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The Impact of Mechanical- and Maintenance-Induced Failures of Main Reactor Coolant Pump Seals on Plant Safety

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EXECUTIVE SUMMARY

This document presents an investigation of the safety impact resulting from mechanical- and maintenance-induced reactor coolant pump (RCP) seal failures in nuclear power plants. The RCP seal failures that have occurred to date, have not resulted in a direct threat to health and safety of the public. However, the potential does exist for seal failures with significant safety consequences if the mitigating systems do not respond. Several seal failures have occurred in which the loss of primary coolant to the containment exceeded the normal makeup capacity of the plant. The intention of this document is to estimate the annual frequency for the spectrum of leak rates induced by RCP seal failures and their impact on plant safety.

A data survey of the pump seal failure for existing nuclear power plants in the U.S. from several available sources was performed. Special care was given to combining the data from various sources in order to obtain a complete set of RCP seal failure data without a miss or repetition. The data were then divided into several populations based on the plant vendor and pump designer. Further classifications of data, accounting for the design of the seal injection system or the size and model number of the pump, were judged unwarranted because of the reduction of the number of observed data in each population. The annual frequency of the pump seal failures in a nuclear power plant, taking into account the possibility of common mode failures, was estimated by means of the concepts of hazard rate and dependency evaluation. The conditional probability distribution of various sizes of leak rates, given a RCP seal failure, was then evaluated by fitting the leak rate data from the combined population (all the pressurized water reactors) into a Weibull distribution. Finally, these results were integrated to provide the annual exceedence frequency for various sized leak rates resulting from mechanical- and maintenance-induced RCP seal failures in nuclear power plants.

The safety impact of mechanical- and maintenance-induced RCP seal failures was measured by estimating their contribution to core-melt frequency. Two categories of RCP seal leakages were considered, depending on the size of the leak rates and how they compared to the normal makeup capacity of the plant. For leak rates below the normal makeup capacity, the impact on the plant safety was discussed qualitatively, whereas for leak rates above the normal makeup capacity, formal PRA methodologies were applied. The study performed was limited to three nuclear power plants, namely, Arkansas Nuclear One Unit 1 (ANO-1), Calvert Cliffs Unit 1 (CC-1), and Indian Point Unit 3 (IP-3). These plants were selected with the intent of providing some insights for various plant vendors.

In Section 1, the importance of RCP seal failure in nuclear power plants and the program outline (Phases I and II) of this project are briefly discussed. The deficiencies in various data banks and the assumptions used to compensate for these deficiencies are also discussed.

In Section 2, an overview of some of the statistical analyses used in the study is given. The treatment of dependency is briefly discussed, and the maximum likelihood estimates for the dependency factors and failure intensities are derived.

In Section 3, the life distribution of RCP seals and the conditional leak rate distributions, given a RCP seal failure, are determined. The contributions of various root causes and estimates for the dependency factors and the failure intensity for the different combinations of pump designers and plant vendors are determined. This information is then translated into an exceedence frequency of various sized primary coolant leak rates through the RCP seals.

Section 4 summarizes the results and discusses some of the insights gained from the first stage of this study.

In Section 5, an overview of the methodologies used in Phase II of this study is given. The potential safety impact of mechanical- and maintenance-induced RCP seal failures leading to leak rates below the normal makeup capacity for the three representative nuclear power plants is discussed. The scenarios of core melt initiated by RCP leakages beyond the normal make up capacity are also discussed. The quantification of core-melt frequency initiated by mechanical- and maintenance-induced RCP seal failures is detailed in Section 6.

Section 7 summarizes the results and discusses the various safety aspects of mechanical- and maintenance-induced RCP seal failures.

In summary, for the PWR plants, a total of 113 events were collected in which 173 RCP seal failures are noted, indicating the failure of more than one pump in some events. Seven of these events resulted in leak rates greater than 25 gallons per minute (gpm). Forty percent of the seal failures were attributed to seal wearout, twenty percent were judged to be maintenance induced, and the remainder were considered to be the result of plant transients, seal surface contamination, etc. The hazard rate (failure intensity per month) for the Combustion Engineering (CE) plants with Byron Jackson (BJ) pumps and Babcock & Wilcox (B&W) plants with BJ pumps are respectively estimated to be 2.2×10^{-2} and 3.6×10^{-2}. The Westinghouse (W) plants with Westinghouse (W) pumps indicated a hazard rate of 1.7×10^{-2}. The estimated hazard rates for B&W with pre- and post-1974 Bingham pumps are 10.2×10^{-2} and 2.0×10^{-2}, respectively. The annual exceedence frequency for RCP leak rates (the annual frequency of RCP seal leak exceeding a certain limit) for various populations of PWR plants with four RCP pumps is also discussed in Section 3.

Similarly, for the boiling water reactor (BWR) plants, a total of 61 reported events involving 72 recirculation pump seal failures were collected. Two of these events which have occurred at the Brunswick Nuclear Power Plant (Unit 2), during August and September 1975, were the only potential candidates for creating a small LOCA. However, the evaluation of the seal problem and the corrective actions taken by the utility have virtually eliminated the possibility of recurrence. In addition, the contribution to core-melt frequency induced by small LOCA in BWRs is generally less than that for PWRs. Therefore, the risk associated with the recirculation pump seal leakage in the BWRs is judged to be negligible. The hazard rate estimated for recirculation pump seal failures in BWRs is about 1.3×10^{-2} per month. Failure attributed to seal wearout was ~42%, while 21% of the failures were judged to be maintenance induced. Harsh environments and plant transients accounted for 23% of the failures, while only 8% were attributed to the seal surface contamination.

The estimated hazard rates of pump seal failures for both PWRs and BWRs have indicated a higher failure rate for those seals with service life in excess of two years. This may indicate a slow degradation of the seals after two years of operation. However, owing to the assumptions of the study and the sparsity of data for seals in service more than two years, any statistical inferences in this regard are highly uncertain.

The safety impact of the mechanical- and maintenance-induced RCP seal leakages in excess of normal makeup capacity of the plant was evaluated through formal PRA methodologies. The results obtained are heavily dependent on the existing PRA documents for the three representative plants. Therefore, it is important to be aware of the assumptions made in performing these PRAs, the level of detail, and differences among them in order to justify the results. The mechanical- and maintenance-induced core-melt frequencies estimated for these three plants, given the vast design/operational differences among the plant and the PRA approaches, are within the range of 0.5E-5 to 2.0E-5 per year. The values for annual core-melt frequency are based on point estimates calculated using the mean values of primary events for three representative plants. Therefore, they reflect the effect of plant to plant variability. The uncertainty associated with RCP seal failure frequency and the associated leak rates are discussed in the body of the report. The uncertainty associated with the unavailabilities of mitigating systems are discussed in reference PRA documents. The dominant accident sequences of small-small LOCAs induced by RCP seal failures are generally caused by failure of High Pressure Recirculation or Injection Systems (HPRS/HPIS). For the CE plants with the four stage seal Byron Jackson RCPs represented here by CC-1 plant, if the failure of the vapor seal is assumed whenever the other three seal stages have failed, the frequency of RCP seal-failure-induced core melt would be comparable to the other two plants. However, if credit is given to the successful operation of the vapor seal to the core-melt frequency caused by RCP seal failures would be much smaller for the CC-1 plant compared to the other two plants.

In conclusion, the safety impact of mechanical- and maintenance-induced RCP seal failures measured in terms of percentage contribution of annual core-melt frequency is estimated to be between 16% and 18% for Babcock and Wilcox (B&W) and Westinghouse (W) plants. For the CE plants with four-stage RCP seals, the percentage increase for annual core-melt frequency would be dependent on the reliability of the vapor seal. If a failure probability of 0.2 is assigned to the vapor seal during exposure to full reactor pressure, the percentage increase of annual core-melt frequency would be 4%. The percentage contribution is defined here as the ratio of RCP seal failure-induced core-melt frequency over the total core-melt frequency of the plant excluding the RCP seal failures. This ratio, if estimted from the original PRAs for representative plants, indicates a range of value between 14% and 26%, which is slightly higher than what is estimated here from refined evaluation.

1.0 INTRODUCTION

This report summarizes the work performed under the contract entitled "Assessment of the Effect of Mechanical- and Maintenance-Induced Reactor Coolant Pump Seal Failures on Risk," FIN A-3771. The purpose of this study is to evaluate the frequency of seal failures for the three major RCP seal vendors (Byron-Jackson, Bingham, and Westinghouse), to determine the spectrum of leak rates expected, and to calculate the frequency of core melt due to mechanical- and maintenance-induced seal failures.

Failure of RCP seals and excessive leakage of primary coolant are known to result from extended loss of seal cooling. Other causes of RCP seal mechanical failure may be excessive vibration, defective parts, introduction of contaminants, improper maintenance and installation, or simply the end of seal life. Leakage through the seals due to mechanical- and maintenance-induced seal failures may be aggravated by the operator's failure to respond properly because of either inadequate instrumentation or lack of proper training and procedures. The RCP seal failures that occurred to date have not resulted in a direct threat to health and safety of the public; however, the potential does exist for seal failures with significant safety consequences. Several seal failures have resulted in a loss of primary coolant to the containment greater than the normal makeup capacity of the plant. Seal failures can, therefore, result in a small LOCA, which may lead to a core melt.

1.1 Background

Reactor Coolant Pump seals limit the leakage of reactor coolant along the pump shaft, directing most this flow back to the Chemical and Volume Control System (CVCS) with the remainder being directed to the reactor coolant drain tanks. In limiting the reactor coolant leakage to containment, the RCP uses multistage seals in series. Although very different, three major RCP seal designs, namely, Bingham, Byron Jackson, and Westinghouse, fall into two generic types: the balanced hydrodynamic seal and the hydrostatic seal. The Byron Jackson and Bingham seals are of the balanced hydrodynamic type, while the Westinghouse seals (Westinghouse pump) use hybrid hydrostatic seals. The Westinghouse seal is hybrid in nature because the first-stage primary seal is hydrostatic, while the second- and third-stage seals are hydrodynamic.

The performance and life characteristics of RCP seals have been the subject of several studies.[1-4] The major goal of these studies was to determine the means for reducing the refueling and maintenance outage times. From these studies, valuable information regarding seal lifetime, causes of seal failures in nuclear power plants, and potential areas for improvement in seal auxiliary systems, seal maintenance, and replacement has been obtained. However, the safety aspects of seal failure and the potential for excessive leakage have received little attention, and hence, the accumulated data and the estimated failure rates from the above studies do not necessarily lend themselves to use in a quantitative probabilistic risk assessment. Rather, they are more appropriate for drawing qualitative conclusions.

Generic Issue 23, "Reactor Coolant Pump Seal Failures," addresses the potential safety consequences of failures in PWR Primary Coolant Pump and BWR recirculating pump seals. To this end, a Probabilistic Risk Assessment (PRA) is needed to determine the core-melt frequency resulting from mechanical failures of the RCP seals.

This study is limited to evaluating the core-melt frequency resulting from mechanical- and maintenance-induced RCP seal failures. The failure of RCP seals due to either extended loss of component cooling system or station blackout is not considered as a part of this study. The program outline and the two separate phases of this study are discussed in the following section.

1.2 Program Outline (Phase I, Phase II)

To assess the frequency of core melt induced by pump seal leakage, two steps are required: 1) to determine the frequency of pump seal failure and the expected spectrum of leak rates, and 2) to evaluate the conditional core-melt probability, given the various sizes of the primary coolant leakage. The approach taken is likewise divided in two separate phases: Phase I of the program deals with the data collection and statistical analysis, while Phase II deals with a probabilistic approach to evaluate core-melt frequency induced by pump seal leakage. A detailed outline for each phase follows.

Phase I

Five available data sources are searched in a survey of pump seal failures in existing U.S. nuclear power plants. The following data sources are used in this study:

1. Nuclear Safety Information Center (NSIC) Files.
2. EG&G Licensee Event Report (LER) Summaries - pumps.[5]
3. Nuclear Power Experience[6] (NPE).
4. Nuclear Plant Reliability Data System (NPRDS).
5. Data Collected for Prioritization of Pump Seal Failures,[7] and EPRI-NP-351.[2]

The collected data for each nuclear power plant are classified in a manner to facilitate the performance of statistical and qualitative inferences on seal life, causes of seal failures, expected leak rate estimation, and common mode and cascade failures for a specific pump design and plant vendor.

The following are three notable deficiencies in the various data banks that may hinder proper statistical analysis:

(a) Some of the reported events do not indicate which pump in a plant has failed. This information is very important for any type of statistics for lifetime estimation.

(b) The age of the seals, namely, the time interval between seal failure and the last seal replacement, is not reported.

(c) In many cases the root causes of seal failures could not be identified. Therefore, the causes reported in various data banks are not reliable. Any statistical inferences for root cause analysis based on the existing data are therefore susceptible to large errors.

To overcome the aforenoted problems, the following steps have been taken:

(a) The events for which the pump units are not identified are discarded for seal life estimation. However, these events have been used for other statistical analyses such as root cause analysis, leak rate, common mode, cascade failure rate estimations, etc.

(b) The age of the seals at the time of failure is determined on the assumption that the seals are replaced, inspected, or refurbished at each refueling outage unless otherwise reported. This means that the seals are considered statistically to be as good as new after each refueling outage. The time to failure for each unit is then determined by taking the smallest time interval between failure time and the last refueling outage or the previous seal replacement. The period of useful life for the RCP seals estimated with this assumption tends to be shorter than the actual seal life.

(c) The root cause analysis, given the lack of the proper data, must rely heavily on the judgment of the engineer reviewing the event. To assure consistency, the following guidelines are recommended.

(i) The failure of RCP seals at early stages of the service cycle (within 2 months) is assumed to be due to maintenance errors during seal replacement, unless otherwise reported.

(ii) The failure of RCP seals within a short period (within 2 months) after a plant transient of loss of offsite power or system failures resulting in inadvertent containment isolation and loss of seal injection or cooling is assumed to be associated with the harsh environment at the seal surface, unless otherwise reported.

(iii) The failure of RCP seals for which no specific cause can be determined are to be categorized as End-of-Life (EOL) failures.

The methodology and the results obtained from statistical evaluations are described in detail in Chapters 3 and 4. In addition, the root causes of seal failures leading to large leakages are determined on the basis of engineering judgments of a small expert group.

Phase II

This phase entails the estimation of the annual core-melt frequencies induced by various-sized LOCAs, caused by RCP seal failures in three representative nuclear power plants: Indian Point 3, Calvert Cliffs 1, and Arkansas Nuclear One Unit 1. The available PRA documentation for these plants seems sufficiently complete to provide a reasonable estimate of core-melt frequency. In addition, a qualitative analysis will be performed to detect the possibility of system interactions that may exist between seal failure with high leakage and other mitigating systems. A flow chart depicting the program outline (integration of Phase I and Phase II) is given in Figure 1.

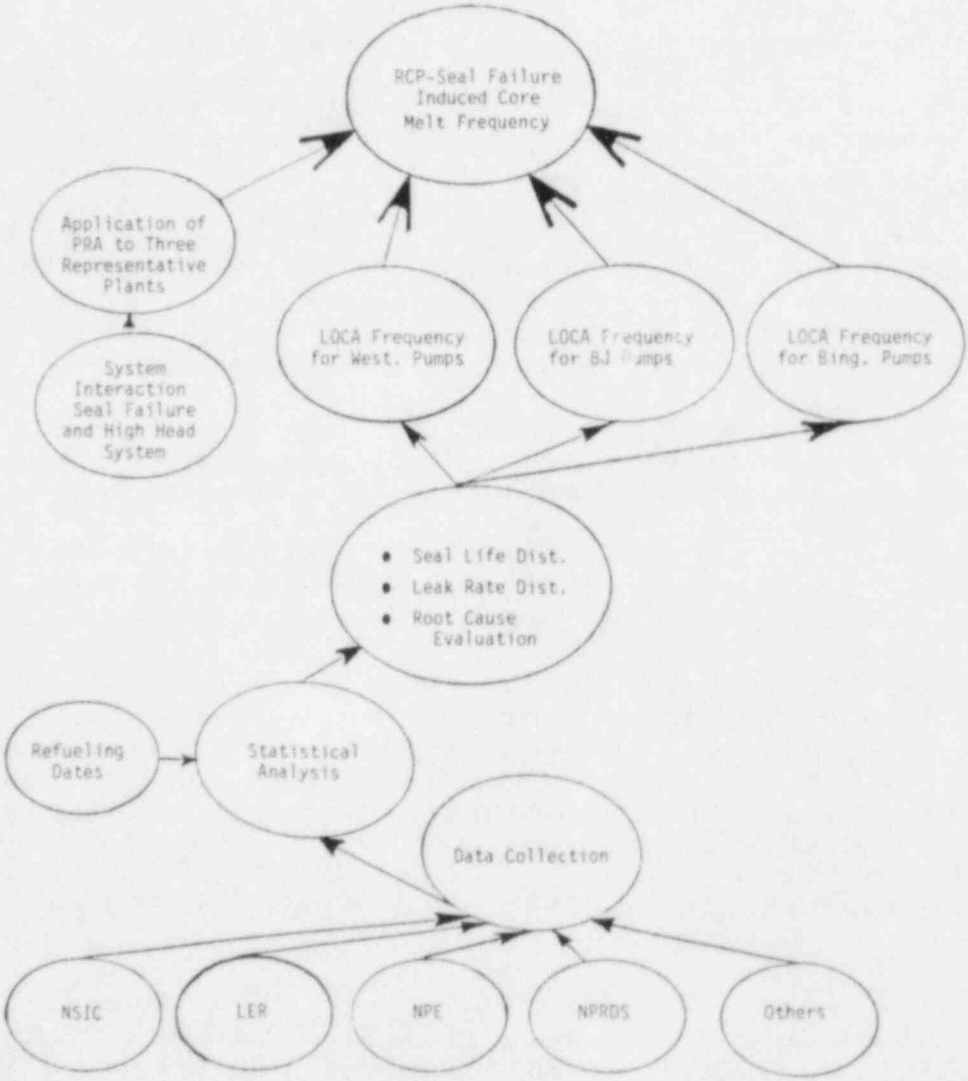

Figure 1. Flow chart diagram layout.

2.0 PHASE I - METHODOLOGY

The main purpose of performing a statistical analysis of these data is to determine 1) the annual frequency of seal failure per unit pump and the expected leakage, 2) the annual frequency of a single or multiple seal failure in a plant with a certain number of pumps (including common cause effect), and 3) the exceedence frequency of primary coolant leakage due to seal failure in a nuclear power plant. The following section describes the type of statistics used in these analyses.

2.1 Statistical Analysis

The method used for statistical analysis is best described by simulating a hypothetical test similar to the following example:

> A number of components are under surveillance. n out of these components have not experienced any failure up to time t. However, by time $t + \Delta t$, m components have failed by some combination of single, double, triple, and quadruple failures. In general
>
> $$m = m_S + 2m_D + 3m_T + 4m_Q \quad , \tag{1}$$
>
> where m_S, m_D, m_T, and m_Q are the number of single, double, triple, and quadruple failures respectively.

Given that $\beta_S P$, $\beta_D P$, $\beta_T P$, and $\beta_Q P$ represent the probabilities for single, double, triple, and quadruple failures within the interval $(t, t + \Delta t)$, the probability of observing the test described above can be expressed by:

$$P(n, m_S, m_D, m_T, m_Q, t, \Delta t) =$$

$$= C(n, m_S, m_D, m_T, m_Q)(\beta_S P)^{m_S}(\beta_D P)^{m_D}(\beta_T P)^{m_T}(\beta_Q P)^{m_Q}(1-P)^{n-m} \quad , \tag{2}$$

where $C(n, m_S, m_D, m_T, m_Q)$ is the possible combinations for selecting m_S singles, m_D doubles, m_T triples, and m_Q quadruples out of n components. It is also assumed that the summation over β's is equal to 1, i.e.,

$$\beta_S + \beta_D + \beta_T + \beta_Q = 1 \quad . \tag{3}$$

Using the maximum likelihood approach, the estimated value for P is

$$P = \frac{m_S + m_D + m_T + m_Q}{(n-m) + m_S + m_D + m_T + m_Q} \quad . \tag{4}$$

Similarly, using Eq. (3) the maximum likelihood estimates for β_S, β_D, β_T, and β_Q are also estimated with the following expressions:

$$\beta_D = m_D/(m_S + m_D + m_T + m_Q) \quad ; \tag{5}$$

$$\beta_T = m_T/(m_S + m_D + m_T + m_Q) \quad ; \tag{6}$$

$$\beta_Q = m_Q/(m_S + m_D + m_T + m_Q) \quad ; \tag{7}$$

$$\beta_S = 1 - \beta_D - \beta_T - \beta_Q = m_S/(m_S + m_D + m_T + m_Q) \quad . \tag{8}$$

The values for P, β_S, β_D, β_T, β_Q are all functions of time, namely, the service life of the component. For the RCP seals, more complexity is added by the degradation of the seals after transients or plant disturbances. Hence, the estimated values of the parameters are not only functions of component service life, but also depend on the number and occurrence times of various transients in the plant. This complexity has not been taken into account owing to sparsity of data and lack of proper existing modelings.

In this study, the parameters of β_S, β_D, β_T, β_Q are considered to be time invariant (constant) and are estimated on the basis of the average of the overall data for each pump and plant vendor. However, the estimated value for P is considered to depend only on the seal lifetime, implying that the number of transients experienced by the seals is proportional to their service lives.

To determine the expected annual frequency for pump seal failures in a plant, assumptions of renewal process and constant hazard rate are incorporated. (The reason for assumption of constant hazard rate is elaborated in Section 3.1.) At present, the uncertainty bounds on frequency of seal failures in a plant cannot be estimated systematically for the aforenoted parameters. Hence, the uncertainty bounds are estimated by propagating the bounds of hazard rates accounting for plant-to-plant variability. A detailed discussion of this process is given in Section 3.3.

2.2 Statistical Analysis for LOCA Initiator

The leak rate distribution is determined using the collected leak rate data reported for the mechanical- and maintenance-induced RCP seal failures. This distribution does not differentiate among the various RCP seal designers, because of the sparsity of leak rate data for each specific RCP vendor. The leak rate distribution (i.e., a two-parameter Weibull distribution), coupled with the annual frequency of seal failures in the plant, is used to determine the exceedance annual frequency for various LOCA sizes. In these calculations, the leak rate associated with simultaneous failures of two pumps is assumed to be twice the leak rate of each pump; namely, the leak rates are assumed to be tightly coupled. The frequency of a certain type of LOCA initiator is then estimated by calculating the expected frequency of leak rates within the range of that type of LOCA. A detailed discussion of the procedure and the associated calculations for estimating LOCA initiator frequencies are also given in Section 3.3.

3.0 PHASE I - EVALUATION

Data are collected on RCP seal failures for existing nuclear power plants in the U.S. from five available sources. The data are classified such that the pertinent information required for RCP seal failure analysis can be easily obtained. The type of information collected from each event is given in Table 1; a sample work sheet is provided in Table 2. The data under surveillance covers the period from July 1969 through May 1984, although each of the five data sources mentioned above does not necessarily cover the entire time period. For example, the NPRDS data source obtained through the NRC extends from January 1974 to May 1984. Special care is taken for combining the data from various sources in order to obtain a complete set of RCP failure data without a miss or repetition. Some existing plants are not included in this study, either because they have operated for only a short time or because of sparsity of data.

For the PWR plants under consideration, 118 reported events were collected. Of these, 46 events are for Westinghouse (W) plants with W pumps; 31 are for Combustion Engineering (CE) plants with Byron Jackson (BJ) pumps; 28 are for Babcock and Wilcox (B&W) plants with BJ pumps; 9 are for B&W plants with old Bingham (pre-1974-2 stage seal design) pumps; and 4 are for B&W plants with new Bingham pumps (post-1974-3 stage seal design). In addition, the following quantitative conclusions are made using the classification system discussed earlier.

1) Root Cause Evaluation: In most cases, the cause of seal failures is determined on the basis of engineering judgment and some insight obtained from reviewing the event report. It is not unusual for multiple causes to be identified for a single event. Five major categories of root causes for seal failures identified in this study are: a) end-of-life seal failure (2)*; b) maintenance, fabrication, and installation errors (1,13); c) temperature and pressure transients (5, 7-9); d) seal contamination with crud or abrasive (6), and other causes as given in Table 1. A pie chart diagram representing the contribution of the aforenoted root causes is shown in Figure 2.

2) Dependency Evaluation: Of 118 reported events, 30 were associated with simultaneous seal failures of two or more pumps. With the methodology discussed in Section 2.1, the values for β_D, β_T, and β_Q for various plant vendors and pump designers have been estimated, and they are given in Table 3. The population of data for new/old Bingham pumps is too small to yield any meaningful statistical values. The lower and upper bounds on the β values are also estimated on the basis of the binomial distribution using the following equations[8]:

$$\beta_U = m_1\, F(\lambda, m_1, m_2)/[m_2 + m_1\, F(\lambda, m_1, m_2)] \quad , \tag{9}$$

---

*The number (2) in parenthesis corresponds to associated causes given in Table 1.

Table 1. Data Classification for RCP Seal Failure Events

CLASSIFICATION

Plant Information:

Name, Date of Criticality, Date of Commercial Operation, Power Level, NSS Vendor, Architect Engineer

Pump Information:

Pump Vendor, Pump Model Number, Number of Pumps, Number of Seal Stages per Pump

Event Information:

Event Date, Plant Status, Primary Failure (System, Component Identification), Primary Subcomponent, Cause of Secondary Failure (System, Component Identification), Secondary Subcomponent, Cause, Leak Rate (gpm), Leak-Total (gal), Event Type, Reference Source

CODIFICATION

Plant Status (Power Level)		Cause		Event Type	
Refueling	(0)	Maintenance Error	(1)	Cascade	(C)
Steady-State Power	(1)	End of Life	(2)	Commonmode	(CM)
Startup & Power Maneuvering	(2)	Vibration	(3)	Others	(NC)
Hot Standby	(3)	Corrosion	(4)		
Hot Shutdown	(4)	Plant Transient	(5)		
Cold Shutdown	(5)	Contamination	(6)		
		Abnormal Pressure Staging	(7)		
		Overheating Seal Cavity	(8)		
		System Disturbances	(9)		
		Operator Error	(10)		
		Improper Venting	(11)		
		Lack of Instrumentation	(12)		
		Defective Parts (Fabrication)	(13)		

Table 2. A Sample Data Sheet

Plant Identification: Calvert Cliffs 1, PWR, CE, BECH, 5/75, 4 RCPs

RCP Identification: BJ, DFSS, 4-Stage Seals

Date and Type of Modification: None

Date of Event/Data Source:	08/22/75, 1
Power Level/Plant Status:	97%, 1
System/Component Failed:	RCP-?, Middle and Upper Seal Failure (Second and Third Stage)
Cause Code:	2
Means for Detection:	Staging Pressure and Flow, High Vapor Seal Leakage
Operator Response:	Manual Shutdown
Automatic Plant Response:	None
System/Component Affected:	None
Cause Code:	
Means for Detection:	
Operator Response:	
Automatic Plant Response:	
Were there Cascade Failures?	No
Leak Rate and Total Leakage:	2.7 gpm, NA
Was the Leak Confined?	Yes
Recovery Action Taken:	Replacement
Comments:	

Table 3. Estimations of the Fractions of Multiple Failures and Their Associated Uncertainty Bounds (5 and 95 percentiles)

Plant/Pump Vendor	5 Percentile $\beta_D(L)$	Point Estimate β_D	95 Percentile $\beta_D(U)$	5 Percentile $\beta_T(L)$	Point Estimate β_T	95 Percentile $\beta_T(U)$	5 Percentile $\beta_Q(L)$	Point Estimate β_Q	95 Percentile $\beta_Q(U)$
W, W	0.08	0.13	0.24	0.008	0.04	0.13	0.06	0.11	0.21
CE, BJ	0.09	0.19	0.34	0.01	0.06	0.19	0.	0.04	0.09
B&W, BJ	0.03	0.11	0.26	0.01	0.07	0.21	0.	0.05	0.1
B&W, Bing*		0.30			0.1			0.23	

*The accumulated data for Bingham pumps are not sufficient to yield meaningful statistical estimations.

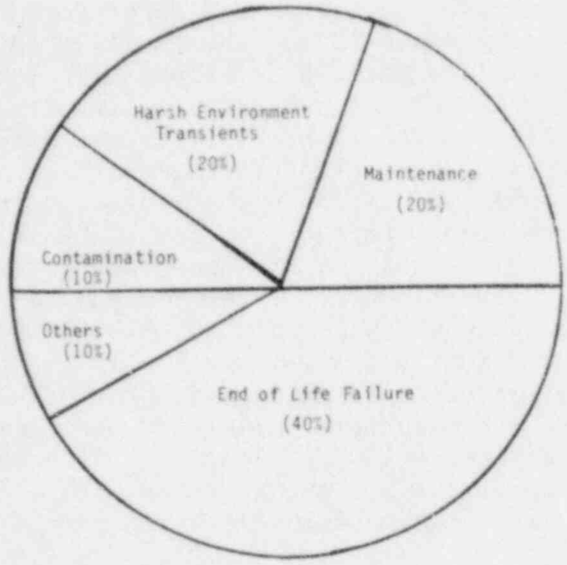

Figure 2. The contributions of various root causes to seal failures in PWRs.

$$\beta_L = m_4/[m_4 + m_3 F(\lambda, m_3, m_4)] \quad , \qquad (10)$$

where $m_1 = 2(x+1)$, $m_2 = 2(n-x)$, $m_3 = 2(n-x+1)$, $m_4 = 2x$, $\lambda = \alpha/2$, n is the total number of trials, x is the number of observed events, $1-\alpha$ is the confidence interval of interest, and F is the Fisher-Snedecor distribution.

In cases where the value of x is zero, the 50 percentile upper bound is considered a conservative estimate of the β value.

3) Scenarios of Seal Failure That May Lead to Excessive Leak Rates: In addition to the statistical inferences made in regard to seal failure, engineering judgment is also relied upon to determine how seal failures may lead to large leakages. These judgments are based on the existing summary event reports for large RCP leakages as given in Appendix A. Identification of the following five important faults as significant for high RCP leak rates is based on the discussion given in Appendix A: a) failure of the operator to isolate leak off line; b) isolation of seal injection to operating pumps; c) undersized thermal barrier heat exchanger; d) failure of operator to reestablish seal cooling in a short period of time; e) failure of other seal stages due to transport of chipped materials from the failed seal stage.

Similarly, for the BWR plants under consideration, a total of 61 reported events were collected. Of these, 49 events are for GE plants with jet pumps, while 12 events are for GE plants without jet pumps (Nine Mile Point and Oyster Creek). A pie chart diagram representing the contribution of various root causes is given in Figure 3. For the BWR plants with two unit pumps, the dependence of seal failure is evaluated by estimating the associated upper bound, best estimate, and lower bound for the value of β_D. These values are

0.08, 0.15, and 0.25, respectively. An evaluation 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. The smallest LOCA considered for the BWRs consists of a seal-failure-induced leak rate beyond the capacity of the makeup water through the control rod drive system (generally between 50 and 60 gpm). The events following seal failures in the Brunswick Plant Unit 2, during August and September 1975, are the only potential candidates for creating a small LOCA. However, the evaluation of the seal problem[2] indicated two basic design deficiencies; one with the pump thermal barrier design; the other with the seal injection system design. It appeared that hot reactor coolant was leaking between the outer diameter of the thermal barrier and the seal cavity because of the warping of the thermal barrier. Consequently, the seals were running too hot. In addition, a malfunction in the injection system following reactor scrams resulted in large demands on the CRD charging pump and, hence a reduction in the discharge pressure below the required level for proper seal injection. In effect, a loss of seal injection water followed every reactor scram. Since corrective actions were taken to eliminate these problems, such failures are expected to be unlikely in the future.

The pump seal reliability curve vs its service life is estimated in Section 3.1 for both PWRs and BWRs. However, further analysis for evaluating pump-seal-induced LOCA frequency is limited to PWR plants. The seal-induced LOCAs for the BWR plants will not be discussed further because of the unlikelihood of such events, as indicated in the data on past operating experience.

3.1 Seal Life Distribution

The time to seal failure must be known in order to evaluate the seal life distribution. This information is collected under the assumption that the

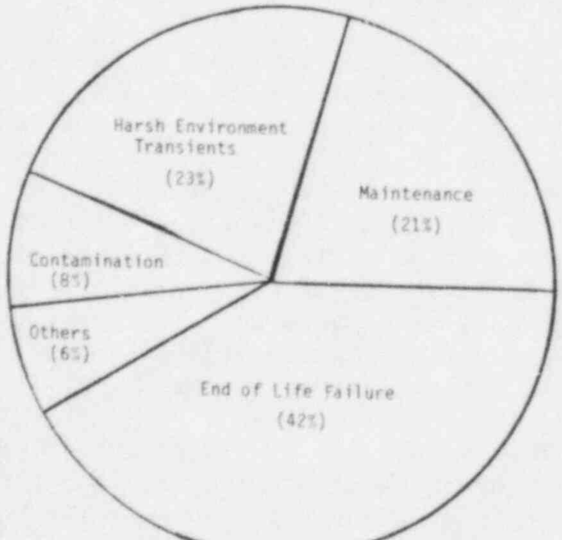

Figure 3. The contributions of various root causes to seal failures in BWRs.

seals are as good as new after each refueling outage. The dates of refueling outages are extracted from the nuclear power plants status reports.[9] These documents span the years from 1977 to 1984.* A data summary for seal failure probability for every two months of its lifetime is given in Table 4. Complementary data for time to seal failure for BJ pumps in CE plants from Reference 10 are also incorporated. For each time step, the maximum likelihood estimate for the P value as discussed in Section 2.1 is also determined. The seal reliability function defined in terms of hazard rate [h(t)] is given by

$$R(t) = e^{-\int_0^t h(t')dt'} , \qquad (11)$$

where h(t) is estimated by the value of P divided by the period of each time step. These values and the corresponding values for R(t) estimated from the data are given in Table 5.

A graphical result depicting R(t) vs time is given in Figure 4 for the various combinations of pump and plant vendors. The reliability curve for the Bingham pumps, both new and old design, is also shown in Figure 4. However, because of insufficient data, the curves are not extrapolated to seal lives longer than 16 months.

An important conclusion that can be made from these reliability curves is:

For the various pump seal designs (excluding Bingham pumps at present) regardless of the plants (PWRs only) in which they are installed, the seal reliability curves tend to indicate two distinct regions, i.e.,

(a) for seal life of less than 20 months for W pumps and less than 24 months for the BJ pumps, the hazard rate of the seals is almost constant, and seal failures are expected to be caused by random failures;

(b) for seal life of greater than 20 months for W pumps and greater than 24 months for the BJ pumps, the hazard rate seems to be time dependent, and the seal failures are the result of slow seal degradations. It shall be noted that the lines shown in Figure 2 for this region are only an approximation and shall not be considered as a constant hazard rate.

Hence, to the best of the knowledge obtained from the statistical analysis on the data, tempered with the sparsity of data for the seal lives greater than 20 months (normal refueling outages are 18 months or less), it is concluded that the existing conventional seals may not have a useful life, without some degradation, of more than 2 years.

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*Some of the microfiche documents for the years 1978 and 1979 were not eligible.

Table 4. Data Summary of Seal Failure Counts vs Seal Age

Pump/Plant Vendors	Seal Life (month) / Failure Counts	2	4	6	8	10	12	14	16	18	20	22	24	26	28	30
W-Pumps W-Plants	1-Singles	5	4	2	5	5	2	1	1	2	0	0	0	0		
	2-Doubles	3	0	2	2	1	0	1	0	1	0	0	0	0		
	3-Triples	0	0	1	0	0	0	0	0	2	1	0	0	0		
	4-Quadruples	1	0	0	0	0	1	0	0	0	0	0	0	0		
	5-Trials	201	186	182	173	162	150	125	105	68	38	8*	8*	8*		
	6-P Value	4.6E-2	7.2E-2	2.8E-2	4.1E-2	3.7E-2	2.0E-2	1.6E-2	9.5E-3	9.8E-2	2.8E-2	8.1 E-2	8.1 E-2	8.1 E-2		
BJ-Pumps CE-Plants	1-Singles	2	3	4	1	4	3	4	5	2	1	1	1	0	2	2
	2-Doubles	0	0	0	1	1	1	0	0	0	0	0	0	0	0	1
	3-Triples	1	0	0	0	0	1	0	0	0	0	0	0	0	0	0
	4-Quadruples	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	5-Trials	108	103	100	96	93	87	72	48	33	20	15	14	6*	6	4
	6-P Value	2.8E-2	3.0E-2	4.0E-2	2.1E-2	5.4E-2	5.9E-2	5.6E-2	10.4E-2	6.1E-2	5.0E-2	6.7E-2	7.1E-2	2.0 E-2	3.5E-1	6.7E-1
BJ-Pumps B&W-Plants	1-Singles	3	3	2	2	1	2	4	1	1	0	0	0	0	0	
	2-Doubles	0	0	1	2	0	1	0	0	0	0	0	0	0	2	
	3-Triples	0	0	0	0	0	0	0	0	0	0	0	1	0	0	
	4-Quadruples	0	0	0	0	0	0	0	0	0	0	0	0	0	0	
	5-Trials	52	49	46	42	38	37	33	23	18	17*	9*	5	5*	5	
	6-P Value	5.7E-2	6.1E-2	6.7E-2	10.E-2	2.6E-2	8.3E-2	12.1E-2	4.4E-2	5.6E-2	7.5 E-2	1.4 E-1	1.7E-1	2.4 E-2	6.7E-1	

Table 4. (Continued)

Pump/Plant Vendors	Seal Life (month) / Failure Counts	2	4	6	8	10	12	14	16	18	20	22	24	26	28	30
Bing.old-pumps B&W Plants	1-Singles	2	0	2	1	0	0									
	2-Doubles	3	0	1	1	0	0									
	3-Triples	0	0	0	0	0	0									
	4-Quadruples	0	1	0	0	0	1									
	5-Trials	23	15	11	7	4	4									
	6-P Value	2.5E-1	8.3E-2	9.1E-2	3.3E-1	2.9* E-1	1.0									
Bing.new-pumps B&W Plants	1-Singles	0	0	0	0	0	0	0	0							
	2-Doubles	0	0	0	0	1	0	1	0							
	3-Triples	0	0	0	0	0	0	0	0							
	4-Quadruples	0	0	0	0	0	0	0	0							
	5-Trials	36	36	36	36	36	30	22	20							
	6-P Value	3.8* E-2	3.8* E-2	3.8* E-2	2.9E-2	4.5* E-2	4.8E-2									

*In Case of no failure, the value of P is calculated based on 50% upper bound.

Table 5. The Hazard Rates and the Estimated Reliability Functions for Various Pumps and Plant Vendors

AGE	W Pumps W Plants		BJ Pumps CE Plants		BJ Pumps B&W Plants		Bing. old Pumps B&W Plants		Bing. new Pumps B&W Plants	
	h(t)	R(t)	h(t)	R(t)	h(t)	R(t)	h(t)	R(t)	h(t)	R(t)
1	2.3E-2	0.955	1.4E-2	0.972	2.9E-2	0.944	1.2E-1	0.787	1.9E-2	0.963
3	1.1E-2	0.934	1.5E-2	0.944	3.0E-2	0.889	4.1E-2	0.725	1.9E-2	0.927
5	1.4E-2	0.908	2.0E-2	0.907	3.3E-2	0.832	4.5E-2	0.662	1.9E-2	0.892
7	2.0E-2	0.873	1.0E-2	0.889	5.0E-2	0.753	1.6E-1	0.481	1.9E-2	0.859
9	1.8E-2	0.842	2.7E-2	0.842	1.3E-2	0.733	1.5E-1	0.356	1.5E-2	0.834
11	1.0E-2	0.825	3.0E-2	0.793	4.1E-2	0.676	5.0E-1	0.131	2.2E-2	0.798
13	0.8E-2	0.812	2.8E-2	0.750	6.0E-2	0.599			2.4E-2	0.760
15	0.5E-2	0.804	5.2E-2	0.676	2.2E-2	0.573				
17	5.0E-2	0.728	3.0E-2	0.636	2.8E-2	0.542				
19	1.4E-2	0.708	2.5E-2	0.605	3.7E-2	0.505				
21	8.1E-2	0.602	3.3E-2	0.567	7.0E-2	0.439				
23	8.1E-2	0.512	3.5E-2	0.528	8.5E-2	0.370				
25	8.1E-2	0.435	10.E-2	0.433	1.2E-1	0.291				
27			1.6E-1	0.314	3.3E-1	0.150				
29			3.3E-1	0.165						

The best estimates for the values of λ (per month) for the first 18 months of seal life are:

$\lambda(W,W) = 1.7 \times 10^{-2}$
$\lambda(B\&W, Bing) = 2.0 \times 10^{-2}$
$\lambda(CE, BJ) = 2.2 \times 10^{-2}$
$\lambda(B\&W, BJ) = 3.6 \times 10^{-2}$
$\lambda[B\&W, Bing(old)] = 10.2 \times 10^{-2}$

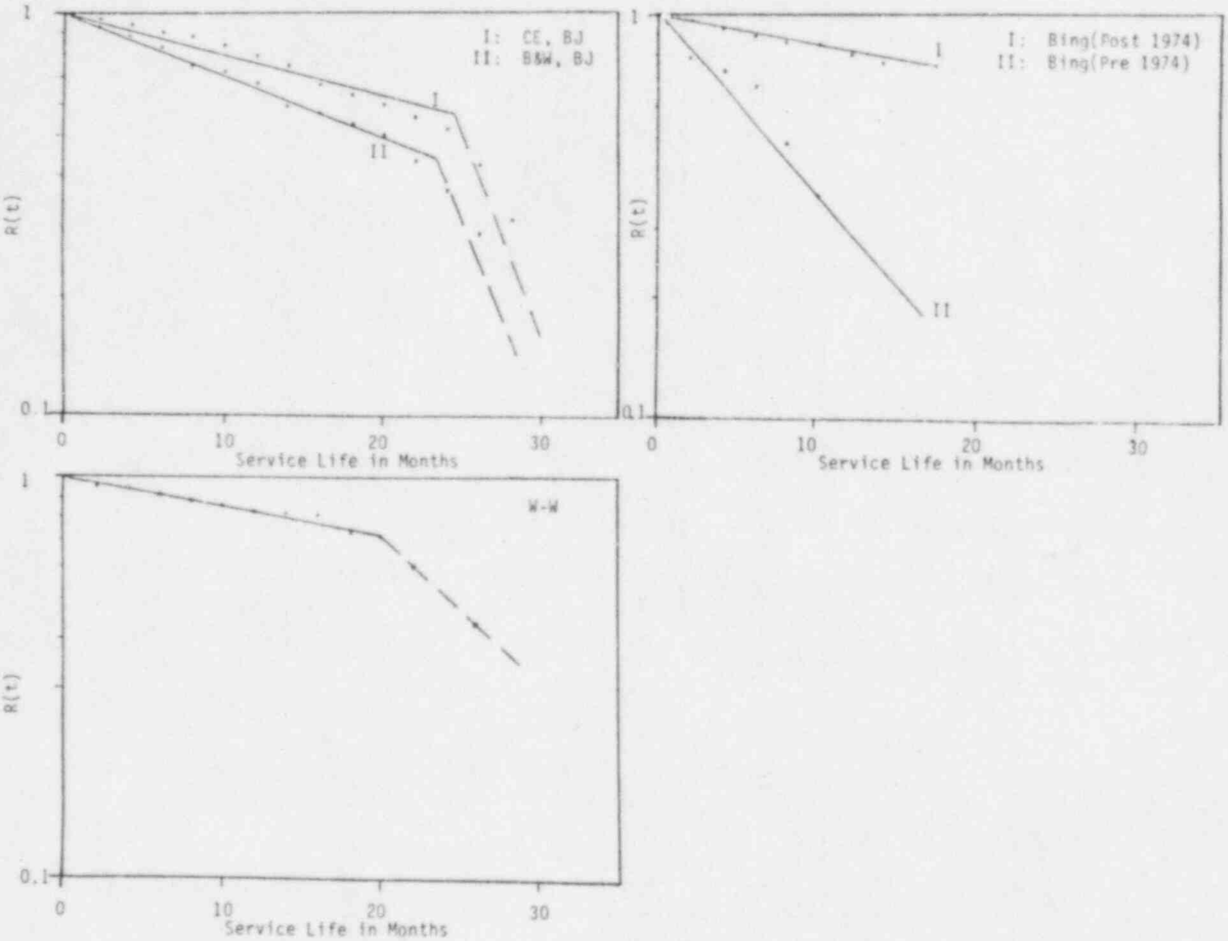

Figure 4. The reliability curves for various RCPs and PWR Plant Vendors.

The hazard rate used for seal failure to estimate the core-melt frequency in the remainder of this study is considered to be constant. The slopes of the graphs in Figure 4 are considered as the best estimate for the seal failure rates (for service life of less than 18 months). These values are given at the bottom of Table 5.

Similarly, for the BWR plants, the values of h(t) and the corresponding values for R(t) estimated from the data are given in Table 6. A graphical result depicting R(t) vs service life of the seals is given in Figure 5. The expected hazard rate of the seals is shown to be almost constant for seal lives less than 18 months, and the seal failures result from random failures. For seal lives greater than 18 months, the hazard rate seems to be time dependent, and the possibility of slow degradation of the seal exists. These results are also tempered with the assumption used in this study which considers the seals to be as good as new after each refueling outage and with the sparsity of data in this region. Therefore, this statistical assertion cannot be confirmed until more accurate data are collected and analyzed.

Table 6. The Failure Counts, Hazard Rate, and the Reliability Function for the BWR Recirculation Pumps

FAILURE COUNTS	SEAL LIFE (MONTH)											
	2	4	6	8	10	12	14	16	18	20	22	24
Singles	3	1	4	2	3	1	1	2	4	2	0	0
Doubles	0	0	0	0	1	1	1	0	0	0	0	0
Total	100	97	96	90	88	64	54	42	26	14	8	4
P†	3.0E-2	1.0E-2	4.2E-2	2.2E-2	4.6E-2	1.6E-2	3.8E-2	4.8E-2	1.5E-1	1.4E-1	8.0E-2*	1.6E-1*
h(t)†	1.5E-2	5.2E-3	2.1E-2	1.1E-2	2.3E-2	7.9E-3	1.9E-2	2.4E-2	7.7E-2	7.1E-2	4.0E-2	8.0E-2
R(t)†	0.97	0.96	0.92	0.90	0.86	0.85	0.82	0.78	0.67	0.58	0.53	0.45

†P(T), h(t), and R(t) stand for the failure intensity per two months, the hazard rate, and the reliability functions respectively.

*In case of no failure, 50 percentile upper bound is used for best estimate.

Note: The best estimate for hazard rate for the service life less than 16 month is 1.4E-2.

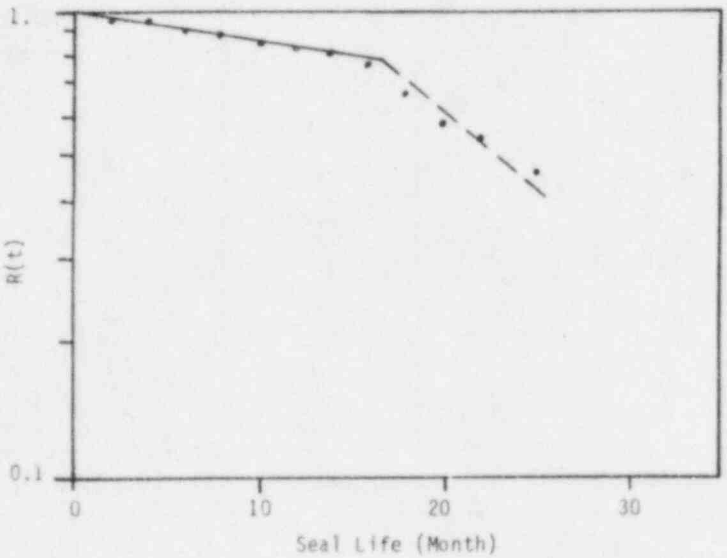

Figure 5. The reliability curve for the BWR recirculation pump seal failure.

3.2 Leak Rate Distribution

Out of a total of 118 reported seal failure events 7 events exhibiting large leakages (greater than 25 gpm) have been identified. The leak rates estimated for the rest of the seal failure events are usually below 25 gpm. Further classification in terms of five leak rate bounds is given in Table 7. These data are fitted to a Weibull distribution using

$$f(LR) = \beta/\eta(x/\eta)^{\beta-1} \exp[-(x/\eta)^{\beta}] \quad , \tag{12}$$

or

$$F(LR) = 1 - \exp[-(x/\eta)^{\beta}] \quad , \tag{13}$$

where f(LR) and F(LR) are the probability density function and the cumulative density function for the leak rates, respectively. The best values for β and η estimated through the graphical representation of data are 0.2 and 0.08, respectively. The exceedance conditional leak rate probability is shown in Figure 6. The uncertainty bounds for β and η have not been quantified because of sparsity of data, especially at the extreme values of leak rates. In addition, the confidence in using the Weibull distribution for presenting the results of observed leak rates is minimal. However, without defining a scenario-specific event starting with the failure of a certain stage of the pump seal in conjunction with the operator actions following the incident, and deterministically evaluating leak rate through the pumps, the above distribution seems sufficient for this preliminary analysis and further complexity is not warranted.

Table 7. The Breakdown of RCP Leak Rate Distribution in Five Cells

RCP VENDOR	LEAK RATES				
	$0 \leq LR \leq 25$	$25 \leq LR \leq 50$	$50 \leq LR \leq 150$	$150 \leq LR \leq 300$	$300 \leq LR \leq 500$
W	$\frac{68}{71}$	$\frac{1}{71}$	$\frac{1}{71}$	$\frac{0}{71}$	$\frac{1}{71}$
BJ	$\frac{73}{76}$	$\frac{2}{76}$	$\frac{0}{76}$	$\frac{0}{76}$	$\frac{1}{76}$
Bing(old)	$\frac{16}{17}$	$\frac{0}{17}$	$\frac{1}{17}$	$\frac{0}{17}$	$\frac{0}{17}$
Bing(new)	$\frac{9}{9}$	$\frac{0}{9}$	$\frac{0}{9}$	$\frac{0}{9}$	$\frac{0}{9}$
Average	0.96	0.017	0.0116	0	0.0116

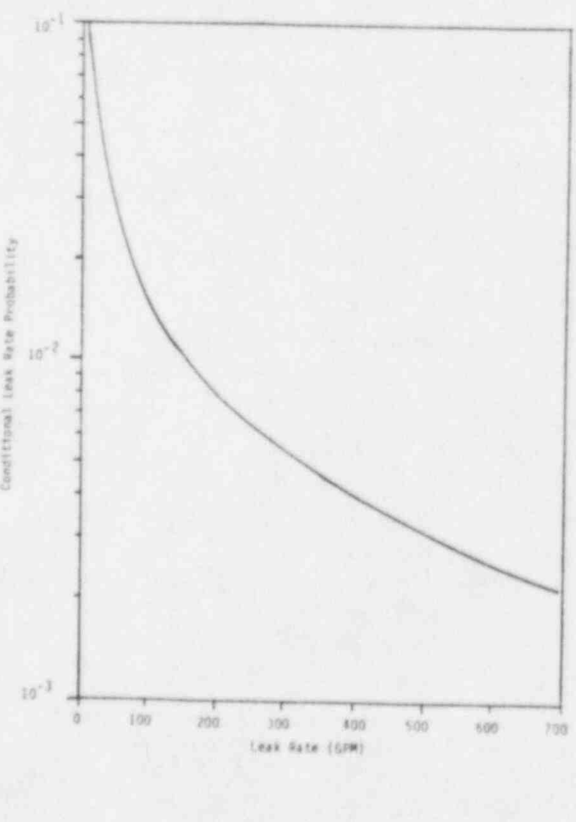

Figure 6. The conditional exeedance probability of leak rate given a RCP seal failure

3.3 LOCA Frequency Estimation

To estimate the annual frequency for various LOCA sizes, two steps are required: a) to estimate the annual frequency distributions for single, double, triple, and quadruple RCP seal failures, and b) to determine the exceedance frequency distribution for the spectrum of RCP leak rates culled from the data.

The following discusses the statistical methods used to obtain the above information.

a) Annual Frequency of RCP Seal Failures

The purpose here is to integrate the separate pieces of information estimated earlier in this report for determining the frequency of RCP seal failures at a plant. The best way to illustrate the methods is through an example.

Consider a plant with four main reactor coolant pumps (having either two or four loops) and assume that upon a RCP seal failure detection and repair are instantaneous. With this assumption in conjunction with the consideration of constant hazard rate, a renewal process (Poisson process) can be used to describe the number of seal failures per pump within a time period, T. To proceed, several definitions and notations are needed.

(a) Letters A, B, C, and D stand for the pump identification code.

(b) Subscripts S, D, T, and Q indicate single, double, triple and quadruple events, respectively.

(c) $\lambda_S(A)$, $\lambda_D(A)$, $\lambda_T(A)$, and $\lambda_Q(A)$ stand for the hazard rate of single, double, triple, and quadruple events involving Pump A.

(d) $m_S(A,\tau)$, $m_D(A,\tau)$, $m_T(A,\tau)$, and $m_Q(A,\tau)$ define the numbers of single, double, triple, and quadruple failures involving Pump A within the time period, τ.

(e) $P(m_S(A,\tau))$ is the probability of observing m_S failures of single-type events involving Pump A within the time period τ. Similarly, by changing the pump identification code and event-type subscripts, other probabilities can be defined.

The hazard rate for single failures in a plant with four units of reactor coolant pumps is then expressed by

$$\lambda_S = \lambda_S(A) + \lambda_S(B) + \lambda_S(C) + \lambda_S(D) = 4\beta_S\lambda \quad . \tag{14}$$

To evaluate the hazard rate for double RCP failures in the same plant, some preliminary explanations are required. The hazard rate of a double failure involving Pump A, namely, $\lambda_D(A)$, represents a failure of either AB, AC, or AD. Similarly, $\lambda_D(B)$ represents a failure of either BA, BC, or BD. Since no distinction is placed on double failures such as AB and BA, the hazard rate for double failures is

$$\lambda_D = [\lambda_D(A) + \lambda_D(B) + \lambda_D(C) + \lambda_D(D)]/2 = 2\beta_D\lambda \quad . \tag{15}$$

In similar fashion, the values for λ_T and λ_Q can be expressed as

$$\lambda_T = [\lambda_T(A) + \lambda_T(B) + \lambda_T(C) + \lambda_T(D)]/3 = 4/3\beta_T\lambda \tag{16}$$

and

$$\lambda_Q = [\lambda_Q(A) + \lambda_Q(B) + \lambda_Q(C) + \lambda_Q(D)]/4 = \beta_Q\lambda \quad . \tag{17}$$

Given the above equations, it can be shown that for short time intervals ($\tau \ll 1/\lambda$) the total number of failures is

$$m = m_S + 2m_D + 3m_T + 4m_Q = 4(\beta_S + \beta_D + \beta_T + \beta_Q)\lambda = 4\lambda \quad . \tag{18}$$

The distribution for the total number of single, double, triple, or quadruple failures within a period τ then can be expressed as;

$$P(m_S(\tau)) = (4\beta_S\lambda\tau)^{m_S} e^{-4\beta_S\lambda\tau}/m_S! \tag{19}$$

$$P(m_D(\tau)) = (2\beta_D\lambda\tau)^{m_D} e^{-2\beta_D\lambda\tau}/m_D! \tag{20}$$

$$P(m_T(\tau)) = (4/3\beta_T\lambda\tau)^{m_T} e^{-4/3\beta_T\lambda\tau}/m_T! \tag{21}$$

$$P(m_Q(\tau)) = (4\beta_Q\lambda\tau)^{m_Q} e^{-\beta_Q\lambda\tau}/m_Q! \tag{22}$$

However, for large time periods ($\tau \sim O(1/\lambda)$), the process for m failures would no longer be a Poisson process.

b) Exceedance Frequency Distribution for the Spectrum of RCP Leak Rates

Given the conditional probability distribution of leak rate and failure of RCP seals as discussed in Section 3.2, the expected unconditional frequency distribution for the spectrum of RCP leak rates on a plant basis for a period τ can be expressed as

$$F(LR,4RCP) = \sum_{m_S=0}^{\infty} m_S P(m_S,\tau)F(LR) + \sum_{m_D=0}^{\infty} m_D P(m_D,\tau)F(2LR) \quad , \qquad (23)$$

$$+ \sum_{m_T=0}^{\infty} m_T P(m_T,\tau)F(3LR) + \sum_{m_Q=0}^{\infty} m_Q P(m_Q,\tau)F(4LR) \quad . \qquad (24)$$

To evaluate this expression, the λ's values are taken from the best estimates* given in Table 5, and an error factor of 9 is conservatively assumed for the ratio of the upper to the lower bound of the λ value due to plant-to-plant variability in each group. The β values are taken from the maximum likelihood estimates given in Table 3. The variations for β values have not been considered, owing to the large uncertainty already incorporated in estimating λ. For the B&W plants with Bingham pumps, the data for determining the β values is sparse; hence, the β values for B&W plants with BJ pumps were incorporated.

Figures 7 through 10 present the annual exceedance frequency of RCP-induced leak rates and the associated bounds for different plant and pump vendors. The results presented for CE plants with BJ pumps (Figure 9) are considered conservative because in the analysis no credit has been given to the fourth-stage vapor seal. This aspect is discussed further in Section 4.

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*Best estimate value of λ's is considered the mean, not the median. The definition of error factor does not imply that the distribution of λ is lognormal, i.e., the data in Table 5 do not indicate lognormality. The error factors are used only to indicate the judgmental bounds on the exceedance frequency of RCP leak rates.

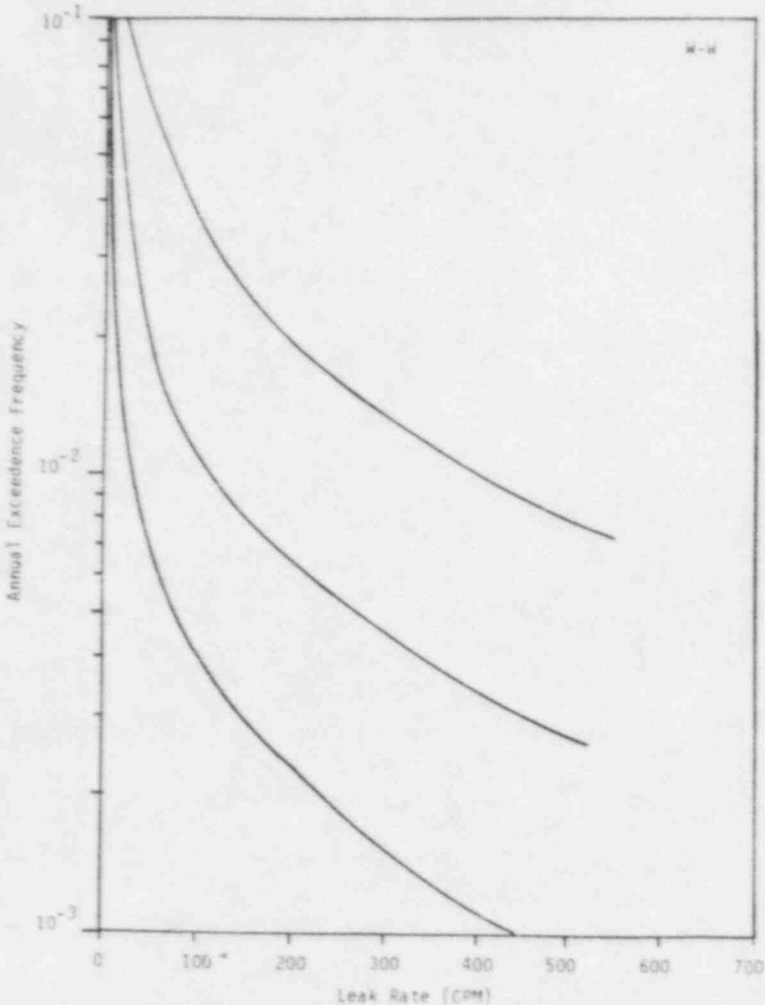

Figure 7. Annual exceedence frequency vs leak rate for W-W with 4 RCPs (lower bound, best estimate, upper bound).

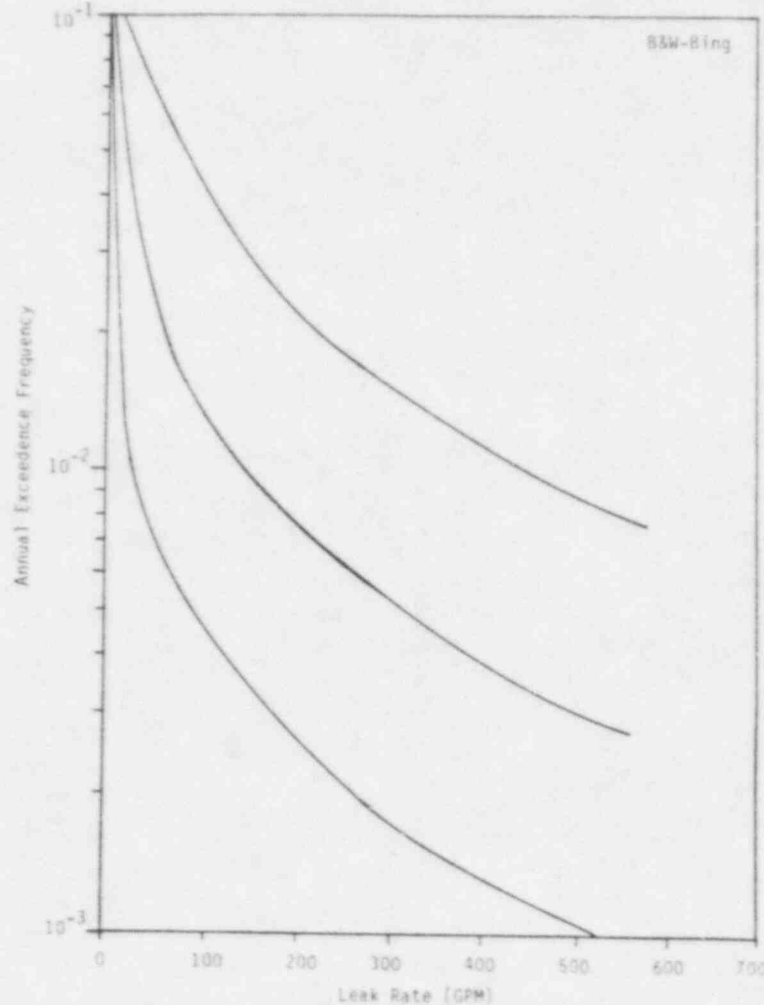

Figure 8. Annual exceedence frequency vs leak rate for B&W-Bingham with 4 RCPs (lower bound, best estimate, upper bound).

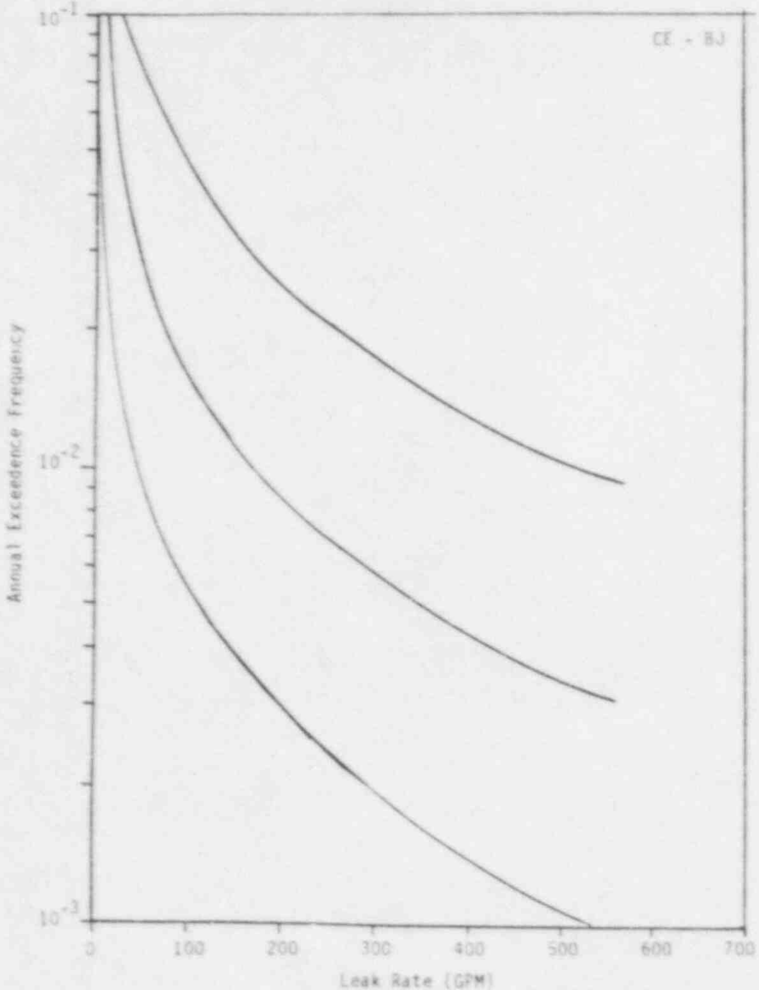

Figure 9. Annual exceedence frequency vs leak rate for CE-BJ with 4 RCPs (lower bound, best estimate, upper bound).

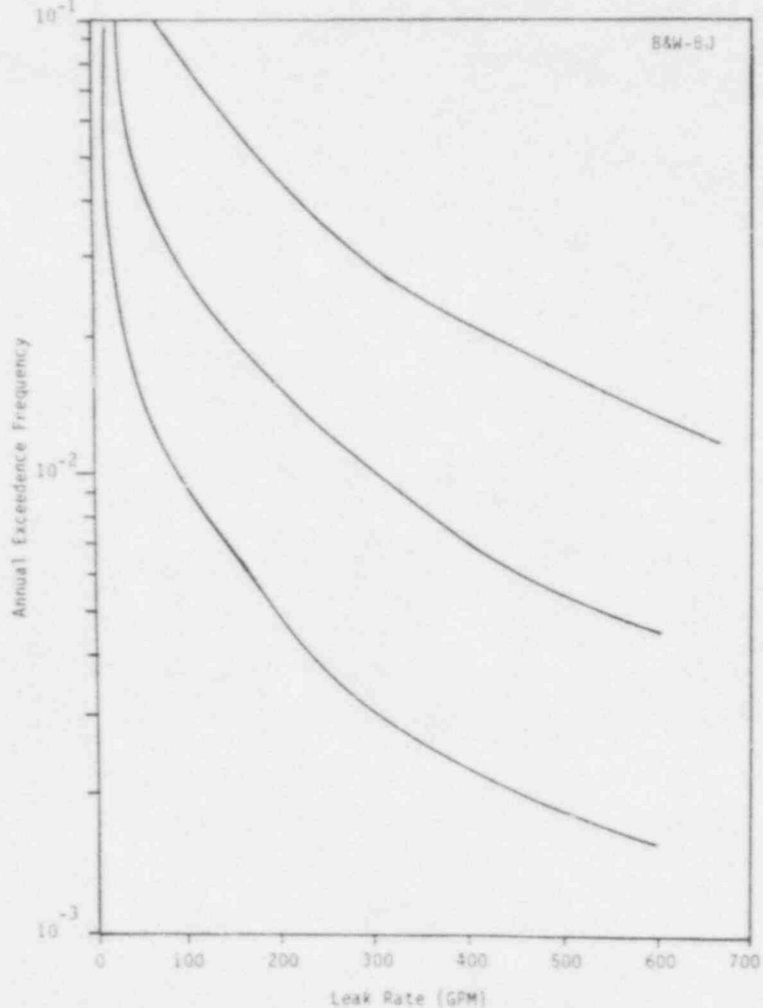

Figure 10. Annual exceedence frequency vs leak rate for B&W-BJ with 4 RCPs (lower bound, best estimate, upper bound).

4.0 CONCLUSION (PHASE - I)

In this study the primary objectives are to determine the annual frequencies of various primary coolant leak rates caused by mechanical and maintenance induced RCP seal failures and to identify improvements which would minimize the frequency of seal failures and the resulting leak rates. The first phase of the study was a data survey of the pump seal failures in nuclear power plants culled from the several available data sources for subsequent statistical analysis. No attempt was made to substantiate the statistical inferences by additional thermohydraulic analyses or from pump tests. The study dealt only with mechanical- or maintenance-induced seal failure during plant operation, and did not address seal failure resulting from extended station blackout or simultaneous loss of cooling and seal injection flow.

In the context of the objectives and the scope of this study, the following conclusions can be drawn:

a) Seal Life Distribution:

The graphical results showing the reliability of RCP seals manufactured by various pump vendors are provided in Figures 4 and 5 for PWRs and BWRs, respectively. These reliability curves are based on the assumption that the seals are as good as new after each refueling outage. Regardless of the pump design, the reliability graphs indicate a slow degradation of pump seals after a service life of about two years for PWRs and 16 months for BWRs. The failure intensity (hazard rate) for the RCP seal failure, with the exception of pre-1974 Bingham pumps (two-stage seal, old design), seems to be constant within a service life less than 20 months for PWRs and 16 months for BWRs. The B&W plants with a BJ pump design have shown failure rates comparatively higher than other combinations of plant vendors and pump designers. No specific explanation for these higher failure rates could be asserted.

b) Seal Leakage Distribution:

A total of seven large leakages (25 gpm or above) induced by RCP seal failure have been reported in PWR plants. The plant name, vendor, RCP designer, event date, and associated leak rate are given in Table 8. Large leakages are observed in only Westinghouse and Babcock & Wilcox plants. Combustion Engineering plants with BJ pumps have experienced no large leakages. The reason may be that the BJ pumps in CE plants have an additional fourth-stage seal (vapor seal) as opposed to BJ pumps in B&W plants. However, an event such as the one that occurred in ANO-1 (5/10/80) could have seriously challenged long-term operability of vapor seal stage if it had happened in a CE plant.

The leakage data for RCP seal failures for various pump vendors are combined and fitted to a Weibull distribution (Figure 6 in the text). The reasons for the aggregation of leak rate data are a)

Table 8. Events with Excessive Leakages Induced by RCP Seal Failures in PWRs

Plant Name	Vendor and RCP Designer	Leak Rates (gpm)	Total Leakages (Gallons)	Event Date
Arkansas Nuclear 1	B&W, BJ	25	NA	08/27/76
Arkansas Nuclear 1	B&W, BJ	300	60,000	05/10/80
Arkansas Nuclear 1	B&W, BJ	28	NA	08/08/82
H. B. Robinson	W, W	500	200,000	05/01/75
Indian Point 2	W, W	75	90,000	07/02/77
Oconee 2	B&W, Bing	90	50,000	01/22/74
Salem 1	W, W	35*	15,000	10/21/78
Connecticut Yankee	W, W	≤25*	4,020	08/21/77

*These values of leak rates are approximately estimated from the event description.

sparsity of data, and b) lack of significant statistical variations among pump designers (with the exception of CE-BJ pumps). The leak rate distribution obtained in such a manner is assumed to be applicable to various pump designers and plant vendors. It is understood that this approach will be very conservative for BJ pumps in CE plants. However, without further investigation on the performance of vapor seal conditional to failure of the other seal stages, no specific safety margin can be determined for 4-stage seals BJ pumps.*

For the BWR plants, two major leakages occurred at Brunswick Unit 2 during August and September 1975. These events are discussed in Section 3.0. It is judged that proper corrective actions were taken to minimize the possibility of recurrence. Therefore, further investigation of recirculation pump seal failures and associated leak rates was not performed.

*The CE plants have not experienced vapor seal failure.

c) Recommendations for Reducing Seal-Failure-Induced Risk:

The risk induced by RCP seal failures in nuclear power plants can be lessened by either reducing the frequency of seal failures or minimizing the leak rates by preventing the potential of cascade failures of the seals.

To minimize the frequency of seal failure, it is important to identify the various root causes and to determine how seal failures may be avoided. From this study, it appears that there are vast differences in the root causes of seal failures in various plants. This statement should be qualified by pointing out the deficiencies in event reporting systems and the difficulties in determining the actual causes of the seal failures. The following are three major causes of seal failures and tentative recommendations for reducing their frequency.

1. End-of-life failures due to slow intermittent failure of the seals contribute to about 40% of seal failures. It is recommended that the seals be inspected at least at each refueling outage, unless the actual plant data support the view that the average seal lifetime exceeds the refueling interval. In addition, changing seal surface with a hard material is a viable solution.

2. Maintenance-induced seal failures contribute to about 20% of the overall seal failures in NPPs. Proper training and procedure for maintenance would reduce the frequency of seal failures.

3. Seal failures due to plant transients and deviation from operational limits of the seals contribute to about 20% of the seal failures. Operator training and proper operational guidelines would be helpful in reducing the possibility of seal failures due to such mechanisms. In addition, inspection of pump seals after experiencing severe operational transients such as; loss of both component cooling and seal injection with plant at normal operating temperature and pressure, inadvertent closure of seal return valve (Byron Jackson and Bingham pumps), rapid temperature changes in seal injection flow, etc., is recommended.

   One important recommendation with implications for all three failure mechanisms discussed above is the periodic review and update of maintenance and quality control practices to reflect recent experience and to potentially identify a pump with below average performance.

   The following are some recommendations for minimizing the possibility of cascade seal failures and excessive leakage, given an RCP seal failure.

i) Assure the fast isolation of seal return line and increase of seal injection flow for Westinghouse pumps, given excessive leakage of any seal.* Automation of seal injection flow control and seal return line isolation is desirable. Similar actions in regard to pump trip and isolation of controlled leak off flow (staging flow) for Bingham and Byron Jackson pumps are recommended.

ii) Assure sufficient heat removal capacity to allow thermal barrier heat exchanger to cool the escaped primary coolant, given a failure of a seal stage. The maximum leak rate of the primary coolant is a function of seal injection flow rate and isolation of return line (Item 1).

iii) Assure uninterrupted cooling flow (CCW) to the thermal barrier heat exchanger. Provide proper instrumentation to allow the operator to reestablish the CCW flow if it is lost because of inadvertent isolation.

iv) Investigate the possibility of implementing an additional seal stage (vapor seal) to the conventional three-stage seal RCPs. Provide supporting information for performance of vapor seal under full reactor pressure by means of testing.

In addition to the above, continued research is recommended on designing a seal with better performance and longer service life.

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*During extended station blackout or loss of both seal injection and cooling, the isolation of the return line may not be appropriate for Westinghouse pumps.

5.0 PHASE II - METHODOLOGY

The impact of mechanical- and maintenance-induced RCP seal failures on plant safety is evaluated by determining the means of maintaining the core coolant inventory, given varying sized leak rates. Two categories of leak rates are considered to make the study tractable:

(a) Leak rates below the normal makeup capacity of the plant: The mechanical- and maintenance-induced RCP seal failures leading to a leakage within this category may initiate a slow core-melt scenario if the normal makeup is lost. The operation of the normal makeup system, its maximum capacity, and the manual/automatic actions needed to provide the maximum flow were investigated to determine the efficiency of this system. The operability of the normal makeup system was analyzed qualitatively. The formal fault tree methodologies and reliability evaluations have not been used in estimating the availability of this system. Therefore, the qualitative judgments and recommendations were based on insights gained in the review of the system design and the operational procedure. In addition, the plant responses (mainly primary pressure and temperature) was reviewed during the RCP leakage within this category and failure of the normal makeup system. This review was made to determine if the primary pressure would drop to the initiation setpoint of the Safety Injection Signal (SIS) which automatically actuates the Emergency Core Cooling System (ECCS). If the setpoint of SIS could not be reached, the required operator actions for maintaining the core coolant inventory were identified.

(b) Leak rates in excess of normal makeup capacity: the mechanical- and maintenance-induced RCP seal failures leading to a leakage within this category were treated like a small Loss-of-Coolant Accident (LOCA), as conventionally considered in PRAs. The plant responses (mainly primary coolant pressure and temperature) to LOCAs of these sizes were reviewed to determine the time required for SIS actuation. The potential of system interaction between the failure of the normal makeup system and the High Pressure Injection System (HPIS) was investigated. The formal PRA methodologies were used in evaluating core-melt frequency. The dominant minimal cutsets at the core-melt accident sequence level were identified and their associated probabilities were evaluated. The uncertainties were propagated for the dominant accident sequences. The point estimates based on the mean values of the probability of primary faults were calculated for the nondominant accident sequences.

The above categorization and approach has been applied to three nuclear power plants because of the generic nature of the problem. The selection of the three nuclear power plants was based on the availability of PRA reports, detailed plant documentations, and the coverage of the three PWR vendors, namely, Babcock and Wilcox (B&W), Combustion Engineering (CE) ,and Westinghouse (W). Arkansas Nuclear One Unit 1 (ANO-1), Calvert Cliffs Unit 1 (CC-1), and Indian Point Unit 3 (IP-3) are considered as representative of B&W, CE, and W, respectively. The products of Interim Reliability Evaluation Program (IREP)[11-12] are used as the base PRA for ANO-1 and Calvert Cliffs. The Indian Point Probabilistic Risk Assessment[13] and the SNL review of this PRA[14] are used as the base PRA for Indian Point Unit 3.

At the outset, it should be emphasized that the data, the PRA logic modelings, and other pertinent information required for the quantitative evaluations are extracted primarily from the published PRA documents. The authors made no effort to verify or modify the existing information. Therefore, this study shall not be considered as either confirmation or reevaluation of the reference documents. The specific objective addressed here is to evaluate the portion of the core-melt frequency due to the RCP mechanical- and maintenance-induced seal failure using the available PRAs. As such, the approach taken in meeting the objective of this phase and the conclusions drawn therefrom must be tempered by these factors.

5.1 RCP Leakage Below the Normal Makeup Capacity

The operator deals with the RCP leakages within this category, primarily by reducing the plant power level (initiation of a plant shutdown) and increasing the makeup flow. For each of the three representative nuclear power plants the following discusses the operation of a normal makeup system, either automatic or manual; the required operator actions conditional to the failure of this system; and the associated instrumentations.

5.1.1 Normal Makeup System for Arkansas Nuclear One Unit 1 (ANO-1)

During normal plant operation, the high pressure injection/recirculation system is known as the makeup and purification system. A simplified diagram of the system, reproduced from Reference 10, is given in Figure 11. At normal reactor pressure, one makeup pump (Pump B) can provide about 150 gpm flow. This flow is injected into the RCS through the RCP seal injection lines and the makeup line (Valve CV-1235). However, at lower RCS pressures, a makeup delivers much higher flow. The pump characteristic curve[15] is given in Figure 12.

If a very small loss-of-coolant accident occurs, say, due to RCP-seal failures, and if the makeup tank level is low, the makeup tank can be supplied from the boric acid/domestic water system at the following rates:

(a) minimum 2 gpm,

(b) normal 60 to 70 gpm, and

(c) maximum ~150 gpm.

For a small-break LOCA, ANO-1 procedures require the full action of the emergency procedure for a SBLOCA at > 30 gpm. This would include a reactor shutdown. As the makeup tank level decreases, the operator obtains makeup for RCS either from Borated Water Storage Tank (BWST) or by resupplying the makeup tank with borated water from the domestic water and boric acid system (called "Batching In"). It is preferable to "Batch-In" if the operator can keep up

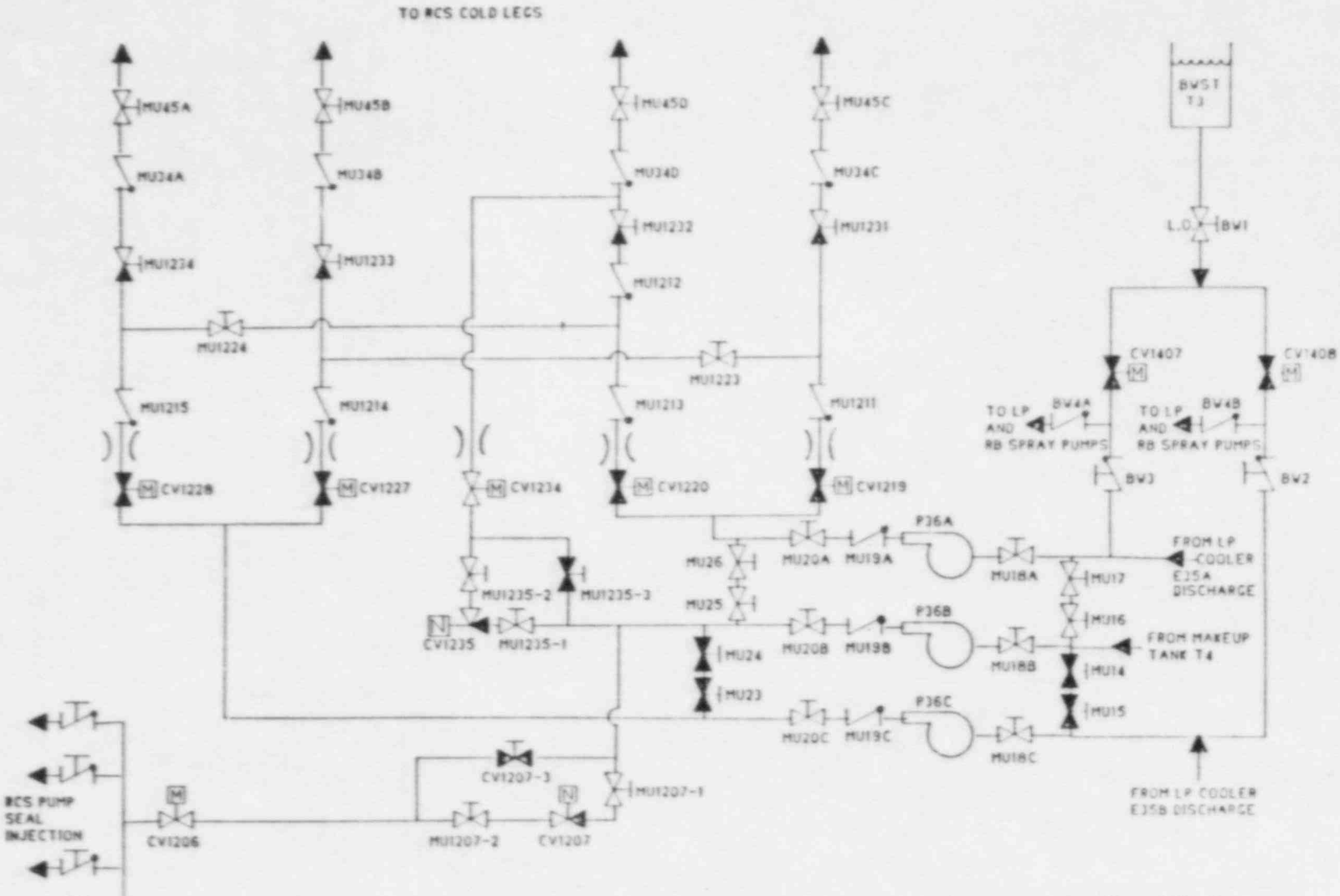

Figure 11. Simplified piping diagram of normal makeup system for ANO-1.

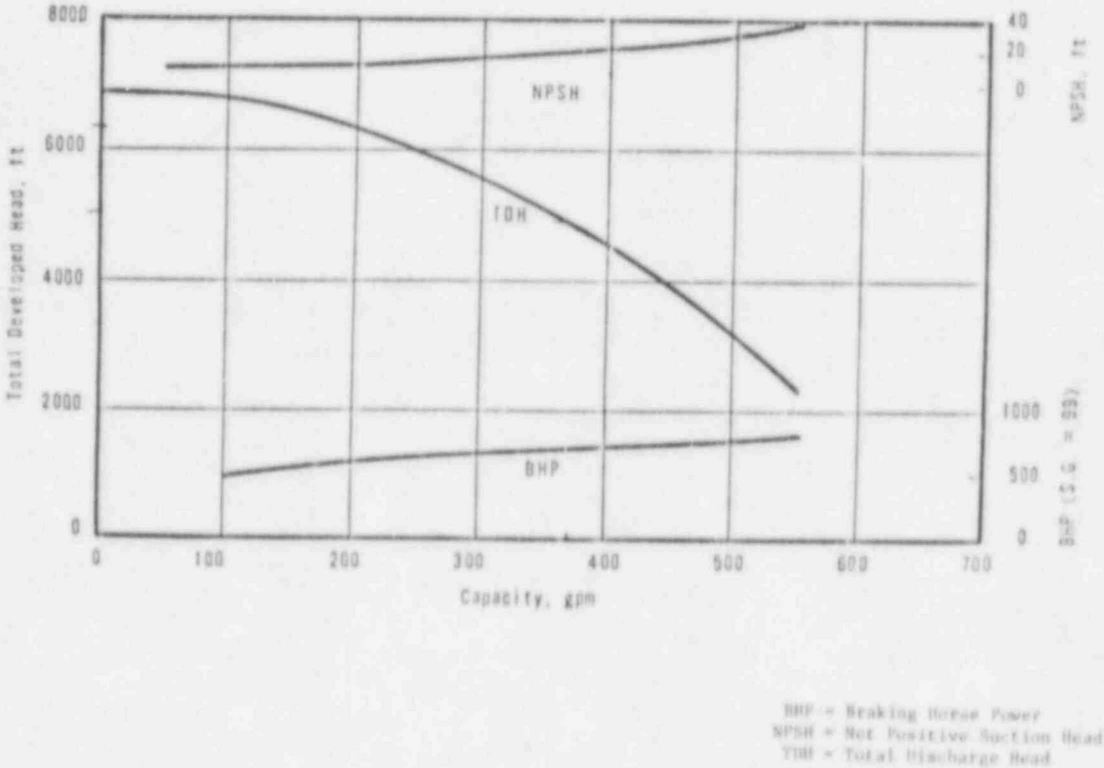

Figure 12. The characteristic curve for a HPI pump in ANO-1.

with the leak from RCS. The "Batch-In" process is manual at ANO-1. This plant has no blender for its "Batch-In." The operator must first set the valve for domestic water, then reset, then set the valve for boric acid, then reset again to complete a batch cycle. If the operator can not keep up with the RCS leak by "Batching In," he will then open the supply valve to BWST.

Upon initiation of RCP leakage, the pressurizer level will drop, opening the discharge flow control valve in the makeup system, which in turn increases the flow rate through the system. The higher flow from the makeup tank (above the normal 45 gpm letdown) will reduce the Makeup Tank (MUT) water level. A low level in the MUT annunciates an alarm which in conjunction with the level indicator informs the operator of the existing situation. The operator then takes action to initiate flow to MUT by blending the boric acid flow and condensate water (domestic water system). The maximum makeup flow from this system to the makeup tank is 150 gpm. Hence, for leak rates below 150 gpm, the pressurizer level, the pressurizer pressure, and the MUT water level will be maintained. The operator has ample time to shut down the plant and proceed with the normal cooldown.

Failure of the operator either to perform the batch in mode or to obtain the makeup for RCs from BWST will result in depletion of the makeup tank and cause damage to the operating makeup pump due to cavitation. The B&W plant response to a small primary leakage, conditional to the failure of normal makeup systems, has been analyzed generically in several NRC and industry reports. The analysis performed by the NRC[16] for 177-FA B&W lowered loop plants concluded that "for breaks smaller than 0.005 ft^2, including those breaks within the capacity of the makeup pumps, should the makeup fail, system pressure will decrease to the reactor trip and HPI actuation setpoints prior to the formation of the steam bubble in the hot leg so that the system remains solid prior to HPI actuation.[11] In the analysis for the 0.005-ft^2 break, no flashing occurred in the system until the primary system pressure decreased to 1400 psia. For smaller breaks, the system will depressurize slower with flashing occurring at a lower temperature due to a reduced decay heat generation rate with time after reactor trip. Since the nominal HPI actuation setpoints for the 177-FA lowered and raised loop plants are all greater than 1500 psig (for ANO-1 it is 1500 psig), and the realistic instrumentation errors are only 50 psi, the ESAS system will actuate HPI prior to the formation of a bubble in the hot leg that could interrupt natural circulation.

The above analyses are based on successful operation of Emergency Feedwater Systems (EFS) in lowered loop plants. Similar analysis by B&W,[16] assuming loss of EFS, indicates the possibility of reactor pressure staying above the 1500 psig, given break sizes less than 0.01 ft^2. In this situation manual actuation of HPI system by the operator is required.

We feel that the probability of concurrent failures of the RCP seals and the described operator errors is negligible, but it is strongly recommended that the process of transferring the makeup pump suction to the BWST when the makeup tank level is low be automated. Finally, it is our judgment that the frequency of RCP seal-failure-induced leak rates below the normal makeup capacity in conjunction with the aforenoted operator error (expected probability of operator error for this task is estimated[11] to be 1.0×10^{-4}) is comparatively much smaller than the frequency of RCP leak rates above the normal makeup capacity. Hence, the core melt frequency induced by the RCP leak rates below normal makeup capacity is not quantified here.

5.1.2 Normal Makeup System for Calvert Cliffs Unit 1 (CC-1)

During normal plant operation, the charging pumps and the volume control tank (VCT), which are the components of the chemical and volume control system (CVCS), perform the function of primary coolant makeup. The CVCS is designed to perform a variety of functions. The functions of interest in this discussion, as described in the Calvert Cliffs FSAR,[17] are as follows:

- control the reactor coolant volume by compensating for coolant contraction or expansion from changes in reactor coolant temperature and other coolant losses or additions;
- control the boron concentration in the RCS; and
- inject concentrated boric acid into the RCS upon a safety injection actuation signal.

During normal operation one charging pump is in operation with its suction aligned to the VCT via the normally open VCT outlet valve, 1-MOV-501 (see

Figure 13). The flow discharged from the charging pump passes through the shell side of the regenerative heat exchanger and returns to the RCS via the charging lines. There are three charging pumps, all of the positive displacement type. Each pump has a discharge capacity of 44 gpm and a design pressure of 2735 psig.

The CVCS automatically adjusts the volume of water in the RCS by comparing the programmed pressurizer level setpoint with the measured pressurizer water level. The programmed pressurizer level setpoint varies with reactor power. The resulting level error signal obtained from this comparison controls the operation of the charging pumps and the letdown control valve, 1-CV-110P. Under normal equilibrium conditions, the controlled bleed-off from all four reactor coolant pumps (a total of 4 gpm) plus the letdown flow (40 gpm) equals the charging flow from an operating pump (44 gpm).

The makeup control system maintains the water level in the VCT. If the level in the VCT reaches a high-level setpoint, the letdown flow is diverted by the three-way VCT inlet valve, 1-CV-500, to the liquid waste processing system. The makeup control system is normally set to the "automatic mode" of operation. In this mode, if the level in the VCT reaches a low-level setpoint, then a predetermined solution of concentrated boric acid and demineralized water is introduced into the VCT.

In the event of an RCP seal failure, the loss of reactor coolant leads to a decrease in the pressurizer level, and, in response, the pressurizer level control program will start one or both standby charging pumps, depending on the amount of reactor coolant lost through the failed RCP seal (i.e., the extent of seal failure). In addition, the letdown control valve, 1-CV-110P, is regulated to minimize the letdown flow in an attempt to maintain pressurizer level. The water level in the VCT will start to decrease because additional charging pumps are in operation and there is a reduction in the letdown flow. When the level in the VCT reaches a low level setpoint, an alarm is annunciated, and makeup water borated to the existing concentration of the reactor coolant is automatically supplied to the VCT. This occurs because the makeup control system is normally set to the automatic mode of operation. Automatic makeup to the VCT is achieved by operation of the RC makeup pumps and the boric acid pumps. Demineralized water is pumped by one of two RC makeup pumps and flows through the control valve, 1-CV-210X. Downstream of the control valve, 1-CV-210X, the demineralized water mixes with the concentrated boric acid and flows into the VCT via the control valve, 1-CV-512. The concentrated boric acid is delivered to the mixing header from the concentrated boric acid storage tanks by the operation of one of two boric acid pumps via the control valve, 1-CV-210Y.

The level in the VCT may continue to decrease either because the loss of reactor coolant via the failed RCP seal exceeds the automatic makeup capacity or because of failures in the automatic makeup system. Should the level in the VCT reach a low-low setpoint, a low-low-level alarm is annunciated, and the isolation valve at the refueling water tank (RWT) suction line, 1-MOV-504, opens automatically to align the charging pump suction to the RWT. Also, the VCT discharge isolation valve, 1-MOV-501, is closed. Thus, the suction of the charging pumps is switched from the VCT to the RWST.

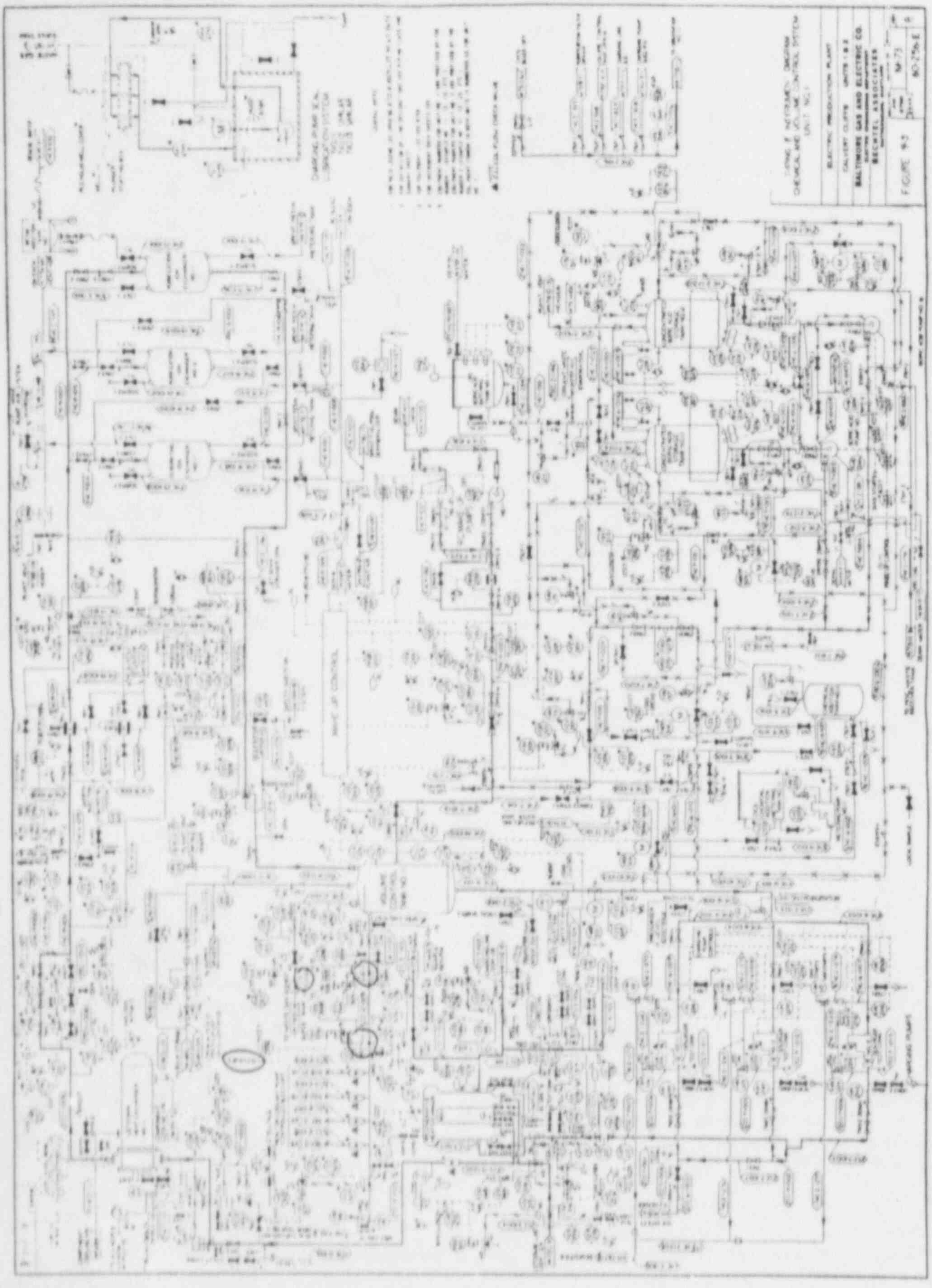

Figure 13. Piping and instrumentation diagram for chemical and volume control system for Calvert Cliffs Unit 1.

The likelihood of concurrent failures of the RCP-seals and the normal makeup system, given all the automatic transfers described above, is judged to be negligible. To assess the small-break response characteristic of CE plants for LOCAs just large enough to exceed the normal makeup capacity (0.3 in. diameter, ~100 gpm), Reference 18 was reviewed. In this case, depressurization to the low pressure reactor trip level (1720 psia) requires just over 1 hour. Upon reactor trip, the reactor coolant pumps are also assumed to be tripped, and system cooldown results in a rapid depressurization to 1200 psia and HPSI flow is actuated when the pressure drops below the actuation setpoint of 1578 psig. Following a 30-sec delay until the HPSI flow begins (because of lower HPSI pump shutoff head of 1289 psia), pressure recovery to near the HPSI shutoff head is accomplished, and is sustained for the remainder of the transient.

5.1.3 Normal Makeup System for Indian Point Unit 3 (IP-3)

During normal plant operation, the charging pumps and the volume control tank (VCT) which are the components of the Chemical and Volume Control Systems (CVCS) perform the function of primary coolant makeup. The total volume of the VCT consists of a liquid space of 130 ft^3 and a vapor space of 270 ft^3. The vapor space is occupied predominantly by hydrogen. Consequently, if the water in the VCT is depleted and the charging pumps are not aligned to the alternate suction sources, the pumps will aspirate the gases in the VCT. Several automatic actions designed in the system to prevent such an event are discussed later in this section.

The CVCS is designed to perform a variety of functions. Of particular interest in this discussion, are those described in the Indian Point FSAR[19] viz.

- maintain the proper water inventory in the reactor coolant system,
- adjust the concentration of boric acid in the reactor coolant for chemical reactivity control, and
- provide required seal water flow for the reactor coolant pump shaft.

During normal plant operation, one charging pump is running with its suction aligned to the VCT via the normally open level control valve, LCV-112C. The charging pump raises the water pressure above the reactor coolant system (RCS) pressure. The high pressure water discharged from the charging pump flows in two parallel paths (see Figure 14). One path flows directly to the reactor coolant system through the tube side of the regenerative heat exchanger and the charging line to the cold leg of RCS Loop 1. The other path injects water into the four reactor coolant pumps between the thermal barrier and the shaft seal. Part of this injection flow enters the RCS through the RCP labyrinth seals. The remainder returns to the VCT through a common header via the seal water filter and the seal water heat exchanger.

There are three charging pumps. Normally, one pump is in operation and the other two are on standby. These pumps are of the variable speed positive displacement type. Each pump is hydraulically coupled to its motor through a

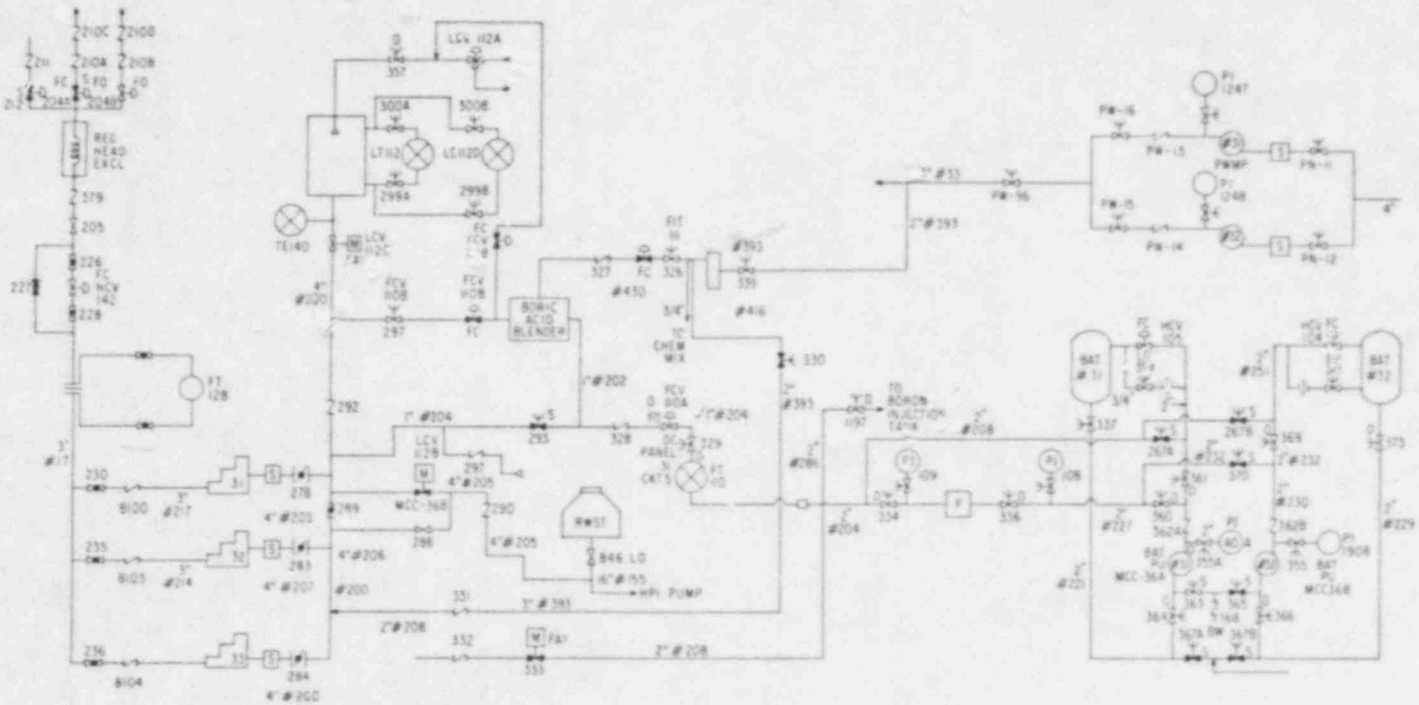

Figure 14. The piping diagram for CVCS in Indian Point 3.

fluid drive. The discharge flow rate from the pump is controlled by changing the speed of the pump by varying the degree of fluid coupling between the pump and its motor. In the automatic mode operating is controlled by pressurizer level by comparing the programmed pressurizer level to the actual pressurizer level. The resulting error signal is used to control the speed and, therefore, the flow rate of the charging pump. The programmed pressurizer level depends on the average temperature of the reactor coolant (i.e., reactor power).

Under normal equilibrium operating conditions, the letdown flow (75 gpm) plus the seal water return flow (12 gpm) equals the charging line flow (55 gpm) and the seal injection flow (32 gpm). Accordingly, the normal discharge flow rate from one charging pump is 87 gpm. However, the capacity for each charging pump is 98 gpm.

The reactor makeup control subsystem of the CVCS maintains the desired operating fluid inventory in the VCT. If the level in the VCT reaches a high-level setpoint, the letdown flow is diverted by the three-way-level controlled VCT inlet valve, LCV-112A, to the holdup tanks. The reactor makeup control is normally set to the "Automatic Makeup" position. In this position, if the level in the VCT reaches a low-level setpoint, then a preset solution of concentrated boric acid and primary water is mixed in the boric acid, and primary water is mixed in the boric acid blender and introduced to the charging pump suction via the flow control valve, FCV-110B. The automatic makeup mode of operation compensates for minor leakage without significantly changing the RCS boron concentration. The control of the operating charging pump based on the programmed pressurizer level and the automatic control of the VCT level is similar to that of Calvert Cliffs 1.

In the event of an RCP seal failure, the loss of reactor coolant would lead to a decrease in the pressurizer level and an increase in the pressurizer level error signal (i.e., the difference between the programmed and the actual pressurizer levels). In response to this error signal, the speed of the operating charging pump will increase in order to maintain the pressurizer level. When this pump reaches full speed, the operator will have to manually start and control a second pump. The second pump controller should be set such that the first pump, which is in automatic control, has sufficient leeway to compensate for slight changes in the demand signal. If the second pump is running at maximum speed and there is no decrease in the speed of the first running pump, then the third pump has to be manually started and controlled.

Meanwhile, as a result of increased flow rate from one or more charging pumps the level in VCT will start to decrease. When the level in the VCT reach a low-level setpoint (21.4%), the automatic makeup to the charging pump suction will be initiated. This is achieved by opening the makeup stop valve, FCV-110B, to the charging pump suction, the concentrated boric acid control valve, FCV-110A, and the primary water makeup control valve, FCV-11A. The flow rate through the two control valves in the automatic mode of operation is preset. The low-level signal from the VCT will start the boric acid transfer pump and the primary water makeup pump. In normal operation, one of two primary water makeup pumps and one of two boric acid transfer pumps are aligned for operation on demand from the makeup control system. The discharge capacities of a primary water makeup pump and a boric acid transfer pump are 150 gpm and 75 gpm, respectively. The primary water makeup pump draws its suction from the 165,000 gallon primary water storage tank. Each boric acid pump is normally aligned to a 7000-gal boric acid tank. The flow rate through the primary water makeup control valve, FCV-111A, is set at 120 gpm, which is approximately the achievable flow rate of the normal makeup system. The flow rate through the concentrated boric acid control valve, FCV-110A, depends on the existing boron concentration in the RCS. This is preset during normal operation.

Automatic initiation of makeup is not alarmed in the Control Room. However, if the level in the VCT continues to decrease and reaches a low-level setpoint of 11.4%,[17] then the alarm "Volume Control Tank Low Level" is annunciated.

If the extent of RCP seal failure is severe, then the automatic makeup to the VCT will not be able to restore VCT level. Consequently, the level in the VCT will continue to decrease, and when the level reaches a setpoint of 8.5%[2] the level control valve, LCV-112B, will open automatically to admit water from the refueling water storage tank (RWST). When the level control valve, LCV-112B, reaches its full open position, then the VCT outlet valve, LCV-112C, will be closed. Thus, the suctions of the charging pumps are aligned to the RWST.

The above discussion on the operation of the charging pumps and its suction sources assumes that during the RCP seal LOCA no safety injection actuation signal (SIAS) was initiated. If, however, a safety injection actuation signal is initiated during the course of the accident, either because of low pressurizer pressure or high containment pressure, then the charging pumps will be tripped. However, the SIAS will automatically start all three high pressure safety injection pumps.

It may be recalled that in Calvert Cliffs 1, the SIAS starts all three charging pumps and high pressure injection pumps.

The possibility of concurrent failures of RCP seals with the leakages within the normal makeup capacity (120 gpm) and of the normal makeup system, given all the automatic transfers described above, is judged to be negligible. In addition, the plant response to an event consisting of a primary leakage within the normal makeup capacity and failure of the normal makeup system has been considered in Reference 20 where it is stated that the plant response to this event would be the same as to a small-small LOCA (mode 2 analysis of this reference regarding to equivalent break diameters between 0.375 and 1 in.). Therefore, it is our judgment that inclusion of this scenario of events would not change PRA results significantly and is well within the uncertainty bounds.

5.2 RCP Leakages in Excess of the Normal Makeup Capacity

The RCP leakages within this category are usually considered as a small-small LOCA in conventional PRAs. The associated scenarios leading core melt for the small-small LOCA-initiating events are well discussed in the available

PRA documents. Therefore, this portion of the study is limited primarily to reviewing the existing PRAs for the three representative plants, identifying the dominant small-small LOCA accident sequences, and recalculating the core-melt frequency due to changes in the frequency of the initiator as identified in Phase I (the exceedance frequency for mechanical- and maintenance-induced seal failures leading to leak rates in excess of makeup capacity).

A discussion of the plant response to the mechanical- and maintenance-induced RCP seal failures leading to a leakage within this category, the functional requirements for ECCS, and the event trees for the potential accident sequences follows.

5.2.1 ANO-1 Small-Small LOCAs

For leak rates exceeding normal makeup capacity (150 gpm), the pressurizer level and makeup tank level will drop simultaneously. It is expected that the makeup tank liquid will deplete before the pressurizer level reaches the height of the heaters. Hence, the failure of one of the makeup pumps due to low suction head is assumed. After failure of the makeup pump, the pressurizer pressure will drop to both the reactor trip and the emergency safeguard actuation setpoints if the emergency feedwater system is available. If it is not available, operator action to initiate the feed and bleed operation is required.

The LOCA-initiating events caused by RCP seal failures rarely exceed 400 gpm per pump. These ranges of leak rates are well covered within the equivalent break diameter of 0.38 to 1.2 in. (creating a break flow equivalent to ~150 to 1500 gpm). This size of LOCA-initiating event namely, Event B(1.2), is considered in Reference 11. The successful criteria for operation of ECCS as described in this reference are reproduced in Table 9. The associated event tree for a B(1.2)-initiating event, taken from the same reference, is given in Figure 15. The nomenclatures used for the event tree construction are given in Table 10.

5.2.2 CC-1 Small-Small LOCAs

For leak rates greater than normal makeup capacity (~100 gpm), the pressurizer pressure will drop to both the reactor trip and Engineered Safety Features Actuation setpoints. However, for the small ranges of LOCA (equivalent circular diameter of 0.3 to 1.9 in.), the rate of coolant loss through the small-small break is insufficient to remove enough decay heat to prevent a core melt and therefore secondary heat removal is required. This function is performed by a secondary system relief with an auxiliary feedwater system.

In the IREP study for Calvert Cliffs Unit 1, no credit was given for the possible use of primary system "Feed and Bleed" as opposed to ANO-1 IREP where "Feed and Bleed" was credited as an alternative way for cooling down the plant, given the failure of secondary heat removal. "Feed and Bleed" is not considered feasible at Calvert Cliffs because of the low shutoff head of the HPSI pump (~1275 psia) and the possibility that the primary pressure cannot be reduced sufficiently by opening the Power Operated Relief Valves (PORVs)

Table 9. ECCS Requirement
(Reproduced from Ref. 10)

LOCA Success Criteria

		INJECTION PHASE			RECIRCULATION PHASE		
LOCA Size	Reactor Subcriticality	Containment Overpressure Protection Due to Steam Evolution	Post Accident Radioactivity Removal	Emergency Core Cooling	Containment Overpressure Protection Due to Steam Evolution	Post Accident Radioactivity Removal	Emergency Core Cooling
7.9×10^{-4}-.008 ft^2 .38"-1.2"D Stuck Open ERV = .0056 ft^2 Max. Recorded RCP Seal Failure = .0035 ft^2	> 6 Control Rod Groups Inserted Into the Core by the Reactor Protection System (RPS)*	1/2 Reactor Bldg. Spray Injection (RBSI) OR 1/4 Reactor Bldg. Fan Coolers (RBCS)	1/2 RBSI	1/3 High Pressure Injection (HPIS) and 1/2 Safety/Relief Valves (SRV) OR 1/3 HPIS and 1/2 Emergency Feedwater (EFS)	1/2 Reactor Bldg. Spray Recirc. (RBSR) and Sump Mixing With 1/3 HPRS and 1/2 LPRS Heat Exchanger OR 1/4 RBCS	1/2 RBSR	1/3 High Pressure Recirc. (HPRS) and 1/2 LPRS Heat Exchanger OR 1/2 EFS (During Injection Phase) & 1/2 Decay Heat Removal System
.008-.015 ft^2 1.2-1.66"D Stuck Open P_z Safety = .0145 ft^2				2/3 HPIS and 1/2 SRV OR 1/3 HPIS and 1/2 EFS			1/3 HRPS and 1/2 LPRS Heat Exchanger
.015-.087 ft^2 1.66-4"D				1/3 HPIS			
.087-.55 ft^2 4-10"D				1/3 HPIS and 1/2 Low Pressure Injection (LPIS)	1/2 RBSR and Sump Mixing With 1/2 LPRS Heat Exchanger OR 1/4 RBCS		1/2 Low Pressure Recirc. (LPRS)
.55-1.0 ft^2 10-13.5"D	No System Needed			1/2 LPIS and 1/2 Core Flood Tanks (CFT)			
>1 ft^2 >13.5"D				1/2 LPIS and 2/2 CFT			

*The HPIS can perform reactor subcriticality by injecting borated water in the event of RPS failure. However, since operation of the HPIS cannot prevent the pressure transient associated with RPS failure, the HPIS should not be considered a reactor subcriticality front line system.

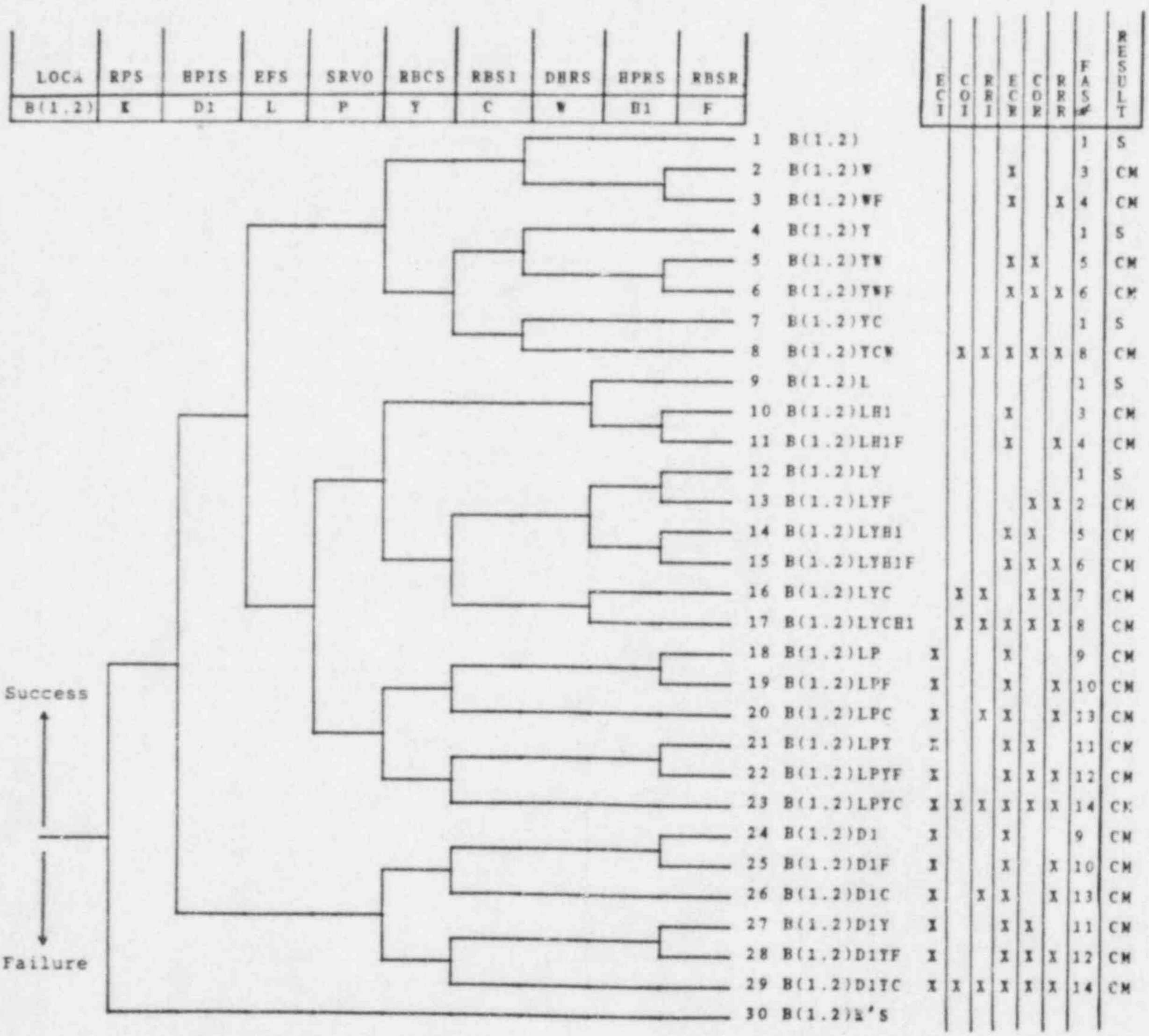

Figure 15. ANO-1 LOCA systemic event tree for breaks 0.38 in.<D≤1.2 in.
(Reproduced from Reference 10)

Table 10. Nomenclatures Used for Event Tree Development

K	- Reactor Protection System (RPS) Failure
D_1	- High Pressure Injection System Failure (HPIS)
L	- Emergency Feedwater System Failure (EFS)
P	- Failure of Pressurizer Safety Relief Valves to Open
Y	- Reactor Building Cooling System (RBCS) Failure
C	- Reactor Building Spray Injection System (RBSI) Failure
W	- Decay Heat Removal System (DHRS) Failure
H_1	- High Pressure Recirculation System (HPRS) Failure
F	- Reactor Building Spray Recirculation System (RBSR) Failure
ECI	- Emergency Core Cooling During Injection Phase
COI	- Containment Overpressure Protection During the Injection Phase
RPI	- Radioactivity Removal During the Injection Phase
ECR	- Emergency Core Cooling During Recirculation Phase
COR	- Containment Overpressure Protection During the Recirculation Phase
RRR	- Radioactivity Removal During the Recirculation Phase

within the short time available (10 minutes). In addition, there are no procedures at Calvert Cliffs for performing this action.

The LOCA-initiating events caused by unlikely failures of four stages of seals (including the vapor seal) in BJ reactor coolant pumps cannot exceed 600 gpm per pump (corresponding to critical flow through the 3/4 in. leak off line). These ranges of leak rates are well within the equivalent break diameters of 0.3 to 1.9 in., which are classed as small-small LOCAs (S_2) in the Calvert Cliffs IREP report.[12] The successful criteria for operation of ECCS as described in this reference are reproduced in Table 11. The associated event tree for an S_2 LOCA and nomenclature are given in Figure 16 and Table 12, respectively.

5.2.3 Indian Point 3, Small-Small LOCAs

For leak rates above normal makeup capacity (120 gpm) but less than the leakage resulting from a cold leg break of 2 in. equivalent diameter, the pressurizer pressure and level will drop to the scram setpoint. Further depressurization is expected subsequent to reactor scram and ECCS injection. System pressure will stabilize shortly after safety injection, at a level above the secondary side pressure relief setpoint and will be held at this level because of the balanced flow rates between safety injection and subcooled or saturated liquid going through the seal. The description of plant response and the modeling assumptions are detailed in Reference 20 for the various sizes of LOCA.

Table 11. LOCA Event Definition and Mitigating Systems Success Criteria for Calvert Cliffs Units 1 (Reproduced from Reference 11)

LOCA Size[1]	Mitigating Function[2]						
	Reactor Subcriticality (RESC)	Injection Phase			Recirculation Phase		
		Reactor Heat Removal (REHR)	Containment Atmospheric Heat Removal (CNHR)	Containment Radioactivity Removal (CNRR)[3]	Reactor Heat Removal (REHR)	Containment Heat Removal (CNHR)	Containment Radioactivity Removal (CNRR)[3]
Small-Small .3"<D*≤1.9"	RPS	1/3 HPSI AND SSR AND 1/2 AFW	1/2 CSSI OR 1/4 CARC[4]	1/2 CSSI	1/3 HPSR	1/2 CSSR with 1/2 SDHX OR 1/4 CARC	1/2 CSSR
Small 1.9"<D*≤4.3"	RPS	1/3 HPSI	1/2 CSSI OR 1/4 CARC[4]	1/2 CSSI	1/3 HPSR	1/2 CSSR with 1/2 SDHX OR 1/4 CARC	1/2 CSSR
Large D*<4.3"	None Required[5]	3/4 SITs AND 1/2 LPSI	1/2 CSSI OR 1/4 CARC	1/2 CSSI	1/3 HPSR	1/2 CSSR with 1/2 SDHX OR 1/4 CARC	1/2 CSSR

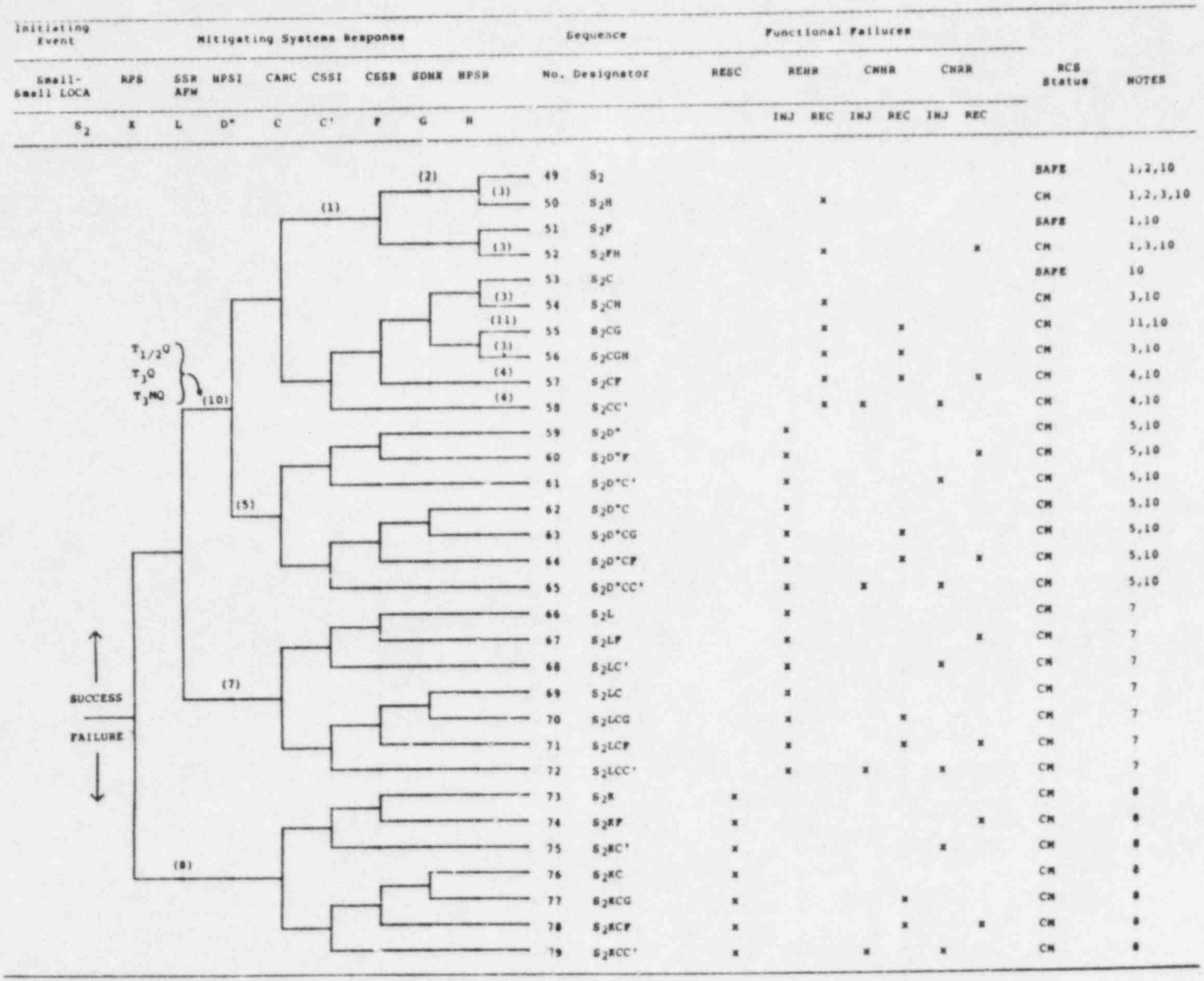

Figure 16. Small-small LOCA (S_2) systematic event tree (Calvert Cliffs 1) (Reproduced from Reference 11)

Table 12. Nomenclatures and Their Definitions for Calvert Cliffs Unit 1

D*	=	Equivalent Diameter of break in inches
RPS	=	Reactor Protection Systems
SIT	=	Safety Injection Tanks
HPSI	=	High Pressure Safety Injection
HPSR	=	High Pressure Safety Recirculation
LPSI	=	Low Pressure Safety Injection
LPSR	=	Low Pressure Safety Recirculation
SSR	=	Secondary Steam Relief (atmospheric dump valves or steam generator safety valves)
AFW	=	Auxiliary Feed Water
CSSI	=	Containment Spray System Injection
CSSR	=	Containment Spray System Recirculation
SDHX	=	Shutdown Cooling Heat Exchanger
CARC	=	Containment Air Recirculation and Cooling

The LOCA initiating events caused by RCP seal failures in W pumps rarely exceed 500 gpm per pump. Therefore, the ranges of leak rates induced by RCP seal failures will always be less than the equivalent break diameter of 2 in., which in the Indian Point Probabilistic Safety Study[13] (IPPSS), falls into the category of small-small LOCA. The successful operation of ECCS as described in this reference is reproduced in Table 13. The associated event tree for a small-small LOCA-initiating event is given in Figure 17. The nomenclature used for the event tree construction is given in Table 14. As seen from the event tree, the "Feed and Bleed" system is credited in IPPSS for Indian Point 3.

Table 13. IPPSS LOCA and Transient Mitigating Systems Success Criteria
(Reproduced from Reference 14)

LOCA SIZE	Emergency Core Cooling Early (RWST)	Emergency Core Cooling Late (SUMP)	Containment Overpressure Protection	Radioactivity Removal
0-2"	1/3 Safety Injection Pumps (SI) and 1/3 Auxiliary Feedwater Pumps (AFWS) <u>OR</u> 1/3 SI and 2/2 PORVs	1/3 SI and 1/2 RHR Pumps <u>OR</u> 1/3 SI and 1/2 Recirc. Pumps	1/2 Containment Spray Pumps <u>OR</u> 3/5 Containment Fans	1/2 Containment Spray Pumps
2-6"	2/3 SI and 1/2 RHR Pumps	2/3 SI and 1/2 RHR OR 2/3 SI and 1/2 Recirc. Pumps	Same	Same
>6"	3/4 Accumulators and 1/2 RHR Pumps	1/2 Recirc. Pumps <u>OR</u> 1/2 RHR Pumps	Same	Same
Steam Generator Tube Rupture	1/3 SI and RCS Depressurization	1/3 SI and 1/2 RHR <u>OR</u> 1/3 SI and 1/3 Recirc. Pumps	Same	Same

TRANSIENTS

Emergency Core Cooling Early (Secondary or RWST)	Emergency Core Cooling Late (Secondary or SUMP)	Containment Overpressure Protection	Radioactivity Removal
1/3 AFWS <u>OR</u> 1/3 SI and 2/2 PORVs	1/3 AFWS <u>OR</u> 1/3 SI and 1/2 RHR <u>OR</u> 1/3 SI and 1/2 Recirc.	Same	Same

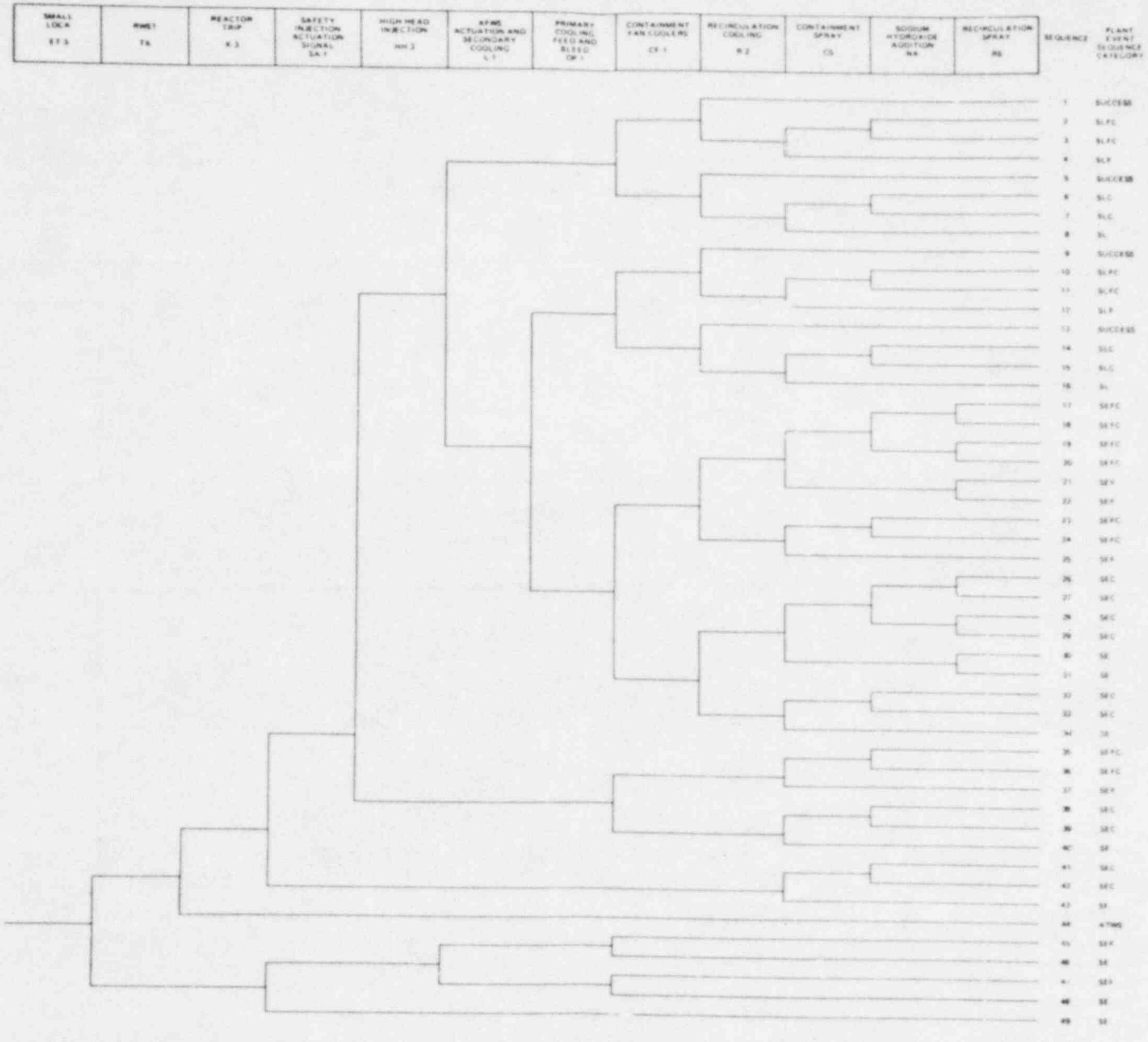

Figure 17. Small LOCA event tree for IP-3 (Reproduced from Reference 13)

Table 14: Nomenclatures Used for IP-3 Event Tree

S	Small LOCA
A	Large LOCA
T	Transient
L	Late Core melt
F	Containment Fan Cooling (Operating)
C	Containment Spray (Operating)
RWST	Refueling Water Storage Tank
ATWS	Anticipated Transient Without Scram

Note: Other nomenclatures are defined in Figure 17 and Table 13.

6.0 PHASE II - EVALUATION

The annual frequency of the mechanical- and maintenance-induced RCP seal failure leading to a primary leakage in excess of normal makeup capacity of the plant and leading into a core melt is estimated probabilistically in this chapter. The quantifications are primarily based on existing information and logic trees contained in the plant-specific PRAs for representative plants identified in Section 5. However, before results are presented and discussed one must take note of the following factors which affect the reported results.

Use of the mean values of the distributions for the primary faults are considered as consistent statistical measures for logic tree quantification. Therefore, the core-melt frequency estimated here may differ from those recorded in the reference PRAs (e.g., median values were used in ANO-1 IREP). The probabilities for recovery actions are incorporated into the calculations as point estimates and they basically correspond to the best estimates recorded in the referenced PRAs. The frequencies of initiating events, namely, small-small LOCAs caused by mechanical- and maintenance-induced RCP seal failures, are determined from the results of Phase I of this report.

The identification of the major contributors to the frequency of a core-melt sequence initiated by a mechanical- and maintenance-induced RCP seal failure depends on the number of dominant cutsets. If the results of the PRA quantification indicate that the frequency of a specific core-melt scenario is dominated by a small set of cutsets (usually 10 or less), the major contributors can then be easily identified. However, for a large set of dominant cutsets, the major contributors cannot be identified except through formal sensitivity/importance analyses. The limitations of this study do not allow for such analyses, and therefore, the dominant contributors in these cases are identified on the basis of the insights provided by the referenced PRA documents.

Finally, the uncertainty propagation is performed for the dominant minimal cutsets using the SAMPLE computer code[21] because of the limitations of the code in dealing with a large number of primary events.

6.1 ANO-1 RCP-Induced Core-Melt Frequency

The quantification of RCP-induced core-melt frequency for the ANO-1 plant is based on the event tree given in Figure 15 and the associated discussion in Section 5.2. The following pertinent information was used for quantification:

(1) The estimated frequency for the initiating event is based on the leak rate distribution for a generic B&W plant with four B-J RCPs exceeding 150 gpm. The frequency of mechanical- and maintenance-induced RCP seal failures leading to a small-small LOCA, B(1.2), is estimated with the mean of 2.1E-2 and upper and lower bounds of 6.3E-2 and 7.0E-3 (Figure 10).

(2) The data used for the quantification of the logic trees are from the IREP Procedures Guide.[22] The mean values were used for calculating the point estimate core-melt frequency.

(3) Selection of the dominant accident sequences was based on the numerical values given in the referenced PRA. The frequencies of these sequences were, however, reestimated using the mean values for the failure rates of the primary events and the modified frequency of the initiating events.

(4) The events associated with a station blackout leading to RCP seal failure were removed from the study because this event is not initiated by mechanical- and maintenance-induced RCP seal failures. The common mode failure event of the operator failing to start the HPI pumps and not recognizing the occurrence of excessive RCP seal leakages (the event HPI-PUMP-CM) was also removed from the logic trees because of uncertainty regarding the practicality of such a scenario.

The dominant accident sequences were identified using the point estimates recorded in the reference PRA, taking into account the slight modifications described above under item (4). It shall be emphasized that the sequence frequencies are estimated using the median probabilities of the primary faults and are only used for screening of dominant cutsets. For the major supercomponents, the mean values of primary faults were estimated and are presented in Table 15. The six dominant accident sequences identified are:

$B(1.2)\overline{K}D_1\overline{Y}C$: occurrence of a RCP seal LOCA with successful operation of RPS and RBCS* but failures of HPIS and RBSI with the estimated annual frequency of 4.0E-6 based on the median values and 7.2E-6 based on mean values.

$B(1.2)\overline{K}D_1YC$: similar to above, except RBCS is failed. The estimated annual frequency of this sequence is 6.4E-7 based on median values and 1.3E-6 based on mean values.

$B(1.2)\overline{KD}_1\overline{YCF}$: occurrence of a RCP seal LOCA concurrent with the failures of HPIS but successful operations of RPS, RBCS, RBSI, and RBSR with the estimated annual frequency of 8.3E-7 (note that event HPI-PUMP-CM was removed) based on median values and 2.3E-6 based on mean values.

$B(1.2)\overline{KD}_1L\overline{PY}H_1F$: the occurrence of RCP seal LOCA concurrent with the failure of EFS, RBSR, and HPRS, but successful operations of RPS, HPIS, RBCS, and "Feed and Bleed" with the estimated annual frequency of 1.0E-7 based on median values. The core-melt frequency based on mean values is not calculated.

$B(1.2)\overline{KD}_1L\overline{PY}H_1F"$: the same as above except RBSR operated successfully. The estimated annual frequency for this sequence is 8.8E-7 based on median values (mean values not calculated).

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*The codification of systems and sequences are defined in Tables 9 and 10.

Table 15. The Probabilistic Characteristic of Major Supercomponents

Name*	Probability Median	Measure Mean	Operator Recovery Failure Probability	Ratio of Mean Over Median
1-LF-HPI-H14	1.4E-2	2.5E-2	1	1.8
2-LF-SWS-VCH4B	1.9E-2	2.3E-2	0.01	1.2
3-LF-SWS-S14	1.0E-2	2.0E-2	0.01	2.0
4-LF-SWS-S5	1.0E-2	2.0E-2	0.01	2.0
5-LPI1407A-VCC-LF	8.4E-3	1.3E-2	0.23	1.5
6-HPI-PUMP-CM	1.0E-4	---	1	---
7-LF-SWS-S2	5.0E-3	1.0E-2	0.05	2.0
8-LF-ECS-ROOM100	4.9E-3	7.5E-3	0.01	1.5
9-LF-AC-B5	3.7E-4	7.5E-4	0.05	2.0
10-LF-AC-A3	3.7E-4	7.5E-4	0.23	2.0
11-LPI1408B-VCC-LF	8.4E-3	1.3E-2	0.23	1.5
12-LF-SWS-VCH4A	1.9E-2	2.3E-2	0.01	1.2
13-LF-LPI-L25	1.0E-4	1.2E-4	1	1.2
14-LF-SWS-S_1	5.0E-3	1.0E-2	0.01	2.0
15-LF-ECS-RO[illegible]99	4.9E-3	7.5E-3	0.01	1.5
16-LF-ESF-TB01	1.2E-3	1.3E-2	0.03	1.1
17-LF-ESF-TC01	1.2E-3	1.3E-2	0.03	1.1
18-LF-ESF-TA01	1.2E-3	1.3E-2	0.03	1.1
19-LF-AC-B6	3.7E-4	7.5E-4	0.23	2.0

*Note that the name of supercomponents is the same as those used in IREP study of ANO-1.[10] For the description of these events and their impact on mitigating systems, the reader may refer to Reference 10.

The major cutsets of these accident sequences and their associated frequencies are given in Table 16. Five cutsets contributes about 58% of the annual core-melt frequency initiated by the mechanical- and maintenance-induced RCP seal failures. These dominant cutsets and their associated frequencies are given in Table 17.

The overall core-melt frequency initiated by mechanical- and maintenance-induced RCP seal failures based on mean values of the primary event probabilities is estimated to be 1.3E-5. The mean value for the annual core-melt frequency reported by IREP study for ANO-1 excluding the RCP seal LOCAs is 7.9E-5.* Therefore, the expected fractional contribution in core-melt frequency due to mechanical- and maintenance-induced RCP seal LOCAs is estimated using the point estimates based on the mean values of the failure probability of primary faults is about 16.5%.

The computer code SAMPLE was also used for estimating the bounds of the core-melt frequency resulting from the dominant cutsets given in Table 17. The 90 percentiles of the core-melt frequency, excluding the uncertainty associated with RCP seal LOCA frequency, are estimated to be within 3.5E-6 and 2.0E-5 per year. The mean and median values of the distribution are 8.8E-6 and 7.3E-6, respectively.

6.2 Calvert Cliffs Unit 1 (CC-1) RCP-Induced Core-Melt Frequency

The quantification of RCP induced core-melt frequency for the CC-1 plant is based on the event tree given in Figure 15 and the associated discussion in Section 5.2. The pertinent information used for the quantification is discussed in the following:

(1) The estimated frequency for the initiating event is based on the leak rate distribution for a generic CE plant with four B-J RCPs exceeding 100 gpm. The frequency of mechanical- and maintenance-induced RCP seal failures leading to a small-small LOCA, S_2, is estimated with the mean of 1.5E-2 and upper and lower bounds of 4.5E-2 and 5.0E-3 (Figure 9). These estimates of RCP seal leakage frequency for CE/BJ four-stage pump seals are too conservative. As discussed in Chapter 4, no credit was given to the fourth stage vapor seal. If the vapor seal is not failed, the leak rates are expected to be small because of automatic isolation of controlled bleed off line. No event in which the four stages of RCP seals were failed was reported up to 1984. Therefore, an estimate cannot be made as to the frequency of such events. Given the failure of the vapor seal, the expected leak rates from CE/BJ four-stage seals are expected to be similar to those of other pump manufacturers. Until the adequacy of the vapor seal to stand the full reactor pressure is determined, this study has performed sensitivity analyses on RCP leak frequency. Factors of 1/2 and 1/5 are used for sensitivity

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*This value is calculated from Table 8-4, pg 8-60, Volume 1 of Reference 10.

Table 16. The Dominant Minimal Cutsets for RCP-LOCA Accident Sequences

Sequence: B(1.2)$\overline{K}$D1$\overline{Y}$C

Cutsets	Frequency Based on Mean	Frequency Based on Median
B(1.2)*LF-LPI-L2[illegible]	2.5E-6	2.1E-6 (1)
B(1.2)*LPI1407A-VCC-LF*LPI1408B-VCC-LF	3.4E-6	1.5E-6 (1)
B(1.2)*LPI1407A-VCC-LF*LF-SWS-S1	1.0E-6	3.4E-7 (.4)
B(1.2)*LPI1408A-VCC-LF*LF-SWS-S2	1.3E-7	4.5E-8 (.05)
B(1.2)*LPI1407A-VCC-LF*LF-SWS-VCH4A	7.2E-8	4.0E-8 (.01)
B(1.2)*LPI1408B-VCC-LF*LF-SWS-VCH4B	7.2E-8	4.0E-8 (.01)
TOTAL	7.2E-6	4.0E-6

Sequence: B(1.2)$\overline{K}$D1YC

Cutsets	Frequency Based on Mean	Frequency Based on Median
B(1.2)*LF-SWS-VCH4A*LF-SWS-VCH4B	1.6E-7	1.1E-7 (.01)
B(1.2)*LF-SWS-VCH4A*LF-SWS-S14	1.1E-7	4.8E-8 (.01)
B(1.2)*LF-SWS-VCH4A*LF-SWS-S5	1.1E-7	4.8E-8 (.01)
B(1.2)*LF-ESF-TB01*LF-EFS-TC01	1.0E-7	8.8E-8 (.03)
B(1.2)*LF-ESF-TA01*LF-EFS-TC01	1.0E-7	8.8E-8 (.03)
B(1.2)*LF-ESF-TA01*LF-EFS-TB01	1.0E-7	8.8E-8 (.03)
B(1.2)*LF-SWS-VCH4B*LF-SWS-S1	6.0E-8	2.5E-8 (.01)
B(1.2)*LF-SWS-VCH4B*LF-ECS-RØØM99	4.5E-8	2.5E-8 (.01)
B(1.2)*LF-SWS-VCH4A*LF-SWS-S2	6.0E-8	2.5E-8 (.01)
B(1.2)*LF-SWS-VCH4A*LF-ECS-RØØM100	4.5E-8	2.5E-8 (.01)
B(1.2)*LF-SWS-S14*LF-SWS-S1	4.0E-8	1.0E-8 (.01)
B(1.2)*LF-SWS-S14*LF-ECS-RØØM99	3.0E-8	1.0E-8 (.01)
B[illegible]2)*LF-SWS-S5*LF-SWS-S1	4.0E-8	1.0E-8 (.01)
B(1[illegible])*LF-SWS-S5*LF-ECS-RØØM99	3.0E-8	1.0E-8 (.01)
B(1.2)*LF-SWS-S2*LF-SWS-S1	1.0E-7	2.7E-8 (.05)
B(1.2)*LF-SWS-S2*LF-ECS-RØØM99	1.5E-8	5.2E-9 (.01)
B(1.2)*LF-SWS-S1*LF-ECS-RØØM100	1.5E-8	5.2E-9 (.01)
B(1.2)*LF-ECS-RØØM99*LF-ECS-RØØM100	1.2E-8	5.2E-9 (.01)
B(1.2)*LF-AC-B6*LF-SWS-VCH4B	5.0E-9	2.1E-9 (.01)
B(1.2)*LF-AC-B5*LF-SWS-VCH4A	5.0E-9	2.1E-9 (.01)
B(1.2)*LOSP-AC-DG1*LF-AC-DG2		NOT OF CONCERN
TOTAL	1.3E-6	6.4E-7

*The factors in parentheses indicate the failure probability of operator recovery action.

Table 16. (Continued)

Sequence: $B(1.2)\overline{KD}1\overline{LYCF}$		
Cutsets	Frequency Based on Mean	Median
B(1.2)*LF-HPI-H14*LF-SWS-VCH4B	1.5E-7	6.7E-8 (.01)*
B(1.2)*LF-HPI-H14*LF-SWS-S14	1.0E-7	2.9E-8 (.01)
B(1.2)*LF-HPI-H14*LF-SWS-S5	1.0E-7	2.9E-8 (.01)
B(1.2)*LPI-1407A-VCC-LF*LF-HPI-H14	1.5E-6	5.5E-7 (.23)
B(1.2)*HPI-PUMP-CM		NOT OF CONCERN
B(1.2)*LF-HPI-H14*LF-SWS-S2	2 6E-7	7.3E-8 (.05)
B(1.2)*LF-HPI-H14*LF-ECS-RØØM100	4.0E-8	1.5E-8 (.01)
B(1.2)*LF-HPI-H14*LF-AC-B5	2.3E-8	6.5E-9 (.05)
B(1.2)*LF-HPI-H14*LF-AC-A3	5.8E-8	1.6E-8 (.23)
TOTAL	2.3E-8	8.3E-7

Sequence: $B(1.2)\overline{KD}1L\overline{PY}H1F$		
Cutsets	Frequency Based on Mean	Median
B(1.2)*(LF-RBI-B1+LF-RBI-B9)*LF-SWS-VCH4B*LF-EFW-E4	NC†	2.5E-8 (.13)
B(1.2)*(LF-RBI-B1+LF-RBI-B9)*LF-SWS-VCH4B*LF-EFC-VCD2	NC	2.0E-8 (.13)
B(1.2)*(LF-RBI-B1+LF-RBI-B9)*LF-SWS-S14*LF-EFW-E4	NC	1.1E-8 (.13)
B(1.2)*(LF-RBI-B1+LF-RBI-B9)*LF-SWS-S5*LF-EFW-E4	NC	1.1E-8 (.13)
B(1.2)*(LF-RBI-B1+LF-RBI-B9)*LF-SWS-VCH4B*LF-EFC-D1D2	NC	9.9E-9 (.13)
B(1.2)*(LF-LPI-RØØM100*LF-SNS-VCH4B*LF-EFW-E4	NC	1.4E-9 (.02)
B(1.2)*(LF-RBI-B1+LF-RBI-B9)*LF-SWS-S14*LF-EFC-VCD2	NC	8.8E-9 (.13)
B(1.2)*(LF-RBI-B1+LF-RBI-B9)*LF-SWS-S5*LF-EFC-VCD2	NC	8.8E-9 (.13)
B(1.2)*(LF-RBI-B1+LF-RBI-B9)*LF-VCH4B*LF-EFW-E8	NC	8.8E-9 (.13)
B(1.2)*LF-LPI-L51*LF-SWS-S14*LF-EFW-E4	NC	1.9E-10 (.01)
B(1.2)*LF-LPI-L51*LF-SWS-S5*LF-EFW-E4	NC	1.9E-10 (.01)
TOTAL	NC	1.0E-7

*The factors in parentheses indicate the failure probability of operator recovery action.
†NC means not calculated.

Table 16. (Continued)

Sequence: $B(1.2)\overline{KD}1L\overline{PY}H1\overline{F}$		
Cutsets	Frequency based on mean	median
B(1.2)*LF-SWS-VCH4B*LF-EFW-E4	NC†	5.9E-8 (.01)*
B(1.2)*LF-SWS-VCH4B*LF-EFC-VCD2	--	4.6E-8 (.01)
B(1.2)*LF-SWS-S14*LF-EFW-E4	--	2.5E-8 (.01)
B(1.2)*LF-SWS-S5*LF-EFW-E4	--	2.5E-8 (.01)
B(1.2)*LF-SWS-VCH4B*LF-EFC-D1D2	--	2.3E-8 (.01)
B(1.2)*LF-SWS-VCH4B*LF-EFW-E8	--	2.0E-8 (.01)
B(1.2)*LF-SWS-S14*LF-EFC-VCD2	--	2.0E-8 (.01)
B(1.2)*LF-SWS-S5*LF-EFC-VCD2	--	2.0E-8 (.01)
B(1.2)*LF-SWS-S2*LF-EFW-E4	--	6.3E-8 (.05)
B(1.2)*LF-ECS-RØØM100*LF-EFW-E4	--	1.3E-8 (.01)
B(1.2)*LF-SWS-S2*LF-EFC-VCD2	--	5.0E-8 (.05)
B(1.2)*LF-ECS-RØØM100*LF-EFC-VCD2	--	9.9E-9 (.01)
B(1.2)*LF-SWS-S14*LF-EFC-D1D2	--	9.6E-9 (.01)
B(1.2)*LF-SWS-S5*LF-EFC-D1D2	--	9.6E-9 (.01)
B(1.2)*LF-SWS-S14*LF-EFW-E8	--	8.6E-9 (.01)
B(1.2)*LF-SWS-S5*LF-EFW-E8	--	8.6E-9 (.01)
B(1.2)*LF-SWS-S2*LF-EFC-D1D2	--	4.8E-9 (.01)
B(1.2)*LF-ECS-RØØM100*LF-EFC-D1D2	--	4.8E-9 (.01)
B(1.2)*LF-SWS-S2*LF-EFW-E8	--	4.4E-8 (.05)
B(1.2)*LF-ECS-RØØM100*LF-EFW-E8	--	4.4E-9 (.01)
B(1.2)*LF-AC-A3*LF-EFW-E4	--	6.1E-8 (1.)
B(1.2)*LF-AC-A3*LF-EFW-VCD2	--	4.8E-8 (1.)
B(1.2)*LF-ECS-B4*LF-EFW-E4	--	2.5E-8 (1.)
B(1.2)*LF-DC-D01*LF-EFW-E4	--	2.5E-8 (1.)
B(1.2)*LF-AC-A3*LF-EFC-D1D2	--	2.3E-8 (1.)
B(1.2)*LF-DC-D01*LF-EFC-ACBD4	--	2.3E-8 (1.)
TOTAL	NC	8.8E-7

*The factors in the parentheses indicate the failure probability of operator recovery action.
†NC - Not calculated.

Table 17. Major Accident Sequence Minimal Cutsets Initiated by RCP-Induced LOCA for ANO-1 Plant

Minimal Cutset	Accident Sequence	Operator Recovery Factor	Frequency Based on Median	Frequency Based on Mean
1. B(1.2)*LF-LPI-L25	B(1.2)D1C	1	2.1E-6	2.5E-6
2. B(1.2)*LPI1407A-VCC-LF*LPI1408B-VCC-LF	B(1.2)D1C	1	1.5E-6	3.5E-6
3. B(1.2)*LPI1407A-VCC-LF*-SWS-S1	B(1.2)D1C	0.4	3.4E-7	1.1E-6
4. B(1.2)*LPI1407A-VCC-LF*LF-HPI-H14	B(1.2)D1C	0.23	5.5E-7	1.6E-6
5. B(1.2)*LF-SWS-VCH4A*LF-SWS-VCH4B	B(1.2)D1YC	0.01	1.1E-7	1.1E-7
TOTAL	-----	---	4.6E-6	8.9E-6

analysis to credit the sealing operation of the vapor seal in the event that other seal stages has failed.

(2) The data and dominant accident sequences are basically the same as those in the referenced PRA which used mean values for the failure rates of the primary events, and, therefore, no major recalculation was necessary as in the previous example.

(3) A set of dominant cutsets were identified for the dominant accident sequences. These are reported in reference PRA. A small set of dominant contributors could not have been identified because of the large number of minimal cutsets reported. A systematic sensitivity/importance analysis could have provided some insights regarding the identification of dominant contributors as discussed earlier. However, for the purpose of this study, the dominant events were identified satisfactorily on the basis of insights provided in the reference PRA.

Two dominant accident sequences were identified using the point estimates recorded in the reference PRA. The discussions on these dominant accident sequences follow:

S_2-H: In this sequence, a small-small LOCA (S_2) occurs followed by successful scram and operation of AFW* and HPSI providing both secondary and primary system makeup. The failure of High Pressure System in Recirculation (HPSR) mode is assumed. Therefore, the lack of primary makeup due to failure of HPSR uncovers the core and core melt ensues. In this sequence, CARC and CSSR succeed and cool the containment. Because of the

*The codification of systems and sequences are defined in Table 12.

large number of cutsets of relatively equal value which comprise this sequence (Section 8.1.3.2, Volume 1, Reference 12), the dominant contributors are determined from engineering insights reported in Reference PRA.

About 25% of the sequence frequency is due to failures of HPSI pump No. 13 combined with failure of pump room cooling to ESF room No. 11. The failure of room cooling is estimated on the basis of all HPSI pumps running. For LOCAs of these sizes, all but one pump would be shut down. Therefore, the PRA estimates in this regard may be quite conservative.

About 40% of the sequence frequency is due to cutsets involving component cooling water (CCW) faults. The failure of CCW is assumed to cause failure of the HPSR pumps while in the recirculation mode because of hot coolant temperature flowing through the pumps and the loss of seal cooling. This assumption is conservative because the loss of cooling either may not fail the pumps or the catastrophic pump failures may occur after 2 hours.

The estimated core-melt frequency for this sequence using mean values for the primary events and its percentage contribution to the overall core-melt frequency are given in Table 18.

Table 18: The Major Core-Melt Sequences of Mechanical- and Maintenance-Induced RCP Seal Failures in Calvert Cliffs Unit 1

	P_f* = 1		P_f = 1/2		P_f = 1/5	
Sequence	Annual Frequency	Percentage† Contribution	Annual Frequency	Percentage Contribution	Annual Frequency	Percentage Contribution
S_2H	1.0E-5	10%	5.0E-6	5%	2.0E-6	2.0%
S_2FH	7.8E-6	7.8%	3.9E-6	3.9%	1.6E-6	2%
S_2D''	1.1E-6	1.1%	5.5E-7	0.5%	2.2E-7	ε
S_2L	1.3E-7	0.1%	ε	ε	ε	ε
TOTAL	1.9E-5	19%	9.5E-6	9.5%	4.E-6	4.0%

*P_f is the failure probability of vapor seal given the failure of other three stages seals in CE/BJ pumps.

†Percentage contribution is the sequence frequency divided by core-melt frequency excluding the core-melt sequences involving RCP seal LOCAs.

S_2-FH: This is similar to the sequence discussed above except for the failure of CSSR. Because of the large number of cutsets of relatively equal value which comprise this sequence (Section

8.1.4.1, volume 1 of Reference 12), the dominant contributors are determined on the basis of engineering insights reported in the PRA. Over 85% of frequency for this sequence involves cutsets with ESF pump room cooling failures. The comments regarding to the conservatism in the assumption of HPSR pump failures, given the loss of room cooling as discussed previously, applies here as well.

The core-melt frequency estimated for this sequence using mean values for the primary events and its percentage contribution to the overall core-melt frequency is given in Table 18 as well.

Two other sequences are important to small-small LOCAs, namely, S_2D'' and S_2L. The sequence S_2D'' is a small-small LOCA concurrent with the failure of HPIS. The sequence S_2L is a small-small LOCA concurrent with the failure of AFWS (keeping in mind that "Feed and Bleed" is not credited in this study). These two sequences were identified as part of dominant accident sequences, but the large probabilities assigned to the associated recovery actions minimized their contributions.

The overall core-melt frequency initiated by mechanical and maintenance induced RCP seal failures with no credit assigned to vapor seal based on the mean value for the probability of primary events is estimated to be 1.9E-5. The annual core-melt frequency reported by the IREP study for Calvert Cliffs Unit 1 excluding the RCP seal LOCA is 1.0E-4. Therefore, the fractional contribution in the core-melt frequency due to mechanical and maintenance induced RCP LOCAs is estimated to be about 19%. For the cases where the integrity of vapor seal is credited, result from the sensitivity analysis are also provided in Table 18.

6.3 Indian Point 3 (IP-3) RCP-Induced Core-Melt Frequency

The quantification of RCP induced core-melt frequency for the IP-3 plant is performed on the basis of the event tree given in Figure 17 and the associated discussion in Section 5.2.3. The pertinent information used for the quantification is discussed in the following:

(1) The estimated frequency for the initiating event is based on the leak rate distribution for a generic W-Plant with four W RCPs (Figure 7) exceeding 120 gpm. The frequency of mechanical- and maintenance-induced RCP seal failures leading to a small-small LOCA, S_2, is estimated with a mean of 1.1E-2 and upper and lower bounds of 3.3E-2 and 3.7E-3.

(2) The logic trees and data used are based on the review and evaluation of IPPSS made by Sandia National Laboratory.[14]

(3) The dominant accident sequences were identified on the basis of the numerical values given in Reference 14. The frequencies for these sequences were estimated using the mean values for the failure rates of the primary events and the modified frequency of the small-small LOCA-initiating events.

(4) The most dominant accident sequence for the IP-3 as identified in Reference 14 is the loss of CCW transient due to a large pipe break which leads to excessive leakage of primary coolant through RCP seals (due to loss of seal cooling). This study has not considered this particular accident sequence as being part of core-melt sequences caused by mechanical- and maintenance-induced RCP seal failures. Extended loss of CCW and the associated impact on core-melt frequency is part of another NRC project at BNL and, as such, will be addressed when the project is completed.

The dominant accident sequences that may be initiated by mechanical- and maintenance-induced RCP seal failures are

SLF: This represents a sequence of events starting with a small-small LOCA followed by successful reactor trip, safety injection, and secondary heat removal (AFWS or Feed and Bleed) but with failure of the high pressure recirculation system. IPPSS estimates that this sequence is the most likely cause of core melt for Indian Point 3 plant. However, disagreement is found in Reference 14 for the following reasons:

(a) the loss of CCW transient which is not considered in IPPSs is the most likely cause of core melt, and

(b) the data used for failure of safety injection pumps in recirculation mode are overestimated.

On the basis of suggested modifications by the Sandia staff, the failure of the high pressure recirculation system decreased from 4.1E-3 (as reported in IPPSS) to 1.2E-3. The major contributors of HPRS are hardware failures, namely, the dependent failures of any four pairs of motor-operated valves (MOVs) or three safety injection pumps. The hardware failure of the low pressure section in the high pressure recirculation mode contributes about 50% to the overall system unavailability.

SEFC: This represents a sequence of events starting with a small-small LOCA and successful reactor trip, but failure of high head safety injection system. The analysis performed in Reference 14 estimates the failure probability of the safety injection system to be 3E-4 instead of 1.3E-4, as reported in IPPSS. The two studies differ primarily in their treatment of common mode failures of the pumps and estimates of the failure rates. The dominant contributors identified are the common mode failure of all three pumps failing to start and run (about 60% contribution), and single failures in the common pump suction line from RWST (about 40% contribution), namely, check valve 847, motor-operated valve 1810, and manual valve 846.

The core-melt frequency initiated by mechanical- and maintenance-induced RCP seal failures is therefore estimated to be about 1.65E-5. The overall core-melt frequency excluding those initiated by RCP mechanical- and maintenance-induced RCP seal failures is 2.33E-4. Therefore, the fractional contribution to core-melt frequency due to mechanical- and maintenance-induced seal failures is about 7%. This low percentage is mainly the result of the high overall frequency of core melt estimated in Reference 14. About 60% of overall core-melt frequency is due to the accident sequences initiated by loss of CCW transient. We feel that this high frequency of core melt is not representative of all W-designed plants, but is specific to IP-3. The comparison of the PRA results for IP-3 and IP-2 regarding the core-melt frequency initiated by loss of CCW transient indicates a difference of a factor of about 4. This is mainly due to the connection of CCW with city water in IP-2 which does not exist in IP-3. In addition, a large conservatism is built into the quantification because of lack of knowledge and data mainly in the areas of a) the frequency of large pipe break, b) the assumption that charging and safety injection pumps will fail in 5 minutes following loss of lube oil cooling, and c) the failure of RCP seals in 30 minutes due to loss of CCW.

For the purpose of this study, a range is defined for the fractional contribution to core-melt frequency due to mechanical- and maintenance-induced RCP seal failures. The lower bound of this range is already estimated to be 7%. An upper bound of about 18% is calculated under the assumption that CCW transients do not contribute to core-melt frequency.

7.0 CONCLUSION (Phase II)

The impact of mechanical- and maintenance-induced RCP seal failures on plant safety was evaluated in this phase of study. It was understood at the early stages of the project that generic evaluations cannot be performed because of the vast differences in system designs for various plant vendors or even among the plants representing the same vendor but different vintages. Therefore, the study was limited to three specific nuclear power plants, namely, Arkansas Nuclear One Unit 1 (ANO-1), Calvert Cliffs Unit 1 (CC-1), and Indian Point Unit 3 (IP-3). These plants were selected on the basis of a) availability of PRA documents, b) inclusion of three PWR vendors in the U.S. and c) to the extent possible, the different RCP seal designs.

This study considered two categories of primary coolant leakages caused by mechanical- and maintenance-induced RCP seal failures, depending on the size of leak rates compared to the normal makeup capacity of the plant. For leak rates below the normal makeup capacity, the following issues were addressed.

(i) estimation of the maximum normal makeup flow rates;

(ii) determination of manual or automatic actions required to establish the maximum normal makeup flow;

(iii) potential ways for degradation or failure of normal makeup system during a RCP seal leakage, namely, the possibility that the normal makeup tank level will become depleted and the makeup pumps will be vapor bound; and

(iv) the plant response and the actuation of mitigation systems when the normal makeup system is failed and RCP seal leakages are below the normal makeup capacity.

It is qualitatively concluded that, the mechanical and maintenance induced RCP seal failures leading to primary coolant leakages within the normal makeup capacity cannot alone lead to any significant safety impact. The plant response to concurrent failure of the RCP seals and the normal makeup system is similar to that of small-small LOCA. However, the associated probability of such an event is expected to be much smaller than the probability of a small-small LOCA caused by mechanical and maintenance induced RCP seal failure. Comparison of the normal makeup systems of the three plants under study indicates that the operation of a normal makeup systems depends heavily on operator actions in ANO-1, in contrast to CC-1 and IP-3. Therefore, for plants similar to ANO-1, either a specific procedure or the automatic realignment of makeup pump suction from the makeup tank to BWST is recommended.

The safety impact of the mechanical- and maintenance-induced RCP seal leakages in excess of the plant's normal makeup capacity was evaluated through the formal PRA methodologies and they are provided in Table 19. The results obtained depend strongly on the existing PRA documents for these plants. Therefore, the awareness of the assumptions implemented in these PRAs, the level of detail, and the possible discrepancies among them is important in order to justify the results summarized in Table 19. The following general conclusions are in order:

a) The frequency of core melt that may be initiated by mechanical- and maintenance-induced seal failures estimated based on these three plants, given the vast differences existing among the plant designs and the PRA approaches, are within the range of 0.5E-5 to 2.0E-5 per year.

b) The dominant accident sequences of small-small LOCAs induced by RCP seal failures are generally caused by failure of HPRS and HPIS respectively. It should be noted that for ANO-1 the failure of HPRS by itself does not lead to core melt unless it is concurrent with the failure of the emergency feedwater system. It should also be emphasized that "Feed and Bleed" is considered a viable means for slow heat removal at ANO-1.

c) If the failure of the vapor seal is assumed conditional to failure of the other three seal stages for the CE/BJ pump designs, the frequency of seal-failure-induced core melt for CC-1 would be comparable to the other two plants. However, if a factor of 5 (or a conditional failure probability of 0.2) is credited for proper operation of the vapor seal under full reactor pressure, the core-melt frequency caused by RCP seal failures would be much smaller for CC-1 than for the other two plants.

In conclusion, the safety impact of mechanical- and maintenance-induced RCP seal failures measured in terms of percentage contribution to annual core-melt frequency is estimated to be between 16% and 18% for B&W and Westinghouse plants. For the CE plants, the percentage contribution for annual core-melt frequency would be dependent on the reliability of the vapor seal. If a failure probability of 0.2 is assigned to a vapor seal exposed to full reactor pressure, the percentage contribution to annual core-melt frequency would be 4%.

Table 19: The Design and PRA Specifications of the Three Representative Nuclear Power Plants Used in This Study

PLANT CHARACTERISTICS	ANO-1	CC-1	IP-3
GENERAL			
Utility	Arkansas Power & Light Co.	Baltimore Gas & Electric Co.	New York Power Authority
NSSS Supplier	Babcock & Wilcox	Combustion Engineering	Westinghouse
Architect/Engineer	Bechtel	Bechtel	UE&C°
Date of Commercial Operation	December, 74	April, 77	August, 76
Total Core Heat Output [MWt]		2560	3025
Net Elec. Capacity of Unit [MWE]	836	850	965
No. of RC Loops		2	4
RCS Normal Operating Pressure [psia]	2140	2250	2249.7
REACTOR COOLANT PUMP			
No. of RC Pumps	4	4	4
Type	vertical, single stage, limited leakage, centrifugal	vertical, limited leakage, centrifugal	vertical, single stage radial with bottom suction and horizontal discharge
Pump Manufacturer	Byron Jackson	Byron Jackson	Westinghouse
Pump Capacity [gpm]	88,000	81,200	89,700
Pump Head [ft]	362	243	272
Design Pressure [psia]	2,500	2,500	2,499.7
No. of Seals	3	4	3
Seal Type	Hydrodynamic	Hydrodynamic	Hybrid
Seal Injection	yes	no	yes
CCW Flow to Thermal Barrier Hx.	yes	yes	yes

Table 19 (Cont'd.)

PLANT CHARACTERISTICS	ANO-1	CC-1	IP-3
SAFETY SETPOINTS			
Reactor Trip (Low Przr. Press.) [psig]	1800	Variable Trip Set Point with min. of 1735.3	1800
Safety Injection (Low Przr. Press.) [psig]	1500	1600	1700
Safety Injection (Hi Contmt. Press.) [psig)	4	2.8	3.5
Containment Isol. Signal [psig]	30	4	23 (Phase B)
HIGH PRESSURE INJECTION SYSTEM			
No. of HPI Pumps	3	3	3
Type, HPI Pump	Horizontal, Centrifugal	Horizontal, Centrifugal	Horizontal, Centrifugal
Shutoff Head [ft]	NA	2390 (1275 psia)	3375
HPI Pump Design Flow Rate [gpm]	100 at 6200 ft Head	345	400
HPI Max. Flow Rate [gpm]	500 at 3300 ft Head	640	650
Design Head [ft]	3300	2500	2325
HPI Pumps Cooled by CCW	yes	yes	yes
Status of HPI Pumps on Loss of CCW During Injection Phase	Continues to operate	Continues to operate	Fails if there is no CCW inventory. (Does not require pumped CCW flow)
HIGH PRESSURE RECIRCULATION			
Realignment of HPI Pumps	Automatic switchdown to cont. sump	Automatic switch down to cont. sump	Manual alignment of HPI pump to LPI pump discharge

Table 19 (Cont'd.)

PLANT CHARACTERISTICS	ANO-1	CC-1	IP-3
Status of HPI Pumps on Loss of CCW During Recirculation Phase	Fails	Fails	Fails (requires pumped CCW flow)
AUXILIARY FEEDWATER SYSTEM			
No. of Pumps	2	2	3
Type of Drive	1 Turbine Driven 1 Motor Driven	2 Turbine Driven	1 Turbine Driven 2 Motor Driven
FEED & BLEED OPERATION			
Feed & Bleed Credited in the PRA Study	yes	no	yes
CHEMICAL & VOLUME CONTROL SYSTEM			
No. of Charging Pumps	Same as HPI Pump	3	3
Type of Charging Pump	Same as HPI Pump	Positive displacement	Positive displacement, variable speed
Normal Charg. Pump Discharge Flow [gpm]	45	44	87
Normal Charg. Pump Discharge Press. [psig]	250	2310	2385
Normal Letdown Flog [gpm]	45	40	75
Total Seal Injection Flow [gpm]	32	NA	32
Total Seal Water Return Flow [gpm]	NA	NA	12
Normal Charging Line Flow [gpm]	17	44	55
Normal Total Controlled Bleed-off [gpm]	4	4	NA
Internal Volume of VCT [ft^3]	600	NA	400

Table 19 (Cont'd.)

PLANT CHARACTERISTICS	ANO-1	CC-1	IP-3
Normal Liquid Volume of VCT [gallons]	400	NA	1000
Normal Operating Temp in VCT [°F]	~125	120	127
Design Pressure of VCT, Internal [psig]	100	75	75
Operating Pressure Range of VCT [psig]	NA	0 to 65	0 to 60
Normal Operating Pressure of VCT [psig]	12-35	50	15
Auto make up to VCT on Low Level	No - (manual action)	yes	yes
Re-alignment of Charging Pump Suction to RWST (BWST) on low-low VCT level	No - (manual action)	Automatic	Automatic
No. of Boric Acid Pumps	2	2	2
Boric Acid Pump Design Flow Rate [gpm]	25	143	75
No. of Demineralized Water Pumps (Used for make up to VCT)	NA	2	2
Capacity of Demin, Water Pump [gpm]	NA	NA	150
Status of Charging Pumps on Intiation of Safety Injection Signal	NA	Auto - start of all pumps and alignment to BA pump discharge	Auto - trip of all pumps
<u>PRA SPECIFICATIONS</u>			
Performed by:	SNL/IREP	SNL/IREP	Utility/SNL Review
Data Source	IREP and Plant Specific	IREP and Plant Specific	WASH-1400, plant specific and others

Table 19 (Cont'd.)

PLANT CHARACTERISTICS	ANO-1	CC-1	IP-3
Frequency of RCP Seal LOCA (per year)	2.1E-2	1.5E-2(3.E-3)*	1.1E-2
Credit for Feed and Bleed	yes	no	yes
Level of PRA	Core Melt	Core Melt	Core Melt and Risk
DOMINANT ACCIDENT SEQUENCES/ FREQUENCY			
S_2D^x	1.1E-5	1.1E-6 (2.2E-7)*	3.3E-6
S_2H^x	NA	1.8E-5 (3.6E-6)*	1.3E-5
S_2L^x	NA	1.37E-7(E)*	NA
S_2LHP^x	2.0E-6	ε^+	ε
Total RCP Induced Core-Melt Frequency Per Year	1.3E-5	1/9E-5(4.0E-6)	1.6E-5
Total Core-Melt Frequency Excluding RCP Seal LOCA Initiator	7.9E-5	1.0E-4	2.3E-4(9.E-5)$^+$
Incremental Contribution of RCP Seal LOCA to Core-Melt Frequency	16.5%	19% (4.0%)*	7% (18%)$^+$

°UE&C stands for United Engineers and Constructors (US)

* The values in the parenthesis are estimated under the assumption that, the failure probability of vapor seal conditional to the failure of the other three seal stages in CE/BJ pump design is 0.2 rather than 1.

x D, H, L and P stand for the failure of HPI, HPR, AFW and Feed and Bleed system respecti ely.

\+ These values are calculated by removing the contribution if CCW transient to overall core-melt frequency (references to the Section 6.3).

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APPENDIX A

Review of Major RCP Seal Leakage Events in Pressurized Water Reactors

The event descriptions for seven major RCP-seal leakages were reviewed by a panel of four engineers with background in the operation of PWRs and associated systems and components. The purpose was to identify the possible reasons for seal failures leading to excessive leakage and to determine qualitatively the means by which future occurrences of such events may be prevented. The following events are considered for the review:

(1) Seal failure in Arkansas Nuclear 1, May 1980, with a leak rate of about 350 gpm.

(2) Seal failure in H.B. Robinson, May 1975, with a leak rate of about 400 gpm.

(3) Seal failure in Oconee 2, January 1974, with a leak rate of about 90 gpm.

(4) Seal failure in Indian Point 2, July 1977, with a leak rate of about 75 gpm.

(5) Seal failure in Salem 1, October 1978, with a leak rate of about 35 gpm.

(6) Seal failure in Tihange 1, April 1983, with a leak rate of about 35 gpm.

(7) Seal failure in Tihange 2, October 1982, with a leak rate of about 40 gpm or more.

For each event, the panel members were requested to answer the following four questions:

(1) What was the primary cause of the seal failure?

(2) Did the operator take the proper actions?

(3) Were the subsequent failures of other seal stages and the resulting leak rates expected?

(4) Can the occurrence of such an event in the future be prevented by implementing certain modifications in the pump design, operational procedure, instrumentation, operator training, etc.?

The results of these questionnaires are provided in this Appendix. Even though the opinions of panel members in several areas are not consistent, the following recommendations can be gleaned from the responses:

(a) Assure the isolation of seal return line after seal failure.

(b) Assure the proper flow of seal injection and increase of seal flow to the pump with a failed seal.

(c) Assure sufficient heat removal capacity for thermal barrier heat exchanger to cool the escaped primary coolant (equivalent flow of primary coolant given failure of one seal and loss of seal injection flow).

(d) Assure uninterrupted CCW flow to heat exchanger. Operator shall take immediate action if the CCW flow is lost because of inadvertent isolation. The CCW flow may be slowly reestablished to prevent thermal shock.

(e) Assure proper inspection of the pump interior after temporary loss of CCW and seal injection. Keep a record of pump failures in order to identify the pump with higher than average seal failure frequency.

Response to Questionnaire

Plant: Arkansas Nuclear One
Date: May 1980

Panel Member Identification: 1

1. Inaccurate installation of the seal, specifically axial seal location, caused failure of the third-stage seal. Shaft excursion caused by break up of the upper (third) stage seal.

2. No, a delay of about one hour in closing the RCP return line and increasing the seal injection.

3. Cascade failures of the other seals are expected under complete break up of the third-stage seal. Overheating of the seals in the other pumps was caused due to delay in isolation of seal return line.

4. Two possible corrective actions:

   (a) Isolating the return line and increasing the seal injection flow shall be emphasized in operational procedure.

   (b) Design modifications may be incorporated for automatic isolation of seal return line, control of seal injection flow and the pump trip during seal failures with excessive leakage.

Panel Member Identification: 2

1. Inaccurate installation of the upper seal.

2. Yes, it appears so.

3. Subsequent failures are expected under these conditions.

4. Improvement in maintenance guidelines.

Panel Member Identification: 3

1. Damage to the third seal was too extensive and the cause could not have been identified properly.

2. Yes, as far as discussed.

3. The follow on failures are expected under such conditions.

4. No comment.

Panel Member Identification: 4

1. The upper seal stage carbon ring failure.

2. Late isolation of seal return line, slow shutdown and cool down.

3. The failure of other seal stages given the failure of the upper seal is not usually expected.

4. Can not identify one probable area; better operational procedure and maintenance training are recommended.

Plant: H.B. Robinson
Date: May 1975

Panel Member Identification: 1

1. It is my opinion that the problem with RCP "C" started several years before. The higher than average seal failure frequency in this pump makes me believe that the problem started in March 1971. On that date following a reactor trip, a loss of seal flow to "A" and "C" pumps resulted in bearing and seal damage to "A" pump. No adverse effect was found for "C" pump. However, the repeated failure of this pump after that time on, may indicate a sustained damage to the bearing which apparently worsened as the time passed by.

2. Three deficiencies were identified. Low capacity thermal barrier heat exchanger, the operator delay in reestablishing CCW (about 1.2 hours of delay) and the failure to isolate the return line.

3. Under these conditions, cascades failures of other seals are imminent.

4. The same comments as ANO-1 events. In addition, a proper failure record keeping should have identified the problem with "C" pump due to its comparatively higher failure rate.

Panel Member Identification: 2

1. Improper installation of the seal.

2. The operator did not take the seal failure seriously based on the similar past experiences.

3. The cascade failures were not expected.

4. No comment.

Panel Member Identification: 3

1. Failure of the seal No. 1 on RCP-C. The primary cause not known.

2. Yes, as far as discussed. However, new procedure demands that the operator shuts leak off isolation valve.

3. Yes.

4. As the utility indicated, a new procedure for shutting the leak off isolation valve was required.

Panel Member Identification: 4

1. A number of things could have caused the failure.

2. Continued operation with degraded seal condition, defeating the CCW containment isolation valve, failure to maintain adequate subcooling and late shutting of seal return line.

3. The complex interaction between the seal failure and isolation of CCW causing common mode failure of all RCP seals was not anticipated as part of design. However, under these conditions the cascade failures are expected.

4. Can not be determined.

Plant: Oconee 2
Date: January 1974

Panel Member Identification: 1

1. Isolation of all seal injection lines coupled with low capacity seal cooling heat exchanger.

2. Given the lack of proper instrumentations, proper operator action seems to have been taken. It is not understood why the core flood tanks were vented to quench tank.

3. Yes.

4. It seems the utility has taken proper actions in regard to increasing the seal cooling heat exchanger capacity and addition of leak diagnostic instrumentations. It is believed that similar failures of this nature will not repeat.

Panel Member Identification: 2

1. Foreign materials.

2. Yes.

3. No, the undersize heat exchanger is not anticipated.

4. Maintenance procedure may be improved.

Panel Member Identification: 3

1. Excessive seal cavity temperature (570°F).

2. Yes.

3. No comment.

4. Increase the heat exchanger capacity, TIC stuffing box, TIC-leakage from upper seal. Proper actions have been taken.

Panel Member Identification: 4

1. Isolation of seal supply flow to all RCPs while operating at power.

2. Not enough information to comment on.

3. No comment.

4. No comment.

Plant: Indian Point 2
Date: July 1977

Panel Member Identification: 1

1. Unknown, possibility of foreign materials.

2. Yes, operator did take proper actions.

3. There is no indication of cascade failure, however a leak rate of 75 gpm resulting from the failure of No. 1 seal seems to be too high. It is possible that some of the "O" rings in seal No. 1 failed but were not reported. If the "O" rings have failed, then this type of cascade failure is not expected.

4. Not enough information for making proper comments.

Panel Member Identification: 2

1. Foreign materials.

2. Yes.

3. Yes.

4. No.

Panel Member Identification: 3

1. Foreign materials.

2. Yes.

3. Not applicable.

4. Control the seal injection water quality to eliminate contamination problem.

Panel Member Identification: 4

1. Foreign materials.

2. Yes.

3. Not applicable.

4. No comment.

Plant: Salem-1
Date: October 1978

Panel Member Identification: 1

1. At system pressure less than 100 psi the RCP No. 1 seal leak off valves remained open, causing a back flush of foreign materials from the seal leak off filter to the seals.

2. Operator actions were appropriate.

3. Failure of No. 1 and 2 seals were expected. The leak rate of about 35 gpm is also anticipated.

4. The design change of installing a check valve in RCP seal No. 1 return line is appropriate.

Panel Member Identification: 2

1. Can not be determined.

2. Yes.

3. Insufficient information.

4. No.

Panel Member Identification: 3

1. Back flush foreign matter from seal leak off filter back into seal cavity at low system pressure.

2. Yes.

3. Not applicable.

4. New ΔP inst. 200 psid minimum No. 1 seal. Less than 100 psig close No. 1 seal return and bypass valve. New check valve in each RCP.

Panel Member Identification: 4

Basically no comments were made.

Plant: Tihange 1
Date: April 1983

Panel Member Identification: 1

1. Incorrect mounting of the "O" ring (equivalent to double ΔP-"O" ring in Westinghouse pumps) between the aluminium oxide ring and its support after its inspection.

2. Operator actions were proper except for a 7 minute delay in increasing the seal injection flow after observing excessive leakage.

3. Failure of No. 2 and 3 seals are expected due to transportation of chipped materials from No. 1 failed seal surface along the direction of seal injection flow.

4. The indicated corrective actions taken subsequent to this event by no means prevent the future occurrences of the events with similar nature.

Panel Member Identification: 2

1. Incorrect mounting of "O" ring.

2. Yes.

3. Yes.

4. No.

Panel Member Identification: 3

1. Incorrect remounting on "O" ring.

2. Yes.

3. Yes.

4. Better Installation guidelines required.

Panel Member Identification: 4

No comments.

Plant: Tihange 2
Date: November 1982

Panel Member Identification: 1

1. Running the pumps without seal injection and cooling for 8 minutes. Also, several times the pumps were operated with No. 1 seal return line isolated, causing higher pressure differential in No. 2 seal.

2. Operator actions seem to be proper except he did not make sure that the No. 1 seal return line is isolated.

3. Failure of seal ages in other pumps indicate the high possibility of common mode fail es of all RCPs when they have experienced similar transients.

4. The corrective action taken were mainly to assure uninterrupted seal injection flow under emergency plant cooling. This action by itself would not be sufficient to prevent future failures. The inspection of pump seals after simultaneous loss of CCW and seal injection, even for a short period of time, is strongly recommended.

Panel Member Identification: 2

1. Lack of seal cooling.

2. Yes.

3. Yes.

4. No.

Panel Member Identification: 3

1. Run without seal water.

2. Yes.

3. Yes.

4. Addition of safety relief valve on seal return line.

Panel Member Identification: 4

No comments.

APPENDIX B

Summary Data Sheets for PWRs and BWRs

A very brief description of each seal failure data is provided in this Appendix. Each plant is identified by name and unit number, type, vendor, number of pumps, pump designer, pump model number, and number of seal stages. The minimum information provided for each event is the date, the pump identification code, the nature of failure, the maximum leakage per minute, and the total leakage in gallons. This information is judged to be sufficient for identification and retrieval of the events of interest.

Table B.1 and B.2 are the summary data for BWRs and PWRs, respectively.

Table B.1. Summary of BWRs Seal Failure Events

1. Browns Ferry 1, BWR, GE, 2, BJ, 28 x 28 x 35, 2
   01/14/80; RCP-A; Seal Failure; Small; Small
   03/22/80; RCP-B; Seal Failure; Small; Small

2. Brunswick 1, BWR, GE, 2, B, 28 x 28 x 32 - RV, 2
   06/24/77; RCP-1A; Second Stage Seal Failure; Small; Small
   11/13/77; RCP-1A; Seal Failure; Small; Small
   11/21/77; RCP-1A; Seal Failure; Small; Small
   11/04/77; RCP-1B; Outer Seal Failure; Small; Small
   12/01/79; RCP-1B; Seal Failure; 7.4; 10,656
   12/02/81; RCP-1A; Seal Failure; Small; Small
   01/21/82; RCP-1B; Seal Failure; Small; Small
   02/08/84; RCP-1B; Seal Failure; Small; Small
   02/--/78; RCP-1A and 1B; Seal Failure; Small; Small

3. Brunswick 2, BWR, GE, 2, B, 28 x 28 x 32-RV, 2
   05/27/76; RCP-2B; Seal Failure; Small; Small
   09/05/75; RCP-2B; Seal Failure; 50; 2600
   08/--/75; RCP-2A; Seal Failure; NA; 1500
   09/--/76; RCP-2B; Seal Failure; Small; Small
   11/--/76; RCP-2B; Seal Failure; Small; Small
   12/07/76; RCP-2B; Seal Failure; 27; NA
   03/30/76; RCP-2B; Seal Failure; Small; Small
   01/13/83; RCP-2B; Lower Seal Failure; Small; Small

4. Cooper, BWR, GE, 2, BJ, DVSS, 2
   09/12/79; RCP-B; Seal Failure; 5; NA

5. Dresden 2, BWR, GE, 2, BJ, DVSS, 2
   02/20/74; RCP-A & B; Seal Failure; 5.4; NA
   12/--/72; RCP-?; Seal Carbon Ring; Small; Small

6. Dresden 3, BWR, GE, 2, BJ, DVSS, 2
   10/27/82; RCP-3B; Rotating Seal Failure; Small; Small
   08/--/76; RCP-3B; Seal Failure; Small; Small

7. Duane Arnold, BWR, GE, 2, BJ, NA, 2
   08/21/74; RCP-B; Seal Failure; 5; NA

8. Fitzpatrick, BWR, GE, 2, BJ, 28 x 28 x 30-DVSS, 2
   04/01/81; RCP-A; Seal Failure; 20; NA

9. Hatch 1, BWR, GE, 2, BJ, DVSS, 2
   01/01/81; RCP-1A; Seal Failure; 5; NA

10. LaCrosse, BWR, GE
   02/24/81; RCP-1B; Seal Failure; Small; Small
   01/--/71; RCP-All; Seal Failure; Small; Small
   08/--/70; RCP-All; Seal Failure; Small; Small

Table B.1 (Cont'd)

11. LaSalle, BWR, GE, 2, B, NA, 2
08/21/82; RCP-1A; Seal Failure; 27; NA

12. Millstone 1, BWR, GE, 2, BJ, DVSS, 2
09/11/72; RCP-?; Seal Failure; 5.6; NA
11/25/83; RCP-A; Seal Carbon Rings; 25; NA
11/09/76; RCP-A; Labyrinth Seal*; NA; NA
11/27/76; RCP-B; Seal Failure; Small; Small

13. Monticello, BWR, GE, 2, GE, 2, B, ?-RV, 2
02/--/71; RCP-?; Seal Failure; Small; Small
03/--/71; RCP-11 & 12; Seal Failure; Small; Small
08/--/71; RCP-11 & 12; Seal Failure; Small; Small
04/19/80; RCP-12; Seal Compression Ring; Small; Small
02/--/76; RCP-12; Seal Failure; Small; Small
03/--/80; RCP-11 & 12; Seal Failure; Small; Small

14. Nine Mile Point 1, BWR, GE, 5, BJ, ? DVSS, 2
11/03/72; RCP-?; Seal Failure; 5; NA
11/09/76; RCP-1A; Defective Seal; Small; Small
10/13/77; RCP-1A; Seal Leakage; Small; Small
02/10/82; RCP-1A & 1B; Seal Failure; Small; Small
07/21/83; RCP-1A; Seal Failure; Small; Small
05/09/77; RCP-1B; Seal Failure; Small; Small
10/22/77; RCP-1C; Seal Leakage; Small; Small
05/11/78; RCP-1D; Seal Failure; Small; Small
11/06/77; RCP-1E; Seal Failure; 5; NA

15. Oyster Creek, BWR, GE, 5, BJ, 26 x 26 x 30 DVSS, 2
03/--/79; RCP-D; Seal Failure; Small; Small
10/14/74; RCP-B; Seal Failure; Small; Small
11/24/82; RCP-A; Seal Failure; Small; Small

16. Peach Bottom 2, BWR, GE, 2, BJ, ? DVSS, 2
04/22/81; RCP-?; 2nd Stage Seal Failure; Small; Small
01/26/82; RCP-A; Inner Mechanical Seal; Small; Small

17. Peach Bottom 3, BWR, GE, 2, BJ, DVSS, 2
03/30/82; RCP-3A; Second Stage Seal; Small; Small
01/--/75; RCP-B; Second Stage Seal; Small; Small

18. Pilgrim 1, BWR, GE, 2, BJ, NA, 2
07/10/81; RCP-A; Seal and Flange "O" Ring; 5.22; NA

19. Quad Cities, BWR, GE, 2, BJ, NA, 2
02/--/77; RCP-1A; Seal Failure; Small; Small
09/--/77; RCP-1A; Seal Failure; Small; Small

20. Vermont Yankee, BWR, GE, 2, BJ, DVSS, 2
09/15/82; RCP-1A; Outer Seal Failure; Small; Small

Table B.2. Summary of PWR Seal Failure Events

1. Arkansas Nuclear 1, PWR, B&W, 4, BJ, DFSS-33 x 33 x 38, 3
   08/--/74 to 07/--/76; Three Seal Failures
   08/27/76; RCP-D; Seal Failure; 25; NA
   12/03/77; RCP-C; Seal Failure; 6; NA
   11/--/77; RCP-B & C & D; Seal Degradation; Small; Small
   05/10/80; RCP-C; Third Stage Seal Failure; 400; 60,000
   08/08/82; RCP-C; Seal Failure; 28; NA
   08/07/76; RCP-C & B; Seal Failure; Small; Small
   01/01/79; RCP-C & B; Seal Failure; Small; Small
   12/01/80; RCP-C; Seal Failure; Small; Small
   09/03/76; RCP-B; Lower Seal Failure; Small; Small

2. Beaver Valley, PWR, W, 3, W, NA, 3
   08/30/77; RCP-1C; Seal Failure; Small; Small
   01/07/81; RCP-1C; Seal Degradation; Small; Small
   12/06/78*; RCP-1C; Gasket Failure; NA; NA

3. Calvert Cliffs 1, PWR, CE, 4, BJ, NA, 4
   08/22/75; RCP-11B; Middle; Upper and Vapor Seal; 2.7; Small
   12/21/79; RCP-11B; Middle Seal; Small; Small

4. Calvert Cliffs 2, PWR, CE, 4, BJ, NA, 4
   01/20/79; RCP-22A; Seal Failure; 2; Small
   12/27/78; RCP-21B; Seal Failure; Small; Small
   12/28/80; RCP-21A; Seal and "O" Ring Failure; NA; NA

5. Crystal River 3, PWR, B&W, 4, BJ, NA, 3
   03/01/79; RCP-1C; Seal Failure; Small; Small
   03/15/79; RCP-1B; Seal Failure; Small; Small
   04/15/79; RCP-1A & 1D; Seal Failure; Small; Small

6. Cook 1, PWR, W, 4, W, NA, 3
   11/19/75; RCP-1; No. 1; Seal Failure; Small; Small

7. Davis Besse 1, PWR, B&W, 4, BJ, 1F6226-2, 3
   08/16/78; RCP-1-1; Seal Failure; Small; Small
   10/09/78; RCP-?; Broken Seal Locking Wire; 1.56; Small
   01/09/79; RCP-1-1 & 1-2; Seal Failure; Small; Small
   04/12/79; RCP-1-2; Seal Failure; Small; Small
   10/23/79; RCP-1-1; First and Second Stage Seal; Small; Small
   (Loss of CCW for 26 minutes)
   10/01/78; RCP-?; Seal Failure; Small; Small
   01/06/81; RCP-2-1; Seal Failure; Small; Small
   01/02/81; RCP-1-2; Seal Failure; Small; Small
   05/21/81; RCP-2-1; Lower Seal Failure; Small; Small
   09/05/83; RCP-2-2; Seal Failure; Small; Small
   06/29/83; RCP-1-1; Seal Failure; Small; Small
   02/13/82; RCP-1-2; Seal Failure; Small; Small
   06/23/82; RCP-2-2; Seal Failure; Small; Small
   11/10/82; RCP-1-2; Seal Failure; Small; Small

Table B.2 (Cont'd)

8. Fort Calhoun 1, PWR, CE, 4, BJ, NA, 4
   04/17/74; RCP-? & ?; Loss of CCW; Small; Small
   09/10/75; RCP-A & B & C & D; Vapor Seal Failure; Small; Small
   05/16/80*; RCP-A & B & C; Gasket Failure; NA; NA

9. Ginna, PWR, W, 3, W, NA, 3
   ?/?/71; RCP-?, ?; No. 2 & 3 Seal Failures; Small; Small
   05/--/69; RCP-?, ?; Seals, NA, NA
   (LOSP for 45 minutes)

10. H.B. Robinson, PWR, W, 3, W, V11001-B1,3
    03/14/71; RCP-A & B & C; Seals & Bearings; NA; NA
    (LOSP and Loss of Seal Injection)
    01/14/76; RCP-A; Seal Failure; Small; Small
    04/--/75; RCP-?; No. 1 Seal Failure; 6; NA
    06/--/74; RCP-C; No. 1 Seal Failure; Small; Small
    05/01/75; RCP-C; No. 1 & 2 & 3 Seal Failure; 400 to 500;
    130,000 to 200,000 (Loss of CCW for 1 hour and 12 minutes)
    02/--/77; RCP-C; No. 1 Seal Failure; Small; Small
    04/--/77; RCP-C; No. 2 & 3 Seal Failure; Small; Small
    (Loss of Seal Injection)

11. Connecticut Yankee (Haddam Neck), PWR, W, 3, W, SV-4M-AI, 3
    07/--/69; RCP-4; Seal Failure; Small; Small
    (LOSP at the same month)
    08/21/77; RCP-2; Seals 1 & 2 & 3 plus "O" Ring; NA, 4020
    03/24/78; RCP-4; Seal No. 1; Small; Small
    07/14/79; RCP-3; Seals No. 1 & 2; Small; Small
    09/30/79; RCP-2; Seal Failure; Small; Small

12. Indian Point 2, PWR, W, 3, W, V11002-A1, 3
    08/02/75; RCP-21; Seal Failure; Small; Small
    08/03/75; RCP-22; No. 1 Seal and Anti-rotation Pin; Small; Smal
    08/22/76; RCP-all; Seal Failure; NA; NA
    07/02/77; RCP-23; Seal Failure; 75; 90,000
    11/11/81; RCP-23; No. 1 Seal and Anti-rotation Pin; 5; NA

13. Farley 1, PWR, W, 4, W, NA, 3
    01/05/78; RCP-?; Seal Degradation; 0.4; Small
    10/21/80; RCP-?; Seal Degradation; Small; Small
    10/10/77; RCP-?; No. 1 Seal Failure; Small; Small

14. Maine Yankee, PWR, CE, 3, BJ, DFSS, 3
    07/--/72; RCP-? & ?; No. 3 Seal; Small; Small
    12/08/80; RCP-1-1 & 1-2; Seal Failure; Small; Small
    12/16/80; RCP-1-2; Seal Failure; Small; Small
    11/18/83; RCP-1-1, Seal Failure; Small; Small
    11/27/83; RCP-1-3, Seal Failure; Small; Small
    12/03/83; RCP-1-3, Seal Failure; Small; Small

Table B.2 (Cont'd)

14. Maine Yankee, PWR, CE, 3, BJ, DFS, 3 (Cont'd)
07/07/82; RCP-all, Seal Failure; Small; Small
08/23/82; RCP-all, Seal Failure; Small; Small
12/01/82; RCP-1-1, Seal Failure; Small; Small

15. Millstone 2, PWR, CE, 4, BJ, NA, 4
02/26/76; RCP-40A; Seal Failure; Small; Small
07/22/76; RCP-40D; Seal Failure; Small; Small
11/29/79; RCP-40D; Upper Seal Failure; Small; Small
06/23/80; RCP-40A; Rolled up "O" Ring and Middle Seal Failure; Small; Small

16. North Anna 1, PWR, W, 3, W, NA, 3
12/27/80; RCP-A, No. 2 Seal Failure; 12.73; NA

17. Oconee 1, PWR, B&W, 4, W, NA, 3
01/13/75; RCP-All; Seal Failure; Small; Small
02/21/75; RCP-1A1, Seal Failure; Small; Small
04/--/77; RCP-All, Excessive Seal Leakage; Small; Small

18. Oconee 2, PWR, B&W, 4, Bing, NA, 3
01/22/74; RCP-2B2; Seal Failure; 90; 50,000
04/30/74; RCP-?, Upper Seal Failure; Small; Small
02/05/75; RCP-?, Seal Failure; Small; Small

19. Oconee 3, PWR, B&W, 4, Bing, 28 x 28 x 41-RQV, 3
04/--/75; RCP-3B1 & 3A1; Seal Failure; Small; Small
(Loss of Load Test)
06/17/75; RCP-?; Seal Failure; Small; Small
02/04/81*; RCP-3A1; Gasket Failure; NA; NA
4 more Seal Failures in 05/--/75, 09/22/75, 02/--/76, 09/--/76

20. Palisades, PWR, CE, 4, BJ, NA, 4
04/05/78; RCP-50B; Seal Failure; Small; Small
09/03/78; RCP-50B & 50D; Seal Failure; Small; Small
02/26/81; RCP-50A, Middle Seal Failure; Small; Small
08/09/81; RCP-50A, Seal Failure; Small; Small
09/29/80; RCP-50B, "O" Ring Failure; Small; Small
11/22/80; RCP-50B, Lower, Middle & Upper Seal Failure; Small; Small

21. Point Beach 1, PWR, W, 2, W, V11001-A1, 3
09/--/71 & 06/--/72; Several problems with seal cavity temperature instability and oscillation of leak off flow.
10/--/72; RCP-B, Seal Failure; Small; Small
01/26/75; RCP-A & B, Seal Stages No. 1 & 2 & 3; Small; Small
(Did not maintain hydrogen concentration)

Table B.2 (Cont'd)

22. Prairie Island 1, PWR, W, 2, W, NA, 3
10/06/74; RCP-11; Seal Failure; Small; Small
03/--/76; RCP-11 & 12, Seal Failure; Small; Small

23. Prairie Island 2, PWR, W, 2, W, W11001-B1, 3
10/15/75; RCP-22; Seal Failure; Small; Small
06/11/81; RCP-?, Shaft Bent and Seal Failure; 4.2; NA

24. Rancho Seco, PWR, B&W, 4, Bing, NA, 3
10/--/75; RCP-all; Seal Failure; NA; NA
02/--/76; RCP-?, Seal Failure; Small; Small

25. Salem 1, PWR, W, 4, W, NA, 3
03/18/78; RCP-?; Seal Failure; Small; Small
05/22/78; RCP-12; Seal Failure; Small; Small
10/21/78; RCP-13; No. 1 Seal Failure; NA; 15,000

26. San Onofre 1, PWR, W, 3, W, NA, 3
06/--/73; RCP-all; Seal Failure; Small; Small

27. Surry 1, PWR, W, 3, W, W-11009-A1, 3
01/16/84; RCP-1A & 1B; Seal Failure; NA; NA

28. St. Lucie, PWR, CE, 4, BJ, NA, 4
06/11/80; Loss of CCW for 1 1/2 hours
04/15/77; Loss of CCW for 30 minutes
04/18/84; RCP-1B2; Third Stage Seal; Small; Small
05/07/84; RCP-1B2; Third Stage Seal; Small; Small

29. Turkey Point 3, PWR, W, 3, W, V110001-B1, 3
12/27/83; RCP-3B; Shaft Bent and Seal Failure; Small; Small

30. Turkey Point 4, PWR, W, 3, W, NA, 3
09/--/75; Third Stage Seal; Small; Small

31. Zion 1, PWR, W, 4, W, W-11001-C1, 3
01/--/76; RCP-?; Third and First Stage Seal; 5; NA
06/10/75; RCP-all; Seal Failure; Small; Small
05/24/75; RCP-210 & 310; Seal Failure; Small; Small

32. Zion 2, PWR, W, 4, W, NA, 3
01/--/76; RCP-all; First and Second Seal Stages; Small; Small
06/--/76; RCP-2D & ?; No. 1 Seal Failure; Small; Small

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16. ABSTRACT (200 words or less) This document presents an investigation of the safety impact resulting from mechanical- and maintenance-induced reactor coolant pump (RCP) seal failures in nuclear power plants. A data survey of the pump seal failures for existing nuclear power plants in the U.S. from several available sources was performed. The annual frequency of pump seal failures in a nuclear power plant was estimated based on the concept of hazard rate and dependency evaluation. The conditional probability of various sizes of leak rates given seal failures was then evaluated. The safety impact of RCP seal failures, in terms of contribution to plant core-melt frequency, were also evaluated for three nuclear power plants. For leak rates below the normal makeup capacity and the impact of plant safety was discussed qualitatively, whereas for leak rates beyond the normal make up capacity, formal PRA methodologies were applied.

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