification, it is assumed that the aggregate valve cost developed here is a reasonable substitute.

The estimate for the 12 new valves is about $9,000. This includes installation labor, factory cost, and engineering and QA. The labor rate, labor productivity adjustment factors, and engineering/QA factor discussed above were employed here.

The design also identifies the need to convert 12 existing valves to dc power to isolate unnecessary CCW system heat loads. For the purposes of this cost analysis, it is assumed that this effort is comparable with the purchase and installation of the 12 new valves discussed above. It is expected that the equipment costs to modify existing valves will be less than the cost of new valves. However, the labor installation costs to remove, modify, and then install the same valve will be higher than the labor cost of just installing a new valve. It is assumed that these factors will balance out.

Thus the valve requirements included in the modification are estimated at a total cost of about $18,000.

Controls, Sensors, Switches, and Wiring

The modification identifies running control circuitry to the main control room (13,200-ft circuits) and the installation of a temperature transmitter, pressure sensor, and two manual operation switches.

EEDB data at this level of detail is very limited and consequently the cost estimate is somewhat speculative. Instrumentation and control accounts in the EEDB include costs for sensors and measuring elements, transmitters, switches, controllers, detectors, and local and remote indicators. A point or unit cost is available for a "typical" instrument or control that represents the average cost per control for a wide array of control points for a given system or location at the plant. For the purposes of this cost exercise, the average cost per control point for the PWR feedwater system was used as a substitute. Applying this average value to the 4 control points included in the design results in a cost of about $26,000. This includes labor costs, site materials, factory costs, and engineering and QA.

The cost of adding 13 circuits was obtained from a lighting and service power account that includes costs for conduit, electric cable and wire, fixtures, and outlets. The cost per circuit for lighting and service power to the emergency feed pump building in a PWR was used as the substitute. These costs adjusted for labor productivity differentials, current labor rates, and engineering and QA requirements produced a cost for the 13 circuits of about $30,000.

Thus the cost per reactor of adding the design are:

	Without Return Piping	With Return Piping
Piping		
Assumes no 4-inch piping	$ 37,000	
Includes 500 ft. of 4 piping		$183,000
Valves	$ 18,000	$ 18,000
Circuits and control points	$ 56,000	$ 56,000
TOTAL	$111,000	$257,000

The costs identified above would be incurred by all 76 PWR power reactors. Therefore, the total industry cost would range from about $8.4 million to $19.5 million. These results suggest that the per reactor and industry costs are highly sensitive to the inclusion of 500 feet of 4-inch-diameter piping used to return the fire water from the RCP thermal barrier heat exchanger to the fire water storage tank.

<u>Additional Hardware</u>

The proposed modification is dependent on the availability of an existing diesel fire pump to service the new system. Although most PWRs are assumed to already have such a diesel fire pump in place, it is assumed for the purposes of this cost analysis that approximately 10% or 8 reactors will have to purchase and install this additional hardware.

The design includes the following:

a. Pump, centrifugal, cast iron, 200 gpm at 75 psig;
b. Diesel driver engine, assume 80 HP;
c. 275-gallon fuel oil tank with level indicator;
d. Ventilated enclosure housing.

The estimate identified comparable equipment included in the EEDB as a basis for estimating cost. In general, the EEDB proxies would tend to overstate the actual cost because the equipment specifications of the proxies are more rigorous.

a/b - EEDB Account 252.2211 Diesel Engine Fire Pumps (technical specs: 2000 gpm, 200 psig, 235 HP)

Hardware and Equipment	--	$42,747
Installation Labor	--	$20,954
Engineering/QA	--	$13,301
Total - diesel fire pump		$77,002

c - EEDB Account 252.2232 Diesel Engine Fuel Oil Tank (550 gallons)

Hardware and Equipment	--	$2,275
Installation Labor	--	$2,364
Engineering/QA	--	$ 864
Total - fuel oil storage tank		$5,503

d - Ventilated enclosure building - for the purposes of this cost analysis, the staff relied upon a 600 square foot concrete block building that was proposed to house batteries for potential BWR Mark I Containment Improvements (NUREG/CR-5278). Its cost was scaled back 50% because a 300 square foot area was deemed adequate. With a 25% allowance for engineering and QA, its cost was estimated at approximately $70,000. Thus the total cost estimate for the additional hardware is about $150,000.

Summary

Total cost on a per reactor basis are expressed as "high estimate" and "low estimate" depending on whether the 500-foot pipeline is incorporated in the design or not. The cost of the additional hardware, which is assumed to affect only 10% of the PWRs, is allocated across the 76 reactors to arrive at an average or representative per reactor cost in the following listing:

COST PER REACTOR

High estimate (assumes 500 ft of 4-inch return pipe)	-	$273,000
Low estimate (omits 500 ft of 4-inch return pipe)	-	$127,000

INDUSTRY COST (76 POWER REACTORS)

High estimate	-	$20.7 Million
Low estimate	-	$9.6 Million

These cost estimates capture only the direct costs of implementing the physical modification. Other costs such as procedural and administrative and costs for periodic maintenance and inspections are not included in these calculations.

Cost/Benefit

An additional cost/benefit calculation was performed to explore the possibility that a relatively inexpensive engineering fix that could address essentially all of the postulated loss-of-seal-cooling scenarios could be found. However, the benefit used in the calculation was only that portion attributed to the SBO/seal failure events. This was because of the very uncertain nature of the benefits that may result from the other loss-of-seal-cooling scenarios. As stated elsewhere, these other scenarios tend to be very dependent on plant-specific design features; therefore a plant may or may not be vulnerable to them, and consequently may or may not realize a benefit.

From the previous cost/benefit work, the safety benefit from off-normal conditions (loss of seal cooling) was calculated to be 62,814 person-rem as an industry total. Approximately one-half of this benefit was attributed to SBO-related events. In the previous work the core damage frequency (CDF) per reactor-year induced by RCP seal failure has been calculated as follows:

$$CDF_{SBO} = 5.6E\text{-}06 \text{ and}$$
$$CDF_{CCW} = 6.0E\text{-}06$$

The cost of the system that could fix almost all of the loss-of-seal-cooling scenarios was estimated (high estimate) to be about $273,000 per reactor. The cost/benefit ratio was then calculated ignoring relatively small values such as NRC cost, averted property damage, and occupational exposure.

$$\text{Cost/benefit ratio} = \frac{\$273{,}000 \text{ per plant x } 76 \text{ plants}}{30{,}000 \text{ person-rem}} = \$692/\text{person-rem}$$

This calculation shows the "inexpensive fix" to be cost effective based on the SBO contribution alone. Since almost all plants can realize some benefit from other scenarios (involving SW and CCW), this ratio would only become more cost effective.

Conclusion

This additional cost/benefit work was done to explore the feasibility, and possibly the desirability, of implementing an engineering fix that can address almost all loss-of-cooling scenarios.

The resolution item 3, while not mandating such a fix, allows for such an approach. This calculation shows that it is feasible and it may also be desirable to take such an approach. This may eliminate the need to evaluate a particular plant for possible plant-specific dependencies involving CCW or SW.

Table C-1, New Equipment

A. Plants With Existing Diesel Fire Pump

1. Pressure Reducing Station

Quantity	Description
1 ea.	2-in. dc motor-powered throttling valve (PCV), 150-lb carbon steel
3 ea.	3-in. gate valves, hand operated, 150-lb carbon steel
2 ea.	3-in. x 2-in. std. wall pipe reducers, welded carbon steel
1 ea.	1.5-in. angle relief valve, 150-lb carbon steel
1 ea.	3-in. dc solenoid-operated plug valve, 150-lb carbon steel, std. wall (FP-1)
200 ft	dc control lines to main control room (PCV, FP-1)
1 ea.	3-in. check valve, 150-lb carbon steel, std. wall
200 ft	3-in. std. wall (Schedule 40) carbon steel piping (from fire protection header to CCW system supply header in auxiliary building)
1 ea.	pressure sensor, 150-lb rating, 0-150 psi
10 ft	3/4-in. std. wall carbon steel piping (pressure sensing line)
1 ea.	manual operation switch for FP-1, main control room

2. <u>Temperature Control Station</u>

Quantity	Description
1 ea.	3-in. dc motor-operated throttling valve (TCV), 150-lb carbon steel
3 ea.	3-in. gate valves, hand-operated, 150-lb carbon steel
1 ea.	3-in. dc solenoid-operated plug valve, 150-lb carbon steel, std. wall (FP-2)
1 ea.	temperature transmitter 0-200°F, 150-lb rating
200 ft	dc control lines to main control room (TCV, FP-2)
1 ea.	manual operation switch for FP-2, main control room
500 ft	4-in. std. wall (Schedule 40) carbon steel piping (from auxiliary building to fire protection fill line)

3. <u>Modifications to CCW System Isolation Valves</u>

Quantity	Description
12 ea.	conversion of existing manual, air, or ac-powered valves to dc solenoid-operated or motor-operated: Assume 6 conversions to solenoid 6 conversions to motor
12 ea.	200 ft control circuitry to main control room

Note: The number of conversions required to isolate the RCP seal thermal barriers from the other CCW heat loads is very plant specific

B. Plants Without Existing Diesel Fire Pump

Plants without an existing diesel fire pump will require, all of the equipment required for plants with a diesel fire pump.

In addition, the following will be required:

Quantity	Description
1 ea.	pump, centrifugal, cast iron, 200 gpm at 75 psig
1 ea.	diesel driver engine, assume 80 HP
1 ea.	275-gallon fuel oil storage tank with level indicator
1 ea.	ventilated enclosure housing

Note: A diesel-driven pump is assumed. Other drives such as steam turbine might be selected by the individual utilities.

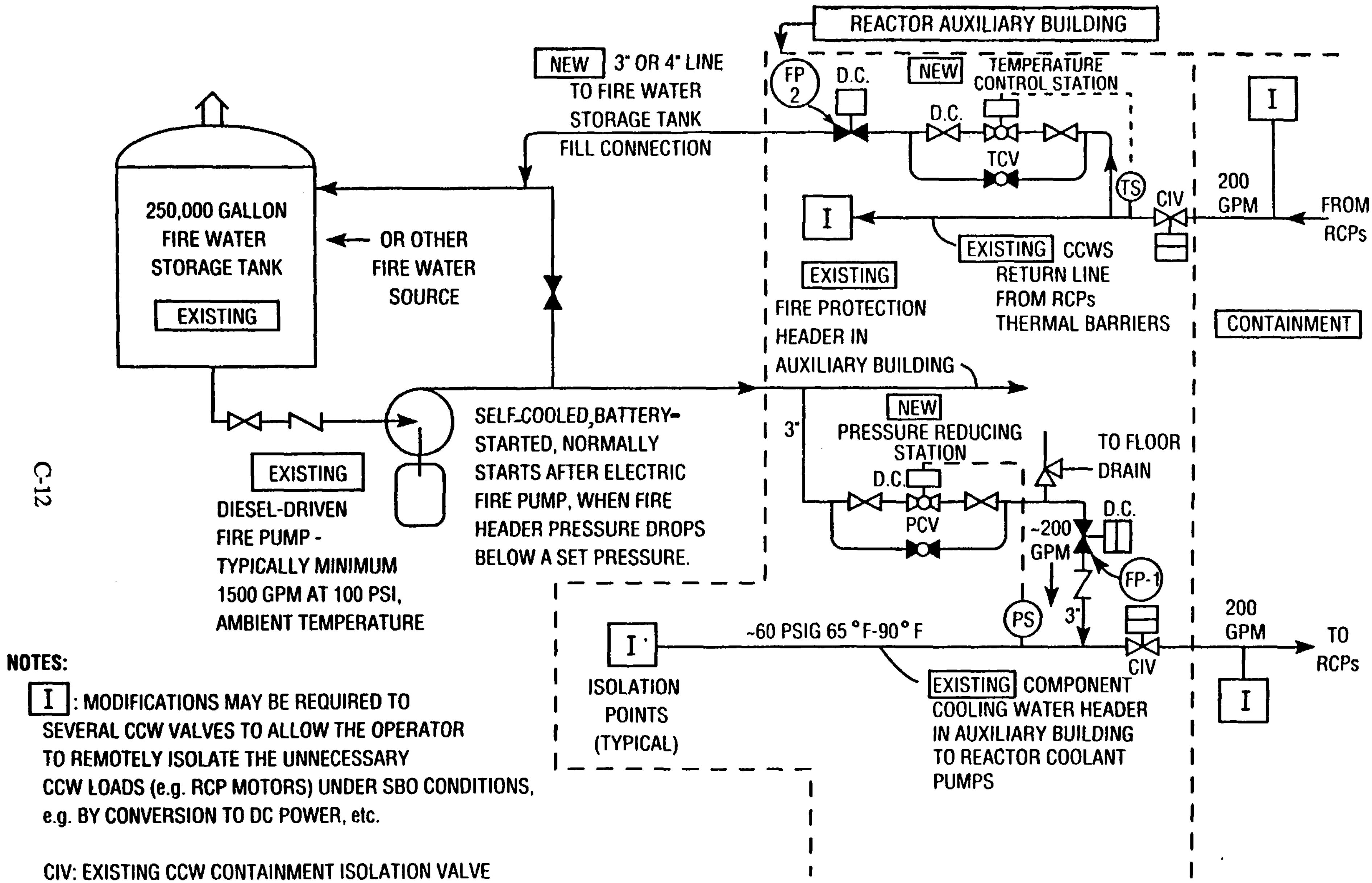

Figure C-1 Independent RCP Seal Cooling Arrangement

Appendix D

COMPARISON OF CORE DAMAGE FREQUENCY FROM GI-23 WITH RESULTS FROM NUREG-1150

Table D-1 presents a comparison of the Core damage Frequency (CDF) per reactor-year from Generic Issue 23 studies with the CDF of the three PWRs of NUREG-1150; Surry, Sequoyah, and Zion. The three NUREG-1150 PWRs are Westinghouse plants that use Westinghouse reactor coolant pumps and pump seals. The reactor coolant pump seal failure model used for NUREG-1150 plants was similar to that used in GI-23 risk assessment studies for loss-of-seal-cooling events. Both models contained multiple-path event trees that represented success/failure paths for a three-stage seal. These success/failure paths contained both secondary seal (O-ring, channel seal) failure modes and face seal hydraulic failure modes.

Normal Operating Conditions

Under normal operating conditions, a comparison of CDF resulting from RCP seal LOCAs during normal operations (i.e., as a small-break LOCA initiating event) is made between GI-23 and NUREG-1150 results. The GI-23 value (1.63 E-05) is close to both Sequoyah and Zion values (2.00 E-05 and 1.00 E-05) but is significantly different from the Surry value (6.30 E-07). The low Surry value for CDF due to seal LOCA under normal operation is due to the lower probability of high-pressure injection (HPI) failure because its HPI system is cross-connected to the other unit. That is, each unit can mitigate an RCP seal LOCA in the other unit.

Off-Normal Conditions

For the NUREG-1150 plants, the overall CDF due to station blackout (SBO) is fairly consistent for Surry and Sequoyah with Zion somewhat lower. The CDF values due to SBO seal failures for Surry (8.60 E-06) and Sequoyah (4.32 E-06) are close to the GI-23 estimate (5.60 E-06). The portions of CDF from SBO that are due to seal failure are proportionally consistent except for Zion, which has a very low contribution (ratio of $CDF_{SBO(Seals)}/CDF_{SBO}$ for Surry = 32%, Sequoyah = 37%, and Zion = 6%).

Zion has a very high contribution to total CDF from other loss-of-seal-cooling conditions, i.e., seal LOCAs due to loss of CCW and SW. Surry and Sequoyah have a very low contribution from these causes. For GI-23 the CDF due to seal LOCA from the loss of CCW is somewhere between the very low values of Surry and Sequoyah and the high number for Zion. The Zion number indicates a significant contribution from CCW failure. Zion also shows an equally significant contribution from SW, which was not specifically considered in the GI-23 studies. It should be noted that the NUREG-1150 studies considered potential passive failures in the CCW and SW systems. For Zion, it was found that the certain failures could not be isolated. This led to the high CDF values shown in Table D-1.

Conclusion

The GI-23 CDF results for normal operation and SBO are in fairly good agreement with the NUREG-1150 results. In the case of the results for CCW, the GI-23 study has shown that seal LOCAs resulting from loss of CCW are very plant specific because CCW designs and the resulting seal cooling dependencies vary widely. The NUREG-1150 results seem to support this conclusion. As a result of the possible wide variations in the benefits from a fix for CCW/SW, GI-23 pursued a fix that could solve all the dependencies (i.e., an independent seal cooling system) at a reasonable cost. The cost/benefit ratio was then calculated assuming only a benefit of fixing the potential SBO portion (See Appendix C).

Table D-1, Core Damage Frequency (CDF) per Reactor-Year

SELECTED PLANTS:	<------- NUREG-1150 * ------> SURRY	SEQUOYAH	ZION	GI-23 76 PWRs
NORMAL OPERATING CONDITIONS				
CDF DUE TO SEAL LOCA	6.30 E-07	2.00 E-05	1.00 E-05	1.63 E-05 (Note 1)
OFF-NORMAL CONDITIONS				
TOTAL CDF (INTERNAL EVENTS)	4.00 E-05	5.72 E-05	3.40 E-04	NOT CALCULATED
CDF DUE TO SBO	2.72 E-05	1.16 E-05	6.50 E-06	NOT CALCULATED
CDF DUE TO SEAL LOCA FROM SBO	8.60 E-06	4.32 E-06	4.00 E-07 (Note 2)	5.60 E-06 NUREG/CR-5167)
CDF DUE TO SEAL LOCA FROM LOSS OF CCW	NEGLIGIBLE (Note 3)	<1.0 E-08 (Note 5)	1.47 E-04 (Note 6)	6.00 E-06 (Note 7)
CDF DUE TO SEAL LOCA FROM LOSS OF SW	NEGLIGIBLE (Note 4)	<1.0 E-08 (Note 5)	1.46 E-04 (Note 6)	NOT CALCULATED

Notes:

1. This is calculated from one W, one CE, and one B&W plant (Indian Point-3, Calvert Cliffs-1, and Arkansas-1, NUREG/CR-4400).

2. Zion has SW and CCW systems cross-connected between two units.

3. At Surry, RCP seal injection does not fail on loss of CCW; it was therefore determined to be insignificant as a separate initiator.

4. Surry has a very reliable (gravity fed from a canal) SW system that is assumed not to fail.

5. At Sequoyah, the frequency of loss of SW is assessed to be very low and not included as an initiating event. The CDF due to loss of CCW is assessed to be very low because HPI and charging pumps are cooled by reliable SW.

6. This is driven by specific nonrecoverable passive failures.

7. This is derived from a sensitivity study (NUREG/CR-4643) performed on one plant.

* The Division of Systems Research, Probabilistic Risk Analysis Branch, (Arthur Buslik) assisted in deriving the NUREG-1150 values.

APPENDIX E

BACKFIT ANALYSIS (REF. 10 CFR 50.109) FOR GENERIC ISSUE 23

E.1 INTRODUCTION

This appendix presents the backfit analysis for Generic Issue 23 (GI-23), "Reactor Coolant Pump Seal Failure." The technical findings for GI-23 are presented in NUREG/CR-4948, and the cost/benefit analyses are presented in NUREG/CR-5167. These studies indicate that reactor coolant pump (RCP) seal failures are significant contributors to the overall plant risk. As a consequence of these technical findings and based on the cost/benefit analyses performed, the staff is proposing that the licensees should (1) treat the RCP seal assembly as an item performing a safety-related function similar to other components of the reactor coolant pressure boundary, applying quality assurance requirements consistent with Appendix B to 10 CFR 50 and applicable General Design Criteria of Appendix A, (2) provide RCP-manufacturer-recommended instrumentation and instructions for monitoring RCP seal performance and detecting incipient RCP seal failures, and (3) provide RCP seal cooling during off-normal plant conditions involving loss of all seal cooling such as station blackout. Note that the first two items are designed to reduce the probability of seal failure during normal plant operations whereas the last item is intended to reduce the probability of seal failure during off-normal plant conditions such as station blackout, loss of component cooling water (CCW), or loss of service water (SW).

The estimated benefit from implementing the three proposed items is a reduction in the core damage frequency per reactor-year and a reduction in the associated risk of off-site radioactive releases due to RCP seal failure. The risk reduction to the public for 76 operating PWRs over an estimated average lifetime of 25 years is estimated to be 114,000 person-rem (best-estimate numbers used), which supports the conclusion that the proposed items provide a substantial increase in the overall protection of the public health and safety. Also the direct and indirect costs of implementation are justified in view of this increased protection.

The total risk reduction value given here takes into account some overlapping benefits that result from combined implementation of Items 1 and 2, both of which address seal failures during normal operating conditions.

The implementation cost per plant for licensees to comply with the three items is estimated at $1,165,000 ($317,000 for Items 1 and 2 combined and $848,000 for Item 3). The industry operation cost per plant for the three items is estimated at a savings of $281,000 ($294,000 savings per plant for Items 1 and 2 combined and $13,000 cost per plant for Item 3; recall that savings represent negative cost). Over the lifetime of the 76 plants, the best-estimate industry cost for implementing all three items is estimated to be $88.5 million (present value), while the industry savings for operation is estimated to be $21.3 million (present value). The net industry cost for implementation and operation is therefore estimated to be $67.2 million (present value). However, the expected reduction in RCP seal failure is estimated to result in substantial net savings in the area of onsite property costs with $2.9 million for Items 1 and 2 combined and $5.6 million for Item 3. Detailed cost/benefit worksheets are presented in Appendix F of NUREG/CR-5167.

The cost/benefit ratio associated with Item 1 is predominantly negative as a result of the large savings realized by the reduced frequency of seal replacements. The cost/benefit ratio of Item 1 is less than the $1000/person-rem guideline; therefore this value supports the implementation of Item 1. The best-estimate cost/benefit ratio for Item 2 is $394/person-rem and is favorable based on the $1000/ person-rem guidance. It should be noted that the cost/benefit ratio associated with Item 2 assumes that all 76 PWRs will install seal instrumentation. However, many plants currently have sufficient instrumentation installed for monitoring seal performance and therefore will have reduced installation costs, which results in a more favorable cost/benefit ratio. Since benefits realized from implementing Item 1 may overlap those from implementing Item 2, the best-estimate cost/benefit ratio for Items 1 and 2 combined is $6/person-rem. The best-estimate cost/benefit ratio for Item 3 is $958/person-rem. Although there is considerable uncertainty in the cost/benefit analysis of Item 3 because of uncertainties in core damage estimates resulting from seal failure assumptions, release consequence uncertainties, and various blackout probability models, it remains the only viable means of precluding common-cause temperature-related seal failures under loss-of-all-seal-cooling conditions and ensures compatibility with the proposed resolution of USI A-44.

The referenced cost/benefit analysis assumes an average remaining plant life of 25 years. It does not take into account the potential for increased plant life as the result of plant license renewal. If it did, the cost/benefit ratio would decrease nearly linearly with the period of life extension. This occurs because the benefit increases linearly with time and the major element of cost is the initial cost, which would be unchanged. Only the operating costs (both industry and NRC) would increase with time. For example, if the cost/benefit analysis for Item 3 assumed a 20-year license renewal extension in addition to the typical 25-year remaining life, the cost/benefit ratio would decrease from $958/person-rem to about $600/person-rem.

The referenced quantitative cost/benefit analysis was one of the factors considered in evaluating the proposed resolution; however, other factors also played a part in the decision-making process. Probabilistic risk assessment (PRA) analyses indicate that the overall probability of core damage due to postulated small breaks could be dominated by such events as RCP seal failures. PRA analyses indicate that, for some PWRs, the annual core damage frequency due to mechanical- and maintenance-induced RCP seal failures may be as high as 19 percent of the total core damage risk from all causes.

The conclusion of this backfit analysis is that a substantial increase in the protection of the public health and safety will be derived from backfitting the RCP seal improvements and that the backfit is justified in view of the favorable cost/benefit ratios. In the following sections of this backfit analysis, the nine factors stipulated by 10 CFR 50.109(c) to be used in the determination of backfitting are addressed individually for each of the items. Items 1 and 2 have an overlapping benefit when implemented together and are addressed as a combined item in Section E.4. They are also addressed individually in Sections E.2 and E.3 to show what the results would be if one but not both were implemented.

E.2 ANALYSIS OF 10 CFR 50.109(c) FACTORS FOR ITEM 1

E.2.1 Objective

The objective of Item 1 of the proposed backfit is to improve the performance of the RCP seal assembly by treating it as an item performing a safety-related function similar to other reactor coolant pressure boundary components, applying quality assurance requirements consistent with Appendix B to 10 CFR 50 and applicable General Design Criteria of Appendix A. This backfit will be applicable to all existing PWR plants.

E.2.2 Licensee Activity

To implement Item 1, each PWR licensee would include the RCP seals in the program of Appendix B to 10 CFR 50. This will mean a tighter system of quality control in the following areas:

1. Increased control over materials and fabrication methods used for the manufacture of RCP seals to improve the seal performance.

2. Increased control over the installation and maintenance of the RCP seals to help maintain seal integrity. This control will be realized through the use of detailed procedures for RCP seal installation and maintenance.

3. Increased procedural control over the operation of the RCPs to be consistent with manufacturer's specifications, particularly during startup and shutdown when the seals are most susceptible to damage.

E.2.3 Public Risk Reduction

Backfitting in accordance with the provisions of Item 1 will yield a reduction in public risk from the accidental off-site release of radioactive materials of approximately 26,000 person-rem (best estimate) for 76 operating PWRs with an average remaining life of 25 years. Item 1 will also reduce CDF/reactor-year by about 18 percent from 1.63E-05 to 1.34E-05.

E.2.4 Occupational Exposure

For the average remaining lifetime (25 years) of 76 operating PWRs, the best-estimate increase in the radiological exposure of facility employees is 532 person-rem because of the increased maintenance and installation time, and the best-estimate decrease is 117 person-rem because of fewer accidental exposures.

E.2.5 Installation/Operation Costs

The installation and operation costs associated with the adoption of Item 1 are:

E.2.5.1 Installation

Labor costs associated with the implementation of Item 1 range from $1.9 million to $5.7 million with a best estimate of $3.8 million for 76 operating PWRs. No additional plant downtime or capital costs are anticipated.

E.2.5.2 Operation

Implementation of Item 1 is estimated to result in net savings for 76 operating PWRs over an average lifetime of 25 years of $13 million (present value) in industry operation because fewer seal replacements will be required per plant outage.

Thus the total industry cost is the algebraic sum of installation (implementation) and operation costs. For Item 1, the net result is a total industry saving of approximately $9.4 million.

E.2.6 Potential Safety Impact

The added operational complexity due to the implementation of Item 1 will not have an adverse impact on plant safety, but it will improve RCP seal performance.

E.2.7 NRC Costs

NRC costs associated with Item 1 are estimated to be:

E.2.7.1 Development

Development of the proposed resolution (Item 1) is estimated at $370,000 (best estimate) for 76 operating PWRs.

E.2.7.2 Implementation

Implementation of Item 1 is estimated at $37,000 (best estimate) for 76 operating PWRs. This estimate assumes minimal resources for review of the responses to the generic letter.

E.2.7.3 **Operation**

NRC inspection costs associated with Item 1 are estimated at $336,000 (present value) for 76 PWRs over an average remaining lifetime of 25 years.

The total NRC cost is the sum of the development, implementation, and operation costs. For Item 1, the total NRC cost is approximately $743,000.

E.2.8 <u>Facility Differences</u>

Item 1 is applicable to all PWR plants regardless of design or age. BWR plants are not included under the resolution of GI-23 because the potential for core damage in BWRs due to recirculation pump seal failure is much less than that for PWRs.

E.2.9 <u>Term of Requirement</u>

The RCP seal resolution is the final resolution of GI-23; it is not an interim measure.

E.3 ANALYSIS OF 10 CFR 50.109(c) FACTORS FOR ITEM 2

E.3.1 <u>Objective</u>

The objective of Item 2 of the proposed backfit is to provide RCP manufacturers' recommended instrumentation and instructions for monitoring RCP seal performance and detecting seal failures. This backfit will be applicable to all existing PWR plants.

E.3.2 <u>Licensee Activity</u>

To implement Item 2, each PWR licensee would provide sufficient instrumentation for monitoring RCP seal performance. This equipment is to be based on RCP manufacturers' recommendations and instructions.

E.3.3 <u>Public Risk Reduction</u>

Backfitting in accordance with the provisions of Item 2 will yield a reduction in public risk from the accidental off-site release of radioactive materials of approximately 33,000 person-rem (best estimate) for 76 operating PWRs with an average remaining life of 25 years. Item 2 will also reduce CDF/reactor-year by about 23 percent from 1.63E-05 to 1.26E-05.

E.3.4 Occupational Exposure

For the average remaining lifetime (25 years) of 76 operating PWRs, the best-estimate increase in the radiological exposure of facility employees is 1569 person-rem because of the increased maintenance and installation time and the best-estimate decrease is 150 person-rem because of fewer accidental exposures.

E.3.5 Installation/Operation Costs

The installation and operation costs associated with the adoption of Item 2 are:

E.3.5.1 Installation

For 76 operating PWRs, the total costs associated with Item 2 range between $19.6 and $20.9 million with a best estimate of $20.3 million. This includes $250,000 in capital costs and approximately $17,000 in labor costs per plant. No additional plant downtime is anticipated.

E.3.5.2 Operation

Implementation of this item is estimated to result in net savings for 76 operating PWRs over an average lifetime of 25 years of $6.5 million (present value) in industry operation because fewer seal replacements will be required per plant outage.

Thus the total industry cost is the algebraic sum of installation (implementation) and operation costs. For Item 2, the net result is a total industry cost of approximately $13.8 million.

E.3.6 Potential Safety Impact

The additional plant and operational complexity resulting from implementation of this item will not have an adverse impact on plant safety.

E.3.7 NRC Costs

NRC costs associated with Item 2 are estimated to be:

E.3.7.1 Development

Development of the proposed resolution (Item 2) is estimated at $370,000 (best estimate) for 76 operating PWRs.

E.3.7.2 Implementation

Implementation of Item 2 is estimated at $37,000 (best estimate) for 76 operating PWRs. This estimate assumes minimal resources for review of the responses to the generic letter.

E.3.7.3 Operation

NRC inspection costs associated with Item 2 are estimated at $336,000 (present value) for 76 PWRs over an average remaining lifetime of 25 years.

The total NRC cost is the sum of the development, implementation, and operation costs. For Item 2, the total NRC cost is approximately $743,000.

E.3.8 Facility Differences

Item 2 is applicable to all PWR plants regardless of design or age. BWR plants are not included under the resolution of GI-23 because the potential for core damage in BWRs due to recirculation pump seal failure is much less than that for PWRs. The installation costs associated with Item 2 assume that the plant presently has no RCP seal monitoring equipment installed. However, many plants currently have sufficient instrumentation installed for monitoring RCP seal performance and will therefore have reduced installation costs.

E.3.9 Term of Requirement

The RCP seal resolution is the final resolution of GI-23; it is not an interim measure.

E.4 ANALYSIS OF 10 CFR 50.109(c) FACTORS FOR ITEMS 1 AND 2 COMBINED

E.4.1 Objective

The objective of Items 1 and 2 of the proposed backfit is to reduce the probability of core damage due to RCP seal failure during normal plant operations. Because the benefits received from implementation of one item may overlap benefits from the other item, the items were examined jointly under normal seal cooling conditions. This backfit will be applicable to all existing PWR plants.

E.4.2 Licensee Activity

To implement Items 1 and 2, each PWR licensee would (1) include the RCP seals in the program of Appendix B to 10 CFR 50 and (2) provide RCP-manufacturer-recommended instrumentation and instructions for monitoring RCP seal performance.

E.4.3 Public Risk Reduction

Backfitting in accordance with the provisions of Items 1 and 2 combined will yield a reduction in public risk from the accidental off-site release of radioactive materials of approximately 51,000 person-rem (best estimate) for 76 operating PWRs with an average remaining life of 25 years. Item 1 and 2 combined will also reduce CDF/reactor-year by about 35 percent from 1.63E-05 to 1.06E-05.

E.4.4 Occupational Exposure

For 76 operating PWRs, the best-estimate decrease in the radiological exposure of facility employees for the average remaining lifetime of the plants is 179 person-rem because of a decrease in the total number of seals changed per plant outage. An additional decrease in occupational exposure associated with seal-related accidents is 228 person-rem. The reduction in the average number of seals changed per outage after implementation of an item is the controlling factor in estimating occupational exposure. Implementation of Items 1 and 2 combined results in a large enough decrease in the number of seals changed per outage to achieve a net decrease in occupational exposure instead of the increase seen in implementing Items 1 and 2 individually.

E.4.5 Installation/Operation Costs

The installation and operation costs associated with the adoption of Items 1 and 2 are:

E.4.5.1 Installation

Labor costs associated with the implementation of these items range from $21.5 million to $26.6 million with a best-estimate of $24.0 million for 76 operating PWRs. No additional plant downtime or capital costs are anticipated.

E.4.5.2 Operation

Implementation of these items is estimated to result in savings for 76 operating PWRs over an average lifetime of 25 years of $22.3 million (present value) in industry operation because fewer seal replacements will be required per plant outage. As with occupational exposure, operating costs are driven by the number of seals changed after implementing an item. Appendices E and F to NUREG/CR-5167 provide additional information.

Thus the total industry cost is the algebraic sum of installation (implementation) and operation costs. For Items 1 and 2 combined, the net result is a total industry cost of approximately $1.7 million.

E.4.6 Potential Safety Impact

The additional plant and operational complexity resulting from implementation of these items will not adversely affect plant safety.

E.4.7 NRC Costs

NRC costs associated with Items 1 and 2 combined are estimated to be:

E.4.7.1 Development

Development of the proposed resolution (Items 1 and 2 combined) is estimated at $740,000 (best estimate) for 76 operating PWRs.

E.4.7.2 Implementation

Implementation of Items 1 and 2 combined is estimated at $74,000 (best estimate) for 76 operating PWRs. This estimate assumes minimal resources for review of responses to the generic letter.

E.4.7.3 Operation

NRC inspection costs associated with Items 1 and 2 combined are estimated at $672,000 (present value) for 76 PWRs over an average remaining lifetime of 25 years.

The total NRC cost is the sum of the development, implementation, and operation costs. For Items 1 and 2 combined, the total NRC cost is approximately $1.49 million.

E.4.8 Facility Differences

Items 1 and 2 combined are applicable to all PWR plants regardless of design or age. BWR plants are not included under the resolution of GI-23 because the potential for core damage in BWRs due to recirculation pump seal failure is much less than that for PWRs.

E.4.9 Term of Requirement

The RCP seal resolution is the final resolution of GI-23; it is not an interim measure.

E.5 ANALYSIS OF 10 CFR 50.109(c) FACTORS FOR ITEM 3

E.5.1 Objective

The objective of Item 3 of the proposed backfit is to provide RCP seal cooling during off-normal plant conditions involving loss of all seal cooling such as station blackout. This backfit will be applicable to all existing PWR plants.

E.5.2 Licensee Activity

To implement Item 3, each PWR licensee would provide for RCP seal cooling during off-normal plant conditions such as station blackout as follows:

1. Initiate cooling flow to the RCP seals in accordance with the specifications used for normal operation within 10 minutes[1] from the start of the off-normal plant conditions such as station blackout.

2. Maintain the cooling flow to keep RCP seal temperatures within specified design limits for the duration of the off-normal condition.

E.5.3 Public Risk Reduction

Backfitting in accordance with the provisions of Item 3 will yield a reduction in public risk from the accidental off-site release of radioactive materials of 63,000 person-rem (best estimate) for 76 operating PWRs with an average remaining life of 25 years. Item 3 will also reduce CDF/reactor-year by about 95 percent from 1.16E-05 to 6.00E-07.

E.5.4 Occupational Exposure

For 76 operating PWRs, the best-estimate total decrease in the radiological exposure of facility employees for the average remaining lifetime of the plants is estimated to be 440 person-rem because of fewer accidental exposures. The radiological operational exposure is negligible.

[1]Analysis has shown that leakage through the RCP seals will remain subcooled during the first 10 minutes. However, after 10 minutes the bleedoff of the remaining cooling water and subsequent heatup of the pump internals can lead to two-phase flow conditions, increased leakage, and possible failure of the RCP seals.

E.5.5 Installation/Operation Costs

The installation and operation costs associated with the adoption of Item 3 are:

E.5.5.1 Installation

Assuming all 76 operating PWRs include provisions for seal cooling during off-normal conditions, the best-estimate total cost (labor and capital) associated with the implementation of Item 3 is $64.5 million ($848,000 per plant). This assumes meeting the intent of Item 3 by (1) installing an independent power source to provide at least one mode of seal cooling (seal injection or thermal barrier cooling) to the RCP seals and (2) performing plant modifications to allow backup cooling of the makeup pump from an existing plant water system for those plants that have a potential vulnerability to loss of seal cooling from conditions other than station blackout. No additional plant downtime is expected.

E.5.5.2 Operation

The best-estimate operation cost associated with Item 3 for the 76 operating PWRs is $1 million. This amount is essentially negligible in the overall total industry cost calculations.

Thus the total industry cost is the algebraic sum of installation (implementation) and operation costs. For Item 3, the net result is a total industry cost of approximately $65.5 million.

E.5.6 Potential Safety Impact

The changes to the plant and the added operational complexity due to implementation of this item will have no adverse impact on plant safety. The RCP seal resolution is closely related to USI A-44, Station Blackout, and is compatible with the rule developed therein.

E.5.7 NRC Costs

NRC costs associated with Item 3 are estimated to be:

E.5.7.1 Development

Development of the proposed resolution (Item 3) is estimated at $370,000 (best estimate) for 76 operating PWRs.

E.5.7.2 Implementation

Implementation of Item 3 is estimated at $37,000 (best estimate) for 76 operating PWRs. This estimate assumes minimal resources for review of the generic letter responses.

E.5.7.3 Operation

NRC inspection costs associated with Item 3 are estimated at $336,000 (present value) for 76 PWRs over an average remaining lifetime of 25 years.

The total NRC cost is the sum of the development, implementation, and operation costs. For Item 3, the total NRC cost is approximately $743,000.

E.5.8 Facility Differences

Item 3 is applicable to all PWR plants regardless of design or age. BWR plants are not included under the resolution of GI-23 because the potential for core damage in BWRs due to recirculation pump seal failure is much less than that for PWRs. However, because of plant-specific variations, four older BWRs (Millstone 1, Oyster Creek, Nine Mile Point 1, and Big Rock Point) may not have sufficient makeup capability during SBO; therefore these plants will be further evaluated.

E.5.9 Term of Requirement

The RCP seal resolution is the final resolution of GI-23; it is not an interim measure.

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This report presents the regulatory/backfit analysis for Generic Issue 23 (GI-23), "Reactor Coolant Pump Seal Failure." The regulatory analysis includes quality assurance provisions for reactor coolant pump seals, instrumentation and procedures for monitoring seal performance, and provisions for seal cooling during off-normal plant conditions involving loss of all seal cooling such as station blackout.

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