NUREG/CR-5268
BNL-NUREG-52177

Aging Study of Boiling Water Reactor Residual Heat Removal System

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Prepared for
U.S. Nuclear Regulatory
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Manuscript Completed: March 1989
Date Published: June 1989

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NRC FIN A3270

ABSTRACT

A study of the aging effects on Residual Heat Removal (RHR) systems in Boiling Water Reactors (BWRs) was performed as part of the Nuclear Plant Aging Research (NPAR) program. The objectives of the NPAR program are to provide a technical basis for the identification and evaluation of degradation caused by age in nuclear power plant applications. The information from this and other NPAR studies will be used to assess the impact of aging on plant safety and to develop effective mitigating actions.

The effects of aging in the RHR system were characterized using the Aging and Life Extension Assessment Program (ALEAP) Systems Level Plan developed by Brookhaven National Laboratory. Failure data from various national data bases were reviewed and analyzed to identify predominant failure modes, causes and mechanisms. Time-dependent failure frequencies for major components were calculated to identify aging trends. Plant specific information was also reviewed to supplement data base results.

A computer program (PRAAGE-1988) was developed and implemented to model a typical RHR design and perform time-dependent Probabilistic Risk Assessment (PRA) calculations. Time-dependent failure probabilities were input to the PRAAGE program to evaluate the effects of aging on component importance and system unavailability.

SUMMARY

As part of ongoing efforts to understand and manage the effects of aging in nuclear power plants, an aging assessment of a vital system, the Residual Heat Removal (RHR) system in Boiling Water Reactors (BWRs)was performed. This report presents the results of the assessment and discusses the impact of RHR system aging on plant safety. This work was performed under the auspices of the U.S. Nuclear Regulatory Commission (NRC) as part of the Nuclear Plant Aging Research (NPAR) program.

A system level study is more complex than a component level study, however, it has several advantages. The system level study can assess the effects of individual components within the system on overall system performance, including component interactions. A system level study can also address the effects of design redundancies within the system along with interfaces with other systems. The NPAR system level studies are, therefore, a necessary supplement to the component level studies to provide a more complete understanding of aging.

The RHR study was performed according to the methodology developed by BNL as part of the Aging and Life Extension Assessment Program (ALEAP) System Level Plan. The approach used involves the use of two parallel work paths, with one path applying deterministic techniques and the second path applying probabilistic techniques to characterize aging.

The findings from this study have formed a technical basis for understanding the effects of aging in RHR systems. In addition, the following specific conclusions were made:

Aging Effects

- Aging has a moderate impact on RHR component failure rates (0 to 17% per year increase) and system unavailability (factor of 2 to 4 increase in 50 years). This mitigation of aging effects may be attributed to two factors, 1) RHR is a safety system and has relatively stringent testing and monitoring requirements which identify aging degradation before performance is adversely affected, and 2) the RHR system is typically maintained in standby which minimizes exposure to wear related degradation.

- Preliminary comparisons of unavailability for standby and continuously operating systems has shown that standby systems are potentially less severely affected by aging. Using this work as a basis, the differences in operation and management of these two types of systems will be further evaluated with the ultimate goal of developing methods that are effective in mitigating aging effects.

- Examination of plant specific failure data has confirmed that plants can have failure trends for certain components which differ from industry averages. Although aging was found to have a moderate impact on the RHR system based on average values, the impact on plants which differ from these average values could be significant. This will be addressed in future work.

Data Analysis

- Results have confirmed that generic failure rates may not accurately represent individual plants for all applications. The uncertainty in risk estimates may be reduced by updating calculations with actual plant data.

- Mechanical components in the RHR system show a low to moderate increase (8% to 17% per year) in failure rate with age, while electrical components, such as switches and sensors, show little or no increase (0 to 3% per year).

Design Considerations

- Plants with a common suction line supplying all loops of RHR while in the shutdown cooling mode should consider placing increased attention on MOVs in the suction line during later years of plant life since aging can increase the probability of MOV failure and lead to a temporary loss of shutdown cooling capability. Piping and other components in non-redundnt supply lines should also be considered.

- Plants using a common minimum flow line for two RHR pumps should closely monitor pump performance since aging can degrade performance and lead to dead headed pump operation and possible failure.

The predominant RHR system aging characteristics identified in this study are discussed in the following paragraphs.

The deterministic work performed for the RHR system study involved the review of past operating data from various national data bases. The data covered all operating modes of RHR. They showed that approximately 70% of the failures reported were due to aging. The dominant cause of failure was found to be "normal service," while the dominant failure mechanisms were "wear" and "calibration drift." The predominant failure mode was "leakage" followed by "loss of function" and "wrong signal."

The data also indicated that approximately 65% of the failures were detected by current test and inspection practices. However, 27% of the failures were not detected until an operational abnormality occurred. This shows that current maintenance and monitoring practices are not completely successful in detecting all aging degradation.

In evaluating the effect of failure on RHR performance it was found that over 50% resulted in degraded system operation, while approximately 20% resulted in a loss of redundancy. Other significant effects of RHR failures include loss of shutdown cooling capability, radiological releases, reactor scrams and actuation of engineered safety features.

To supplement and validate the data base findings, actual plant records were obtained and reviewed for Millstone Unit 1. Results showed that MOVs and instrumentation/controls were the components most frequently failed, which is consistent with data base findings. In addition, a large aging fraction and similar aging characteristics were found.

The probabilistic work entailed the implementation of a PC based computer program (PRAAGE-1988) to perform time-dependent PRA calculations. The RHR model used was based on the Peach Bottom design. Time-dependent failure rates were developed from the data base findings and input to the program to calculate system unavailability and component importances for various ages. The probabilistic work addressed the LPCI and SDC modes of RHR since they are the most commonly aligned and are important to plant safety.

Results from the probabilistic work showed that when the time-dependent effects of aging are accounted for, two significant system effects are seen: 1) system unavailability increases moderately with age, and 2) component relative importances may change with age. For LPCI operation, miscalibration of instrumentation was the most important contributor to system unavailability. However, during later years aging can cause MOVs to become equally important. PRA calculations for SDC operation showed MOVs to be the most important contributor to unavailability throughout plant life.

The findings presented in this report form a sound technical basis for understanding and managing the effects of aging in RHR systems. The results also provide the framework for future phase II work to be performed. Although the time-dependent aging effects appear to be mitigated to some extent for this standby system, additional work is necessary to complete the aging assessment. Since RHR is predominantly a standby system, exposure to operating stresses is limited which could contribute to the mitigation of aging effects. However, as plants continue to age and operating time increases, the RHR system could experience rapid increases in failure rates, as was found in previous work on a continuously operating system. This should be addressed in future work. In addition, the relatively stringent tests and inspections performed for the RHR system may contribute to the moderate aging effects. Future work should, therefore, be performed to determine if the practices which detect and mitigate aging degradation in the RHR system can be identified and adapted for use in other systems.

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ACKNOWLEDGEMENTS

The authors wish to thank the NRC Program Manager, Mr. Satish K. Aggarwal for his comments and guidance in the performance of this work. We would also like to express our appreciation to Northeast Nuclear Energy Company of Connecticut for the valuable RHR maintenance records they provided for this study.

We would like to express our gratitude to various members of the Engineering Technology Division of BNL, including Mr. Robert E. Hall, Dr. John Boccio, Dr. Pranab Samanta, Dr. Neil Oden, Mr. William Luckas, Mr. John Weeks, and Mr. James Higgins for providing technical assistance, guidance, and the review of this report.

We also wish to thank Ms. Ann Fort for her help in the preparation of this manuscript.

ACRONYMS

AFI	Aging Fractional Increase
ALEAP	Aging and Life Extension Assessment Program
AOV	Air Operated Valve
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CCW	Component Cooling Water
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
FI	Functional Indicator
FMEA	Failure Modes and Effects Analysis
FSAR	Final Safety Analysis Report
HPSI	High Pressure Safety Injection
HX	Heat Exchanger
IGSCC	Inter-Granular Stress Corrosion Cracking
INPO	Institute of Nuclear Power Operations
IPRDS	In Plant Reliability Data System
LER	Licensee Event Report
LPSI	Low Pressure Safety Injection
LPCI	Low Pressure Coolant Injection
MIC	Microbiologically Induced Corrosion
MOV	Motor Operated Valve
NII	Normalized Inspection Importance
NPAR	Nuclear Plant Aging Research
NPE	Nuclear Power Experience
NPP	Nuclear Power Plant
NPRDS	Nuclear Plant Reliability Data System
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RHR	Residual Heat Removal
SDC	Shutdown Cooling
SW	Service Water
TDH	Total Developed Head

1. INTRODUCTION

1.1 Background

In recent years aging of nuclear power plants has become an increasing concern to the nuclear community. As plants approach their design life, questions regarding current plant safety, as compared to when they were first built are being raised. To provide answers to these questions, the NRC Office of Nuclear Regulatory Research, Division of Engineering has an ongoing research program for assessing aging effects on equipment and systems in nuclear power plants. The program, entitled "Nuclear Plant Aging Research" (NPAR), seeks to improve the operational readiness of plant systems and components that are vital to nuclear power plant operation and safety by understanding and managing aging degradation. Work under the NPAR program began by evaluating the aging effects on selected plant components.[1-8] Current NPAR studies are focussing on the effects of aging on entire plant systems. A detailed description of the NPAR program is provided in NUREG-1144.[9]

A nuclear power plant system is comprised of various mechanical and electrical components which perform a specific function in the day-to-day operation of the plant. The components may be located in various buildings throughout the plant and be interconnected by pipes and electrical cables. Numerous interfaces between different plant systems exist, which means that a malfunction in one system can have an adverse impact on the performance of another system.

Evaluating the aging effects on a system is complicated by several factors including 1) the various components and subcomponents within the system can degrade at different rates, and 2) there is dynamic interaction of components within the system which could mask certain aging effects. In addition, there are a variety of aging factors which must be considered, including normal wear, plant transients, environmental stresses and human factors.

Although it is more complex, a system level aging assessment has several advantages over a component level evaluation. The effect of individual components within a system on overall system performance can be assessed including the effects of component interactions. Design redundancies and interfaces with other systems and components can be addressed to allow more objective decisions to be made concerning their importance. In addition, test, maintenance and surveillance priorities can be developed or modified as the system ages.

This study addresses the impact of aging degradation on Residual Heat Removal (RHR) systems in boiling water reactors (BWRs). The RHR system was chosen since it is a safety system which performs several functions and plays a vital role in the safe operation of the plant. For some plant designs, studies have found the RHR system to be one of the most important systems for preventing core melt.[10-12] This work is based primarily on the review of existing plant RHR designs, and operating experience from plant specific data and generic data bases.

1.2 Objectives

In accordance with the NRC-NPAR Program Plan[9], the primary goals of this phase I RHR system study are to identify and characterize aging and service wear effects which, if unchecked, could cause degradation of structures, components, and systems and thereby impair plant safety. This will include an investigation of the predominant failure mechanisms, modes and causes, along with the effect of aging on system performance and time-dependent failure rates. In addition, a preliminary review of current inspection, surveillance and maintenance practices will be performed to identify areas where improvements can be made to more effectively detect and mitigate aging degradation.

To achieve these goals, two preliminary tasks were first completed: 1) the system to be studied was defined and its interfaces were identified, and 2) a methodology for performing the system analysis in a structured manner was developed. These items are discussed in the following sections.

1.3 System Definition

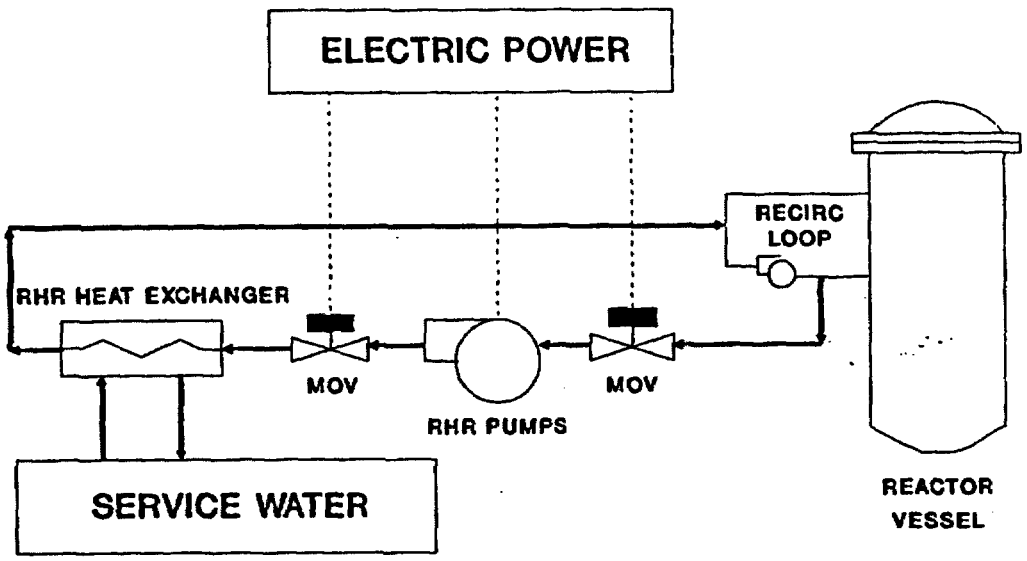
1.3.1 Description of RHR System

For purposes of this study, the RHR system is defined to be that group of components, including pumps, valves, heat exchangers, interconnecting piping, pipe supports/restraints, and instrumentation/controls, whose functions are 1) to provide low pressure coolant injection (LPCI) into the reactor following a loss of coolant accident, 2) to provide shutdown cooling (SDC) capabilities during reactor shutdown, 3) to provide containment spray cooling, 4) to provide suppression pool cooling, 5) to provide steam condensing when the main condenser is not available, and 6) to provide fuel pool cooling capability. Typically, RHR systems in most plants can perform all these functions, however, LPCI and SDC are the most frequently aligned modes for RHR (Figure 1.1). The data analysis portion of this study (Section 5) evaluates failure data from all operating modes of the RHR system. The probabilistic work performed for this study (Sections 7 and 8) addresses only the LPCI and SDC modes of the RHR system. The scope of the probabilistic work was limited to these two modes to make the study more manageable. LPCI and SDC were chosen since they are the most commonly aligned and are important to plant safety. A more detailed description of RHR system design and operation is presented in Section 2.

1.3.2 System Interfaces and Boundaries

The RHR system interfaces with several other systems in the plant. To provide a clearly defined system for analysis, RHR boundaries were established. Failures which occur outside the RHR boundaries are not included in this study. The interfacing systems and boundaries are discussed in detail in Section 2.3.

SHUTDOWN COOLING (SDC) MODE



LOW PRESSURE COOLANT INJECTION (LPCI) MODE

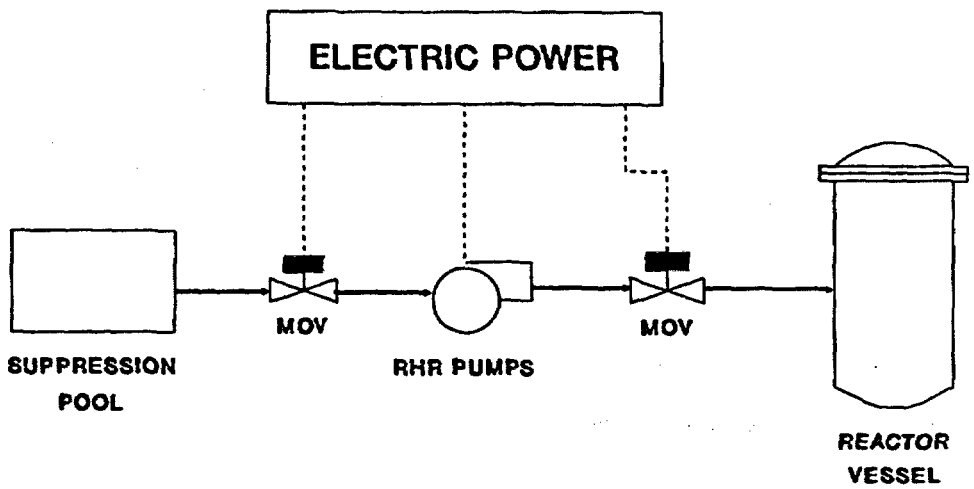


Figure 1.1 Functional Diagram of RHR System

1.4 Analysis Methodology

Recognizing that the characterization of aging in a nuclear power plant system is a complex task, a system level program plan was developed, entitled "Aging and Life Extension Assessment Program (ALEAP)."[13] This plan presents a structured strategy for assessing the aging effects on nuclear power systems during the normal 40 year life and perhaps for extension of plant operation beyond the original license.

The ALEAP plan is consistent with the NPAR program plan and has two phases. The first phase focuses on characterizing the aging effects on the system in terms of the predominant modes and mechanisms of failure, as well as their impact on system performance. Also included in Phase I is a preliminary review of current test, maintenance, and inspection practices. The second phase of the work stresses the assessment of monitoring and maintenance practices and the development of techniques to mitigate aging effects. The specific tasks to be performed in each phase are outlined in Figure 1.2. This report includes the Phase I work. Figure 1.3 presents the overall strategy employed in this system study. This involves a two-pronged approach which assesses aging impact on system performance through both deterministic and probabilistic techniques. The deterministic approach included a review of the various RHR system designs in use. The scope of the design review encompassed all operating BWR plants in the United States.

In addition to the system design review, a detailed analysis was performed of the various failure data bases summarizing the actual operating experience of RHR systems. These data bases include:

- Nuclear Plant Reliability Data System (NPRDS),
- Licensee Event Reports (LER),
- Plant Specific Failure Data
- Nuclear Power Experience (NPE)
- Surveys of 9 BWR Plants

Each data base was analyzed to determine the predominant failure modes, causes, and mechanisms contributing to system failure. The operational stresses and other parameters contributing to the aging of components were considered in assessing their functional characteristics. Other relevant factors such as failure rates, aging fractions, and time to failure were extracted for use in the probabilistic models for predicting the relative importance of particular components and system unavailability as a function of age.

Plant specific data for the system was obtained to supplement the generic data bases. The plant data included maintenance records from an operating BWR which covered a four year period. The data represented plant ages 14 through 18, with ages 14 and 18 only partially accounted for. An analysis, similar to that used for the data base records, was performed on the plant data to identify aging characteristics. This included a determination of aging fraction, failure causes and failure effects, as well as identification of the components failing most frequently. These findings were then compared with those from other data bases as a check on the results.

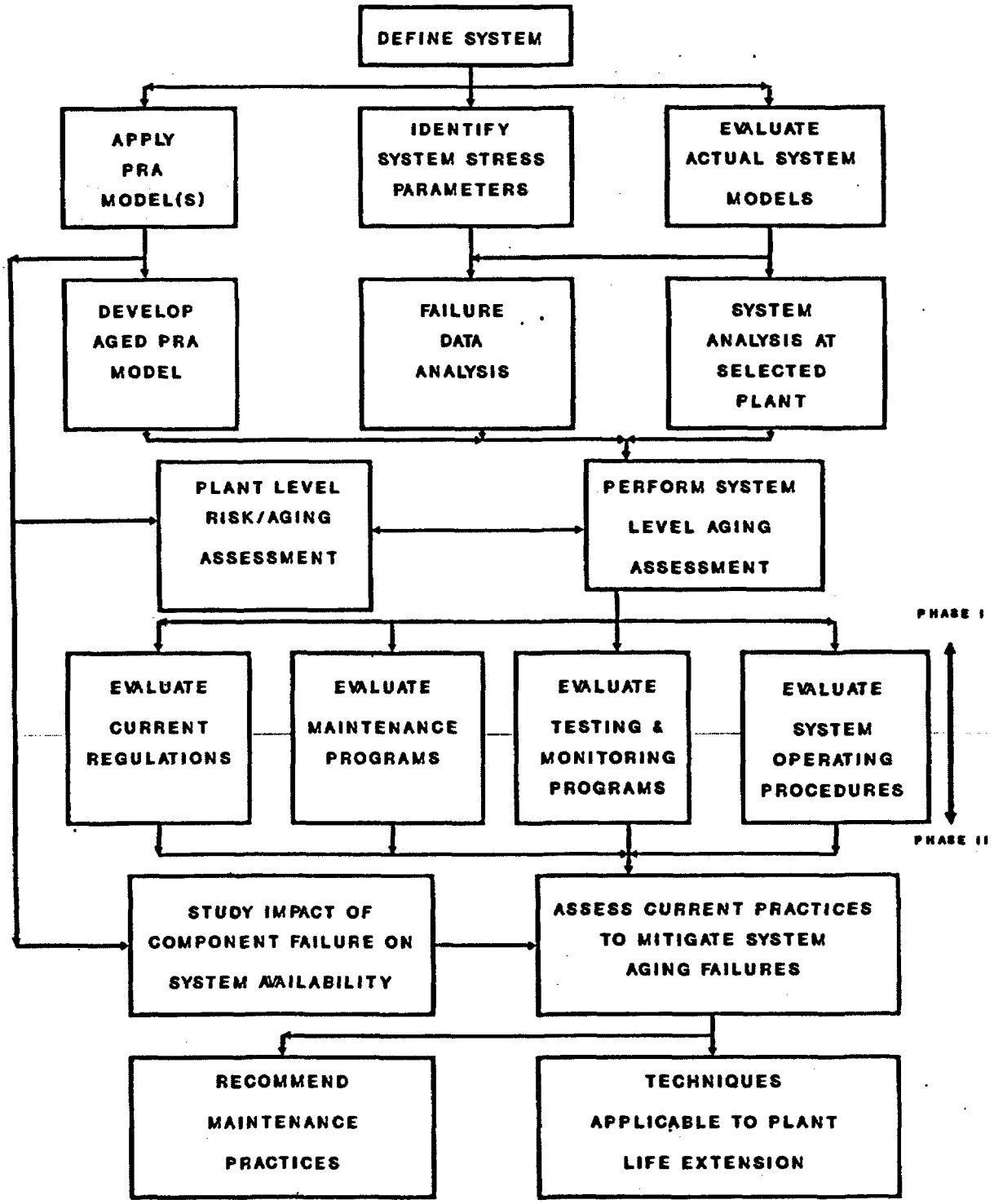


Figure 1.2 Detailed Task Structure of ALEAP System Level Plan

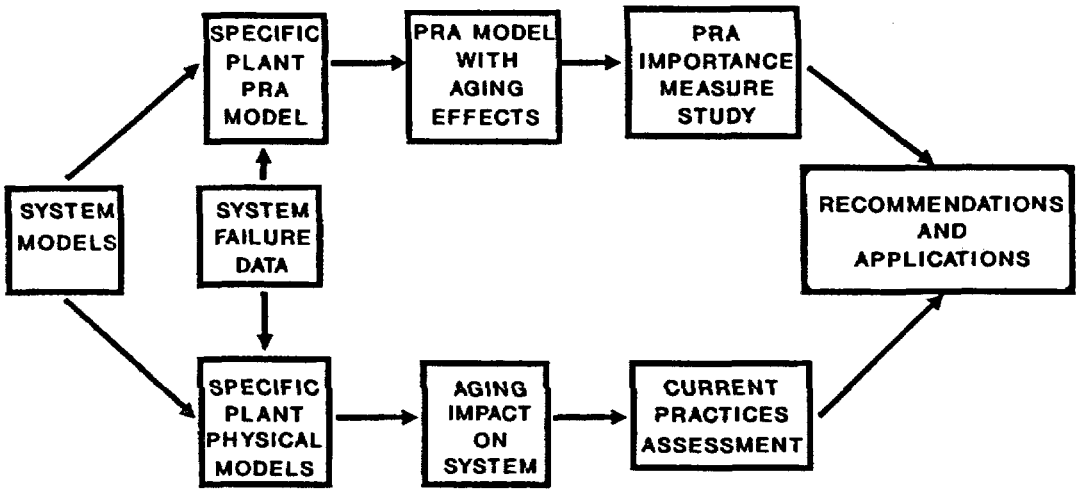


Figure 1.3 Overall Strategy for RHR System Study

In parallel with the deterministic effort, a probabilistic approach on a specific plant probabilistic risk assessment (PRA) model was performed to study the impact of aging on the system availability. This assessment determined the components which have the dominant effect on system availability. Because of the complexity of the plant and system, it was not feasible to perform aging analyses for all components and subcomponents. Therefore, those components that are vulnerable to degradation with age and important to system operation were analyzed.

A plant with a completed PRA was chosen for the analysis. A PRA model and a computer program (PRAAGE-1988) were developed to reflect the essential features of the RHR system design and to accommodate age-related failure rates. The time-dependency of the aging phenomena was modeled to assign priorities to the possible component failures with the age of the plant.

Section 2 of this report describes the design review of the RHR system for all BWRs in the United States. The operational stresses and their correlation with accidents are discussed in Section 3. Section 4 presents a review of current utility practices related to testing and maintenance of RHR systems. Section 5 provides the results of all data base analyses and identifies the predominant RHR system failures from the operating experience at nuclear plants. The detailed review of the RHR system at the Millstone Nuclear Power Station,

Unit 1 (Millstone 1) is summarized in Section 6. Section 7 discusses the PRA model of the RHR system at Peach Bottom Atomic Power Station, Unit 2 and applies the statistical data taken from the RHR system operating experience to rank the importance of components within the system. Section 8 discusses the sensitivity studies, while the results and conclusions of this work are summarized in Sections 9 and 10. Several appendices give detailed information on the specific areas discussed.

2. RHR SYSTEM DESIGN REVIEW

The Residual Heat Removal (RHR) System in a Boiling Water Reactor (BWR) serves a variety of purposes for operation during routine, abnormal, and emergency conditions. The RHR system can be operated in several different modes. Each of the modes has its own design objectives, however, many of the physical components of the RHR system (pumps, piping, valves, instruments) are used in more than one of the operating modes. The major operating modes of the RHR system are:

Low Pressure Coolant Injection (LPCI): An operating mode of RHR used during and following a loss of coolant accident (LOCA) to provide low pressure makeup water to the reactor vessel for core cooling.

Containment Spray Cooling: An operating mode of RHR designed to be used following a LOCA to reduce the primary containment pressure (drywell and suppression pool) by condensing any steam which may be present.

Suppression Pool Cooling: An operating mode of RHR designed to cool the volume of water in the suppression pool and maintain its temperature within allowable limits during normal operation and post accident conditions.

Shutdown Cooling (SDC): An operating mode of RHR used to complete a Reactor System cooldown and maintain the reactor in a cold condition such that the reactor can be vented, refueled, and serviced.

Steam Condensing: An operating mode of RHR used to condense live reactor steam and maintain the reactor in a standby condition when the main condenser is not available. This mode is rarely used.

Fuel Pool Cooling: An operating mode of RHR designed to augment or operate in place of the fuel pool cooling heat exchangers to increase the heat removal capacity of the Fuel Pool Cooling and Cleanup System.

This section summarizes the designs and configuration differences that exist among plants achieved primarily through a review of Final Safety Analysis Report (FSAR) information. The LPCI and SDC modes will be discussed in detail since these are the two modes addressed in the probabilistic analysis for this study.

BWR designs have evolved over time and consequently there have been changes in the RHR system. The basic components have remained the same, however, capacities and configurations have been modified.

Table 2.1 illustrates the six groupings of RHR system designs. As shown, the most common configuration is the 2 loop system, with 2 pumps per loop and 1 or 2 heat exchangers per loop. A number of the more recent plants, including the BWR 6 designs, have 3 LPCI loops with a total of 3 pumps, and 2 heat exchangers. On the other hand, several older plants have either separate LPCI and SDC loops, or no LPCI system at all, as in the cases of Nine Mile Point 1 and Oyster Creek. Figure 2.1 shows the most common RHR system arrangement.

Table 2.1 Grouping of Plants by RHR System Characteristics

GROUP	PLANTS	RHR SYSTEM CHARACTERISTICS
1	Arnold Brunswick 1 & 2 Cooper Fermi 2 Fitzpatrick Hatch 1 & 2 Monticello Pilgrim Quad Cities 1 & 2 Shoreham Susquehanna 1 & 2	Loops: 2 Pumps: 2 Per Loop HX's: 1 Per Loop
2	Browns Ferry 1, 2 & 3 Peach Bottom 2 & 3	Loops: 2 Pumps: 2 Per Loop HX's: 2 Per Loop
3	Hope Creek Limerick HX's: 2 Total	Loops: 4 Pumps: 4 Total
4	Clinton Grand Gulf 1 & 2 La Salle 1 & 2 Nine Mile Point 2 Perry River Bend WNP2	Loops: 3 Pumps: 3 Total HX's: 2 Total
5	Dresden 2 & 3 Millstone 1	Separate LPCI/SDC Systems SDC Loops: 2 LPCI Loops: 2 SDC Pumps: 2 LPCI Pumps: 4 SDC HX's: 2 LPCI HX's: 2*
6	Nine Mile Point 1 Oyster Creek	No LPCI System SDC Loops: 3 SDC Pumps: 3 SDC HX's: 3

* LPCI HX's used for containment cooling only.

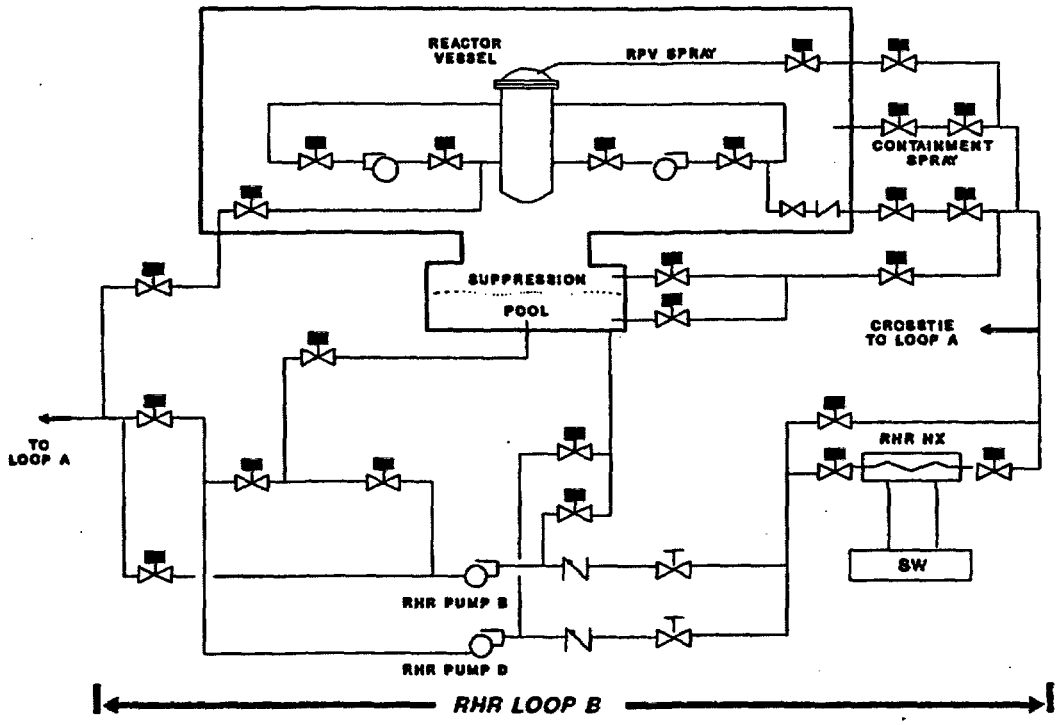


Figure 2.1 Typical RHR System Design for One of Two Loops

2.1 LPCI Design Basis

The RHR system is aligned for LPCI operation whenever the reactor is at power. The function of the LPCI system is to provide a makeup coolant source to the reactor vessel during accidents in which reactor system pressure is low. In most plants, LPCI is an integral part of the RHR system. It is designed to actuate automatically, in combination with other emergency core cooling systems, to restore and maintain the coolant inventory in the reactor vessel such that the core is adequately cooled.

LPCI is a low head, high flow operating mode of RHR that delivers rated flow to the reactor vessel when the differential pressure between the vessel and the primary containment is sufficiently low (typically 20 psi or less). During LPCI operation, the RHR pumps take suction from the suppression pool and discharge into the core region of the reactor vessel through the recirculation loops or to vessel nozzles located above the core in the downcomer region. Any spillage through a break in a line within the primary containment returns to the suppression pool. This is particularly important when considering the design basis large break accident associated with the recirculation piping.

The major components associated with the LPCI operating mode are the pumps, valves, piping, and instrumentation and controls. The following paragraphs describe some of the design parameters associated with each.

2.1.1 RHR Pumps Used in LPCI Mode

LPCI pump flow rates range from approximately 7000 to 10,000 gpm, with the most common flow rate found to be 7700 gpm. The total developed head is approximately 300 psid at a low flow condition.

The RHR pumps are motor driven, vertical can-type pumps with mechanical seals and cyclone separators. For the LPCI mode, the pump suction is from the suppression pool. Strainers are frequently employed in this suction path to prevent pump damage. The strainers are designed to have minimal impact on pump flow; for example, at Hope Creek, no more than 1 foot of head loss at rated flow is expected with 50% of the total strainer area plugged.

The pumps are driven by 4160V ac, 3 phase motors and are capable of achieving rated speed within 20 to 30 seconds. The pump housing is generally constructed of carbon steel with a stainless steel pump shaft and impellers.

To prevent pump damage due to overheating at no flow, the control circuitry prevents a pump from starting unless a suction path is lined up. Limit switches may be used to trip the pump motor breaker if the suction valve closes. Similarly, the RHR pumps have a minimum flow or bypass line which routes water from the pump discharge to the suppression chamber to prevent pumps from overheating. The minimum flow valve opens if low flow is sensed and closes automatically once the low flow setting is exceeded.

2.1.2 RHR Valves Used in LPCI Mode

The LPCI flow path from the suppression pool to the vessel contains a number of key valves which must operate properly to assure system success. These valves are illustrated in Figure 2.2 and include:

- Pump suction valve (open) (#5)
- Minimum flow valve (open then close when minimum flow is established) (#4)
- Pump discharge check valve (open) (#6)
- Full flow test valve (close) (#3)
- Heat exchanger bypass valve (open) (#2)
- Injection valve(s) (open on accident signal) (#1)
- Inside containment check valve (open) (#7)
- Inside containment manual block valve (open) (#8)

The function of these valves are briefly described below:

1. Injection valve(s): In most plants, two valves are employed in series in the injection path. Both are typically outside of primary containment with the inboard valve normally closed and the outboard valve normally open. Upon receipt of a LPCI initiation signal, both injection valves receive signals to open. However, the inboard valve does not actually open until a low reactor pressure permissive signal is received. As long as the accident signal is present, the inboard injection valves cannot be closed. The outboard injection valves can be closed, but only after a time delay.

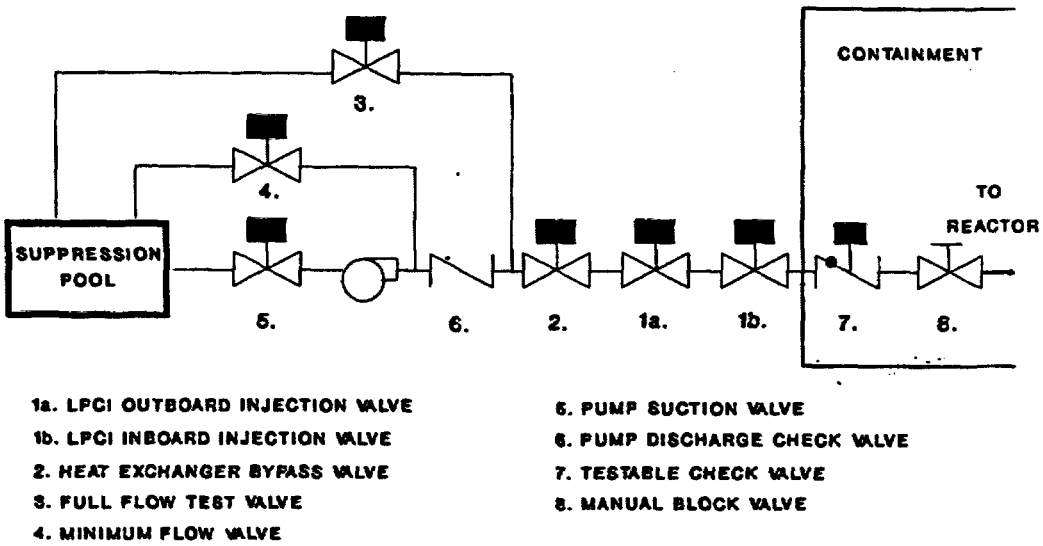


Figure 2.2 Key Valves in the LPCI System

2. Heat Exchanger Bypass Valve: On a LPCI initiation signal, this valve, which bypasses flow around the RHR heat exchanger, receives an OPEN signal. An interlock is also provided to assure that this valve remains open for several minutes. This bypass action increases the flow to the vessel by eliminating the pressure drop incurred due to the heat exchanger.

3. Pump Discharge Check Valve: Common to centrifugal pumps, a discharge check valve is provided. Typically, this valve permits the piping down stream of it to be filled with water to minimize the risk of water hammer and to assure rapid water injection. In addition, this valve prevents backflow through an idle parallel pump when two RHR LPCI mode pumps share a common discharge path and one does not operate.

4. Full Flow Test Valve: The test return lines to the suppression pool permit full flow testing of each pump. These valves will automatically close (if open for test) on a LPCI initiation signal and cannot be reopened while an initiation signal is present.

5. Testable Check Valve: Each of the two LPCI injection lines typically has a testable check valve between the two motor operated injection valves and the manual block valve. This is particularly true for those plants which inject directly to the vessel. These check valves are capable of being tested from the control room to ensure they will operate freely. A bypass line is installed around each check valve to equalize the pressure across the valve and thereby reduce the force necessary to move the check valve disc.

The pump minimum flow valve and pump suction valve were previously discussed as part of the pump protection scheme.

2.1.3 LPCI Actuation

Automatic activation of the LPCI system is accomplished through either of the following signals.

- high primary containment pressure
- low reactor water level

Once activated, LPCI injection will begin when a low reactor pressure permissive signal is obtained.

These signals may also initiate operation of other equipment such as the Emergency Diesel Generators (EDGs) and Core Spray, for example. Time delay relays may be employed to start the LPCI pumps to minimize electrical load surges.

2.2 Shutdown Cooling Design Basis

The shutdown cooling (SDC) mode of RHR uses most of the same pumps, valves, and piping described in the LPCI mode. In addition, heat exchangers and controls are required to remove residual heat from the reactor during normal shutdown operations.

During a controlled shutdown, reactor cooldown is accomplished initially by condensing reactor steam using the main condenser as the heat sink. The shutdown cooling mode is manually initiated when the reactor pressure reaches approximately 100 psig. Generally, only one heat exchanger is required for shutdown cooling, but both may be used depending on the cooldown rate required.

As illustrated in Figure 2.3, the shutdown cooling mode flow path consists of a suction source from the recirculation loop suction line through two containment isolation valves, and is then pumped through an RHR heat exchanger by RHR pumps back to the recirculation system. Except for those plants which have LPCI injection nozzles, the return path for shutdown cooling is the same as the LPCI injection path.

RHR system water passes through the shell side of the RHR heat exchanger, while service water is pumped through the heat exchanger tubes. Automatic controls are in place to maintain the tube side pressure above the shell side pressure so that any internal leakage will be from the service water side into the RHR system to prevent reactor water from being discharged to the environment. Conductivity monitoring instrumentation on the RHR side detects leakage from the service water system.

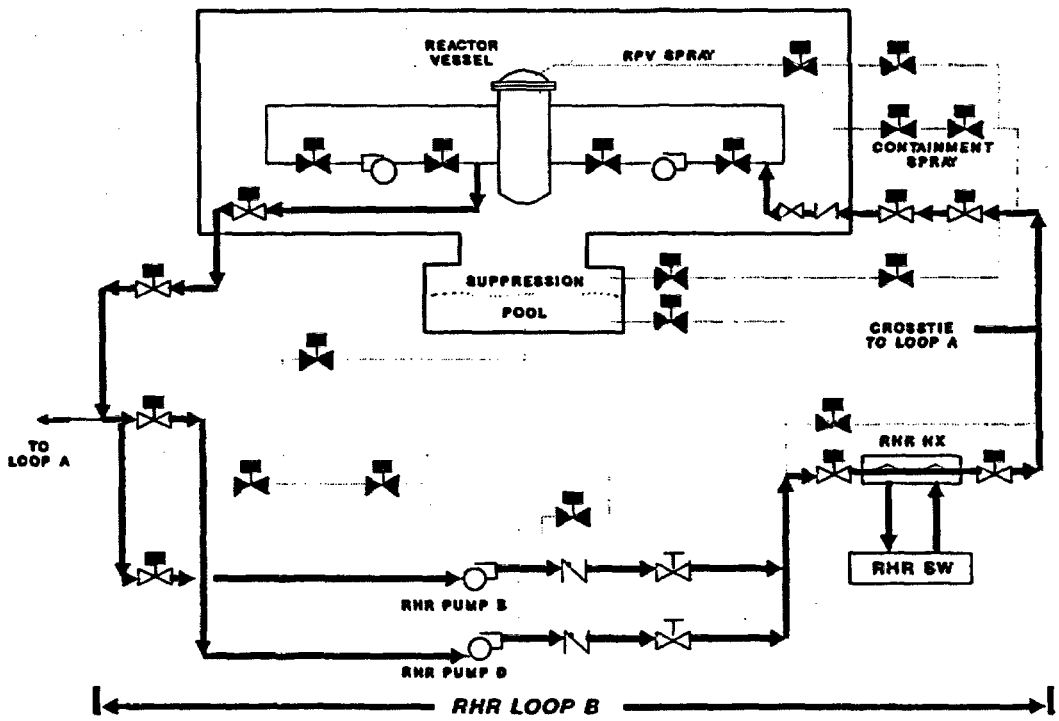


Figure 2.3 RHR Shutdown Cooling Mode Flow Path for One of Two Loops

Technical specifications permit a maximum cooldown rate of 100°F/hour to minimize the thermal transient on the reactor vessel and its internals. The operator manually controls the cooldown by operation of the bypass valve around the heat exchanger, and monitoring reactor coolant temperature instrumentation. In this manner, the reactor can be cooled to 100°F within 24 hours after the reactor is shutdown. A portion of the shutdown cooling flow can be diverted to the head spray nozzle to condense any steam generated by the vessel metal.

In the event of a low reactor vessel level, the shutdown cooling system will automatically isolate. The RHR pump suction valve from the suppression pool must then be manually opened to align the system for LPCI injection.

The design basis for the most limiting single failure for the shutdown cooling mode is to only have one loop available. Cooldown to 125°F can still be accomplished within 24 hours. The design basis for reliability and operability is that this mode of operation is controlled by the operator from the control room. The only operation performed outside the control room for a normal shutdown is manual operation of the local flushing water valves, which is a means of providing clean water to the system prior to commencing the shutdown cooling mode and also to minimize thermal stresses on the pipes.

With the exception of the shutdown suction and return, and head spray line, the shutdown cooling system is part of the Emergency Core Cooling System (ECCS) function, and is, therefore, required to be designed with the redundancy, flooding protection, pipe whip protection, and power separation required of such systems. Shutdown cooling suction and discharge valves are required to be powered from both normal and emergency power for purposes of isolation and shutdown following a loss of offsite power. In the event either of the two SDC supply valves fail to operate, an operator may be sent to open the valve by hand, if conditions permit entry to the area.

The major components associated with the SDC system which have not been discussed in the LPCI section are the heat exchangers, the suction valves, and their associated instrumentation and controls.

The RHR system heat exchangers are typically sized on the basis of the duty for the SDC system. Typical design data for an RHR heat exchanger includes:

Shell Side Flow:	7000-10,000 gpm
Tube Side Flow (Service Water):	8000-9000 gpm
Design Pressure:	450 psig
Design Temperature:	450°F
Pressure Drop at Design Conditions:	
	Shell Side: 12 psi
	Tube Side: 10 psi
Vessel Material	Carbon Steel
Tubes:	Stainless Steel 304L or Copper Nickel
Rated Inlet Temperatures:	125°F Shell Side 90°F Tube Side

The suction valves for the shutdown cooling mode of RHR have a number of interlocks. The inboard and outboard SDC isolation valves cannot be opened, and will close if opened when the reactor pressure is greater than approximately 100 psig, or the reactor low level logic is actuated. Control of valve operation from a remote shutdown panel is also typically available. A failure in this logic circuit can result in a reactor vessel drain path or a loss of isolation protection of a high/low pressure interface.

2.3 LPCI and SDC Support Systems

A number of support systems are required for the proper operation of the LPCI and Shutdown Cooling Systems. These are briefly described in this section and illustrated in Figure 2.4. The boundaries used for this study are also discussed.

2.3.1 AC Power

Power must be available to the 4160 volt safety-related electrical buses for operation of the RHR pump motors. The source of this power is either from offsite power or from the emergency diesel generators.

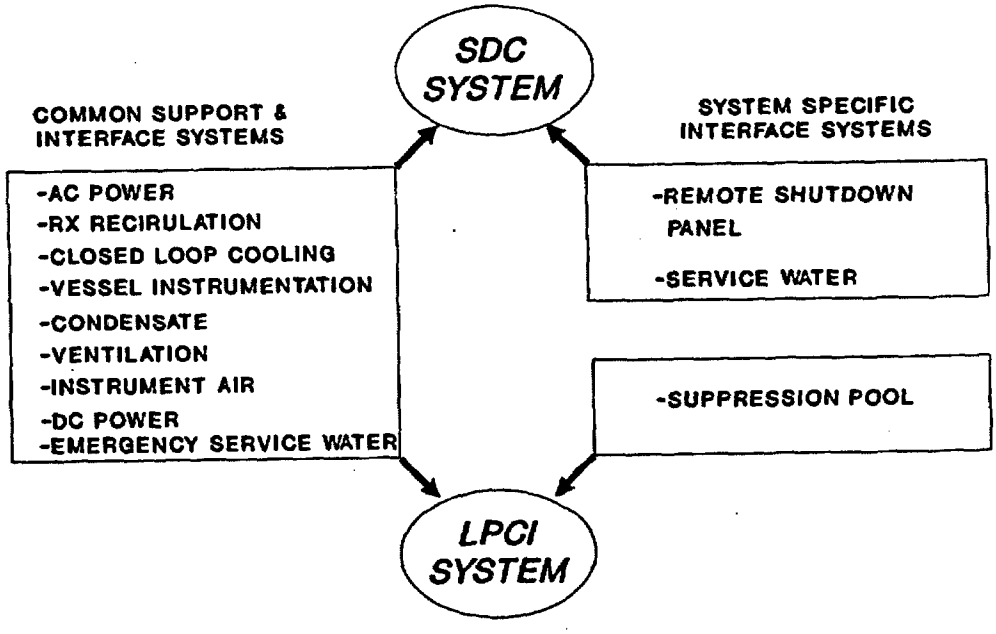


Figure 2.4 RHR Interface and Support Systems

Various motor operated valves in the RHR system require this power in two ways. First, the 480V ac power is used directly by the valve motors for operation. Secondly, the 480V ac power for each valve is transformed down to 120V ac for use in the valve's motor control circuits. Any solenoid operated valves in the RHR system would also require 120V ac for operation.

The boundary for ac power will be at the circuit breaker, and will include the breaker and breaker logic.

2.3.2 Closed Loop Cooling Water

A cooling water system provides cooling of the pump seals and motor bearings for each RHR pump. Typically, small heat exchangers at the pump/motor are used for these purposes.

The boundary will be at the first normally closed block valve in the cooling water system.

2.3.3 Reactor Instrumentation

Reactor vessel water level (low) is one of the nuclear system parameters used as an initiation signal for LPCI. Additionally, a low reactor water level will automatically close the RHR shutdown cooling isolation valves.

Primary containment pressure (high) is also an initiation signal for LPCI since it is used to detect a breech of the reactor coolant system inside the containment structure.

Reactor vessel pressure is a permissive function and must accompany the accident signal in order for LPCI injection to occur. This protects the high and low pressure interface from being compromised. Reactor pressure (high) is also used as an automatic isolation signal for the shutdown cooling valves.

The boundary for all instrumentation will be at the first circuit breaker or fuse from the equipment, and will include the breaker or fuse.

2.3.4 Condensate System

A connection from the condensate system to the RHR piping downstream of the RHR pump discharge check valve is typically provided to maintain the piping full of water. This prevents water hammer on system initiation. At some plants, a small pump takes a suction from the suppression pool to perform this same function. Operators monitor RHR discharge pressure to verify that the "keep-fill" subsystem is maintaining the piping filled.

The boundary will be at the first normally closed condensate system block valve leading to the RHR system.

2.3.5 Ventilation System

At a large number of BWRs, the RHR pumps are located in enclosures which are maintained at design temperatures by room coolers and area heaters. The coolers are part of the reactor building ventilation system with cooling provided by a chilled water system.

The entire ventilation system will be considered as outside the RHR boundary.

2.3.6 Instrument Air

Each LPCI loop testable check valve and check valve bypass valve may be operated for testing purposes by a solenoid valve and air operated valve arrangement. The air operated valves use instrument air as their motive force.

The boundary will be at the first air system valve leading to the RHR operator or other component.

2.3.7 DC Power

The relay and control power for the LPCI initiation system is derived from two independent 125V dc power systems. The logic of the LPCI mode and the normal start logic is of the "energize to operate" type and, therefore, requires that this power be available. The 125V dc power also provides the necessary control power for operation of RHR pump motor circuit breakers.

The boundary will be at the first circuit breaker or fuse from the equipment supplied.

2.3.8 Service Water (SW)

A service water supply to the RHR system is required to support the shutdown cooling water system. It is supplied to both RHR heat exchangers for heat removal.

The boundary will be at the point where SW enters the RHR heat exchangers. Failures, or plugging of the tubes within the heat exchanger will be included, but failures of the SW pipe or valves outside the heat exchanger will not. Any failure outside of the heat exchanger will be treated as a "Loss of Service Water."

2.3.9 Remote Shutdown Panel

Because of the regulation that plants be capable of shutdown from outside the control room, controls and instrumentation associated with the shutdown cooling system are located in a remote or auxiliary shutdown panel. Typically, these controls include pump and valve control switches and vessel temperature and pressure indication.

The remote shutdown panel will be considered outside the RHR system boundaries for this study.

2.3.10 Suppression Pool

The suppression pool provides the source of water for the LPCI mode. In addition, RHR system connections to the suppression pool are associated with the full flow test line and the minimum flow piping.

The boundary will be at the interface between RHR piping and the suppression pool.

2.3.11 Structures and Buildings

The RHR system usually runs through a large portion of the plant, including the Auxiliary Building and Primary Containment. The system is safety related and hence, typically is mounted or supported so that it will withstand seismic shocks. The structures and buildings to which RHR components are attached are not included, but the attaching hardware is included, such as bolts, bedplates, brackets, snubbers, and pipe supports. Hence, the boundary is just beyond the supporting hardware.

Additional details on plant specific designs can be found in Appendix A.

2.4 NRC Activities Related to RHR

The importance of the RHR System is indicated to a certain degree by the amount of NRC activity associated with it, as witnessed through the issuance of Bulletins, Notices, and AEOD studies. This section of the report summarizes the recent documents which are related to RHR system degradation due to aging.

2.4.1 Bulletin 88-04 and Notice 87-59: "Potential RHR Pump Loss"

This Information Notice describes aging related degradation which could occur due to minimum flow problems in Westinghouse plants. With two pumps sharing a common minimum flow line, for instance, one pump could be operated dead headed if its performance characteristics are sufficiently degraded compared to the other. In addition, the Notice indicates that some RHR pump manufacturers are advising that minimum flow capacities should be beyond the 5 to 15% of pump design flow most common in BWR plant designs.

This issue has safety implications because in an accident scenario, where the pumps start and must operate on minimum flow until the low pressure permissive signal is obtained, vibration or overheating could cause pump degradation or inoperability. To a lesser degree, wear can occur during regular testing, if the pump is operated in minimum flow for any appreciable time. Pumps operating in the same loop should be compared to determine if one would cause the other to be operated dead headed due to better operating characteristics. Although this notice was issued for PWR plants, it may be applicable to BWR plants also due to design similarities.

2.4.2 Bulletin 86-01: "Minimum Flow Logic Problems that Could Disable RHR Pumps"

This bulletin was issued to all BWRs, with special emphasis given to two BWRs known to possess a particular instrumentation logic associated with the minimum flow feature of the RHR pumps. In particular, a postulated single failure of a flow instrument could result in all four pumps running dead headed in an emergency scenario. This would quickly (within minutes) render the pumps inoperable due to overheating.

By administrative controls, the operator should verify the opening of the minimum flow valve when the pump is manually started for testing. In the event of misoperation of the minimum flow valve, the operator must be prepared to establish an alternative flow path or trip the pump to preclude damage. Some pump degradation due to overheating may be routinely experienced if logic response time or minimum flow valve stroke time is slower if setpoint drift of the flow instrument permits the minimum flow valve to close before a high enough flow is established.

2.4.3 Notice 87-63: "Inadequate Net Positive Suction Head (NPSH) in Low Pressure Safety Systems"

This notice addresses the concern of pump unavailability due to pump cavitation resulting from inadequate NPSH and from sudden suction pressure oscillations during pump starts. These oscillations could cause pump trips. In particular, several plants found that under specific postulated accident conditions inadequate NPSH would exist. These conditions were due to the following design and installation errors:

In a 15 year old PWR, the system hydraulic resistance was found to be less than assumed. Flow orifices in a discharge line had not been installed.

The RHR system of a 19 year old PWR was found to be deficient for a narrow range of postulated pipe break sizes. Pump cavitation which could occur in the recirculation mode was corrected by throttling the RHR system control valves to balance the flow paths and effectively increase the system hydraulic resistance while maintaining minimum flow requirements.

At a new PWR, the licensee reported that the RHR pump rate would be higher than expected during the cold leg recirculation mode of operation due to lower than expected hydraulic resistances in the RHR pump discharge piping.

While no specific reference is made to BWR RHR systems, the potential for inadequate NPSH exists due to basically the same reasons cited in this notice. For instance, at most BWRs, a cross-tie exists between the two loops of RHR. With the crosstie open, one pump supplying both loops could result in pump cavitation due to inadequate NPSH. Administrative controls are typically implemented to preclude this condition.

2.4.4 Notice 87-51: "Failure of Low Pressure Safety Injection Pump Due to Seal Problems"

When the RHR system is operated in shutdown cooling, higher pump temperatures are experienced due to the higher fluid temperature. At one plant (a PWR), a LPSI pump seal failure occurred with the unit in shutdown cooling. Follow-up analysis revealed that while the system design temperature is 400°F, the temperature rating of the "O" rings used in the pump seal was only 300°F. It is clear that operation of the system at the system design rating would result in accelerated wear and ultimate failure of the seal. The Notice also mentions that the application of a cleaning solvent or lubricant to the seal may cause it to expand and crack.

To preclude reoccurrence, the utility placed an administrative limit on the reactor coolant temperature before the pumps could be placed in service to assist in shutdown cooling. Maintenance procedures were modified to prevent the misapplication of solvents or lubricants to the seals.

Before the event, the licensee observed that pump shaft seal leakage was greater than normal. This detection mechanism, if acted upon, could have prevented the catastrophic failure.

2.4.5 Notice IN 87-50: "Potential LOCA at High and Low Pressure Interfaces from Fire Damage."

Some BWRs have a motor operated bypass valve around the injection line check valve which, under certain postulated fire conditions, could degrade the high-low pressure boundary in the RHR system. This bypass line is used to warm up the RHR system discharge line to prevent thermal shocking of the reactor vessel nozzle safe end. The fix to preclude this condition was to remove power from the MOV during operation.

2.4.6 Notice 87-23: "Loss of Decay Heat Removal During Low Reactor Coolant Level Operation"

This notice and several others discuss the potential loss of decay heat removal capability at PWRs during operation with low reactor water level. Numerous events have been reported where pump suction was lost due to reduction of reactor water level to too low a level. This led to vortexing, air entrainment and cavitation of the pumps. In the majority of the cases, the problem was attributed to incorrect, inaccurate or inadequate reactor level indication.

This problem is relevant to the aging study since aging degradation can lead to failure of level instrumentation. This could potentially cause loss of decay heat removal capability as described in the above events.

2.4.7 Notice 87-10: "Potential for Water Hammer During Restart of Residual Heat Removal Pumps"

Analysis at a BWR revealed that the automatic realignment of the RHR system from the suppression pool cooling mode to the LPCI mode in the event of an accident signal could result in a drain down of the piping. Upon pump start, the system could be subjected to a water hammer event. As an interim corrective action, only one RHR loop is permitted to be in suppression pool cooling. In addition, restart of a pump if it trips in suppression pool cooling will require operator verification that the piping is filled and vented.

2.4.8 Notice 86-96: "Heat Exchanger Fouling Can Cause Inadequate Operability of Service Water Systems"

Some utilities have identified plant heat exchangers which have the potential for fouling, affecting the plant's ability to reject heat in a post accident condition. RHR system heat exchangers are included here. One of the recommendations of this notice is to include biofouling surveillance in a routine maintenance program.

Heat exchanger fouling is an aging concern that can be mitigated through inspection and cleaning practices. This monitoring is not required by technical specifications but should be considered by utilities to insure that FSAR design margin is met over the life of the plant.

2.4.9 Notice 86-74: "Reduction of Reactor Coolant Inventory Because of Misalignment of RHR Valves"

Shutdown cooling makes use of the same piping, valves, and pumps, that the LPCI function uses. The misalignment of shutdown cooling valves with its potential for RHR system damage and draining of the reactor coolant system is of regulatory concern. As indicated in Figure 2.5, five drain paths are identified which have been established inadvertently at various plants. The Notice references a service information letter (SIL 388) issued by General Electric which states that shutdown cooling is entirely controlled by manual operator actions and is subject to errors which could result in hydraulic and thermal conditions not specifically considered in the design process.

The operating experience for the shutdown cooling system indicates that the human input to system aging is an important one because of the required manual operations. Procedures and training for operation of the shutdown cooling system are aging mitigation methods which are as necessary as hardware modifications or maintenance.

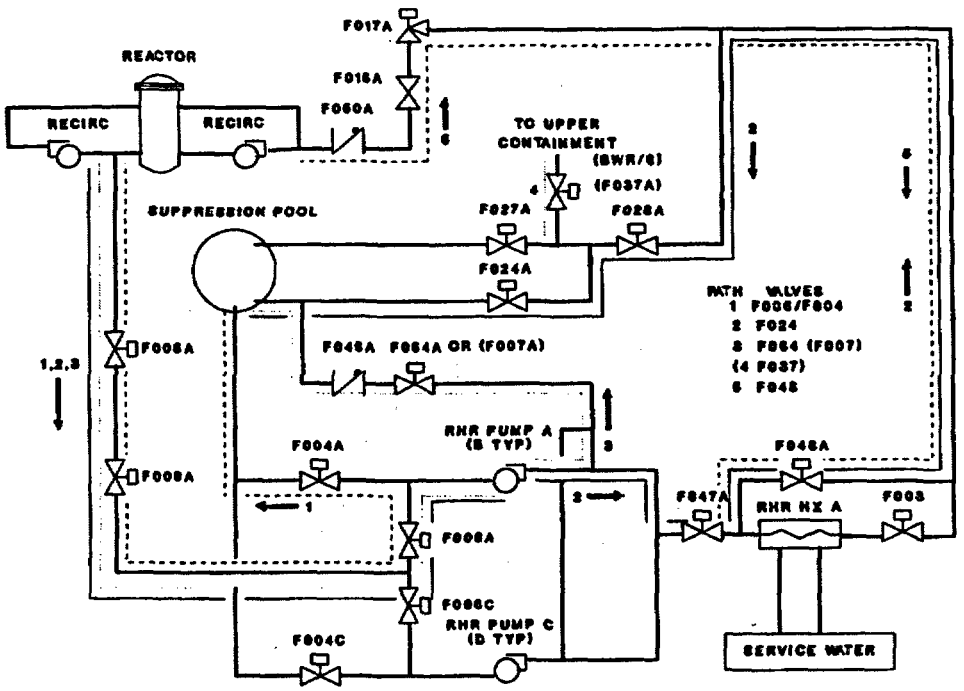


Figure 2.5 Inadvertent Drain Paths Identified in Notice 86-74

2.4.10 Notice 86-40: "Degraded Ability to Isolate the Reactor Coolant System from Low Pressure Coolant Systems in BWRs"

Leaking valves have resulted in RHR lines being exposed to primary coolant at temperatures and pressures above design. The potential flow paths are illustrated in Figure 2.6. Postulated results of this are over-pressurization with possible faulting of the low pressure line and a LOCA, steam binding of one or more of the RHR pumps, and water hammer. Generic Issue 105, "Interfacing Systems LOCA at Boiling Water Reactors" further addresses this important safety concern.

One of the primary aging related failure modes for valves is seat leakage. Technical specification requirements include periodically monitoring isolation valves for leakage during outages. This Notice recommends other on-line activities for monitoring valve leakage such as slowly increasing suppression pool level.

2.4.11 Notice 86-36: "Failure of RHR Pump Motors and Pump Internals."

This notice was initiated due to degraded pump impeller wear rings and motor bearings found in 6 of 8 RHR pumps at a BWR. The motor guide bearing failures could cause failure of the pump motors or pump internal damage. Aging factors involved with these failures are the evidence of intergranular stress corrosion cracking (IGSCC) contributing to the wear ring failures and developed high internal temperatures due to inadequate flow and lubrication leading to pump cavitation and damage.

Contact with the pump manufacturers indicate that they are cognizant of the experiences described and have made recommendations for periodic inspection and replacement as appropriate. This information has been requested and should be available for evaluation as part of the phase 2 NPAR study.

2.4.12 Notice 85-59: "Valve Stem Corrosion Failures."

This Notice alerted nuclear power reactor facilities to a potentially significant problem pertaining to stress corrosion failures of valve stems and shafts. A 24 inch LPCI injection valve broke in two places during disassembly. The cause of failure was IGSCC which had affected over 50% of the stem cross-section. The cracking occurred in internal areas where there could be concentrations of corroding chemicals, such as at the gland packing.

Of significance here is that the cracks were not detected by the routine valve operability test programs, but were only discovered by actual failures or after disassembly during refueling outages. Additional information is required to determine if this degradation is detectable prior to catastrophic failure.

2.4.13 Notice 85-30: "Microbiologically Induced Corrosion of Containment Service Water System."

Microbiologically induced corrosion (MIC) occurs as a result of living organisms in contact with the materials of construction. In standby systems such as RHR, stagnant conditions favor biofouling and concentration cell corrosion. The Notice indicates that rapid fluid flow tends to prevent attachment of organisms.

The quarterly pump testing requirements for RHR pumps may assist in reducing the potential for MIC. However, discussions with a manufacturer involved with pump refurbishment indicates that MIC has been found in RHR pump internals. Protective coatings are being considered to preclude degradation due to this aging mechanism.

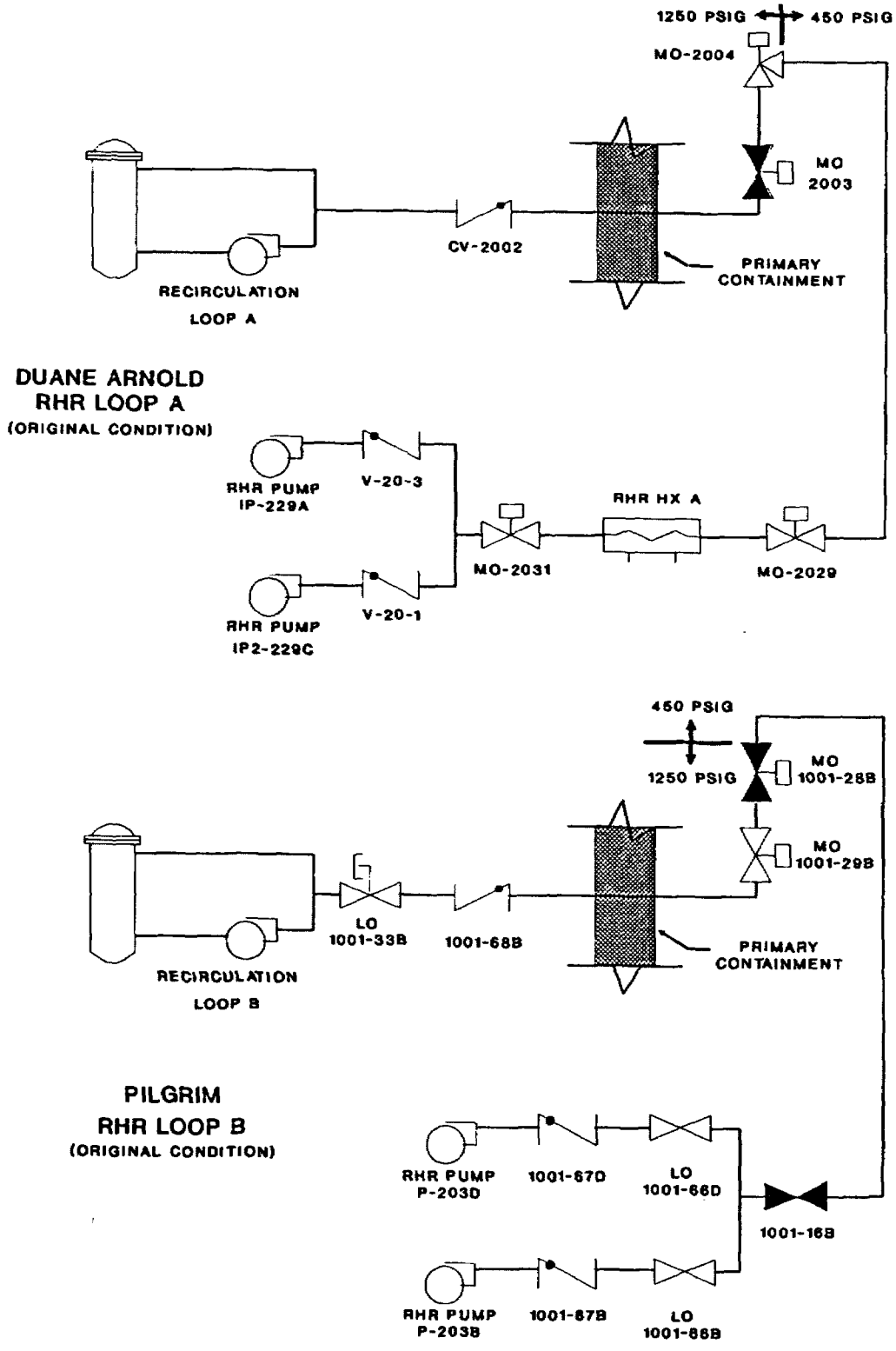


Figure 2.6 High/Low Pressure Interface Identified in Notice 86-40

3. OPERATIONAL AND ENVIRONMENTAL STRESSES

Degradation mechanisms attributed to aging occur in materials subjected to certain stress conditions over a period of time. These processes are well understood when one type of material is exposed to one kind of stress condition. However, with the complexities of composite materials, or a component made of many different materials (which is the actual case for most component designs), and the synergistic effects of several stress conditions, these processes are difficult to understand. Extensive laboratory testing and material analyses are necessary to characterize these complex phenomena. Also, since aging is a time-dependent process, considerable time would be necessary to completely understand the true characteristics in a plant environment and operating conditions.

Aging is a degradation process (or mechanism) which exists at every level in a plant's hierarchy. If unchecked, it can limit the life of a component, system, or structure, and increase the risk to plant safety. Therefore, understanding aging phenomena is important for the safe operation of a nuclear plant. Typical aging mechanisms which cause a material's mechanical strength or physical properties to degrade include fatigue stress cycles (thermal, mechanical, or electrical), wear, corrosion, erosion, embrittlement, diffusion, chemical reaction, cracking or fracture, and surface contamination.

Each mechanism can occur in various materials when they are exposed to particular operating and environmental conditions. Abnormal conditions or accidents accelerate the aging process, thus weakening the material faster than normal. These abnormal conditions include plant mechanical and electrical transients, pipe breaks, exposure to harsh environment, and other abnormal and accident scenarios.

This section discusses the operational, environmental, and accident parameters which can degrade the mechanical strength or electrical/chemical properties of components in the RHR system. These parameters include system and component level stresses such as those induced by testing, human factors, environmental parameters and their synergistic effects. The correlation with accident conditions when the function of the RHR system becomes vital for plant safe shutdown also is discussed.

3.1 System and Component Level Stresses

The RHR system serves a variety of purposes in the operation of a BWR plant during normal, abnormal and emergency (or accident) conditions. The six major operating modes, each of which has its own design functions are summarized in Table 3.1. The last two modes, steam condensing and fuel pool cooling, serve as alternates in cases where the main condenser or the fuel pool cooling heat exchangers are not available. The suppression pool cooling and shutdown cooling modes are periodically used; the former mode cools the suppression pool water to maintain its temperature within allowable limits during normal plant operation, while the latter mode is used each time the reactor is shutdown for refueling, maintenance or any abnormal condition. The low pressure coolant injection and containment spray cooling modes require the RHR system to be in standby during normal operation of the plant.

Table 3.1 RHR System Functions During All Plant Condition

RHR Mode of Operation	RHR System Status During Plant Operation: Normal/Abnormal Conditions	RHR System Status During Plant Operation: Emergency (or Accident) Condition
Low Pressure Coolant Inject.	Standby	Operating(under LOCA)
Containment Spray Cooling	Standby	Operating(under LOCA)
Suppression Pool Cooling	Periodic	Operating(if necessary)
Shutdown Cooling	During Shutdown (Refueling/Servicing)	Operating(if necessary)
Steam Condensing	Normally standby; operates to maintain reactor in hot standby when main condenser is not available.	
Fuel Pool Cooling	Normally standby; operates in conjunction with fuel pool cooling heat exchangers to increase heat removal capability.	

None of the six modes discussed earlier are continuously operating and, hence, are not essential for normal plant operation. Under normal plant operating conditions, the RHR system is aligned to the LPCI mode and is maintained in a standby condition. All other modes are periodically utilized depending on the needs during the life of the plant. The effects of operating stresses on various RHR components are therefore less severe than in a continuously operating system since they are not active all the time. However, these components do experience operating stresses during periodic operations, as well as testing which is discussed in the subsequent section. Aging mechanisms, such as wear, erosion, physical and electrical property deterioration, and contact surface degradations typically are attributed to these stresses. A typical failure resulting from one or more of these mechanisms is reported in LER 259/83-068, in which a pump failed while operating due to bearing failure followed by motor winding failure.

The standby modes require the system to remain filled so that the heat removal function from the reactor can be achieved on ECCS actuation. Operation in a standby condition allows various age-related degradation processes to occur, including corrosion, hardening of polymeric water seals, contamination of electrical contact surfaces with dusts and other chemically active agents. For example, LER 220/84-001 reports circuit breaker failures caused by age induced hardening of grommets in the electromechanical overcurrent device.

3.2 Stresses Induced by Testing

To ensure availability of the flow rate required for LPCI operation, which is the limiting flow rate operating mode, a full flow test line is provided. The test loop consists of suppression pool, pumps, heat exchanger bypass, and the full flow test line. Neither the flow test nor the LPCI mode utilizes the RHR heat exchangers while operating. A frequency of once per 3 months is required for this test. Therefore, the full flow test in many plants is considered to be a separate mode for the RHR system operation.

The operating time in any mode is dependent on the plant operating schedules, technical specification commitments, and maintenance and surveillance testing. Hence, the total time any component within this system is exposed to operating stresses is far less than its actual age. However, each time the system is utilized, the high stress conditions imposed by starting the system or components contribute to their age-related deterioration. Specifically, periodic testing of the system, as well as individual component surveillance and maintenance testing impart stress conditions on RHR components, such as pump starting and stopping. Typical tests on RHR systems are the following:

- Pump operability test - once/month
- MOV operability test - once/month
- Pump capacity test - once/three months
- Simulated automatic actuation test - once/operating cycle
- Logic system functional test - once/six months

In addition to these, maintenance related tests are conducted periodically on components such as pumps, valves, heat exchangers, piping and snubbers to monitor their performance. These include valve stroke testing, motor insulation testing, bearing vibration testing, snubber operability testing, pipe/valve leak testing, and heat exchanger flow and tube leak testing. All these tests are performed at certain frequencies as predictive measures and at the time of corrective maintenance. Therefore, the frequency of these tests vary from plant to plant and are dependent on the utility practices and their operating experience. Since the RHR system is instrumented for monitoring operating conditions, as well as automatic actuation, setpoint drift is expected to be a dominant failure mechanism. System or component level vibrations, as well as the environmental conditions inside the reactor building where most of the system is located contribute to these instrument failures.

Some of the operability and functional tests require the associated part of the system or the component to remain operating (or functioning) for a period of time to arrive at a steady state indication. The stresses due to these tests together with normal operating stresses (on demand) contribute to the age-related degradation in the RHR system and its components. Certain maintenance tests such as high-potential electrical testing or hydrostatic tests on piping could impart a larger stress level than the component is designed to experience normally. This can also contribute to aging degradation.

3.3 Stresses Induced by Human Performance

Like all systems in a nuclear power plant, successful operation of the RHR system requires a great deal of human interaction. This begins with the initial design, manufacture, and installation of the system and its various components, and continues throughout the life of the plant in normal operational and maintenance activities. Even with the rigorous training programs typically implemented, the possibility of human error still exists. This is verified by past operating experience which indicates that human errors do contribute to failure of RHR system components. The failures most commonly reported involve errors in maintenance and installation, along with procedural errors.

Maintenance errors are very diverse, not only in their causes, but also in the components affected. Any component which requires maintenance to be performed on it at any time in its life is subject to maintenance errors. Obviously, this includes virtually all components in the plant, therefore, the potential for this type of error to occur is quite large. It is only through rigorous training and experienced personnel that maintenance errors are kept at a low level. Typical maintenance errors include the use of wrong replacement parts, improper tightening of bolts or screws, and improper lubrication of moving components. An example reported in LER 259/80-043 involves a heat exchanger gasket leak which developed due to improper installation of lock nuts during a previous maintenance activity.

Installation errors are also found which adversely stress equipment and lead to premature failure. These errors can occur during initial component installation when the plant is first built, or they can occur during component replacement throughout the life of the plant. Examples of installation errors include improper alignment of pumps, which can lead to wear and premature failure of seals and bearings, or incorrect adjustment and calibration of instrumentation, which can result in erroneous readings.

Human errors related to operating procedures have also been found which lead to failures. Since most systems and many components have very complex designs, involving various interlocks and subsystem alignment checks, detailed operating procedures must be followed to properly operate them. If the procedures are not properly written and not properly performed, the system or component could fail to operate or it could be severely damaged. For example, LER 237/83-020 reports failure of a core spray injection valve to close when a close signal was given. The failure was attributed to procedural inadequacy in which a breaker was incorrectly set. This occurred because an incorrect data sheet, not controlled by station procedure, was used as part of the procedure.

From this discussion, it is seen that many different types of human error are possible. Some may not affect system performance, however, others can result in failure of a component. These failures may or may not be aging related. While it may not be possible to completely eliminate human errors, it may be possible to mitigate them if the area in which they are predominant is identified. Therefore, part of this study will address the RHR failures caused by human error.

3.4 Environmental Effects

Most RHR components, such as pumps, heat exchangers, piping, valves, and instrumentation are located between the primary and secondary containment. The only interfaces involving piping and valves inside the primary containment are injection lines to the suppression pool, recirculation loops, containment spray headers, and the reactor vessel. These components are exposed to the most severe temperature, humidity and radiation condition. Typical values are shown in Table 3.2.

Table 3.2 Environmental Stresses on RHR Components in Primary Containment

Parameter	Normal	Accident
Temperature	135-150°F	200-340°F
Relative Humidity	40-90%	100% (all steam)
Radiation Integrated over 40 years		
Gamma	1.8×10^7 rads	2.6×10^7 rads During LOCA 1.3×10^6 rads/hr
Neutron	1.8×10^{14} neutrons/cm^2	None

Other RHR components located in the secondary containment area are exposed to a normal temperature range of 70-140°F with accident condition temperatures up to 212°F. The humidity condition is similar to that of the primary containment. There is no design basis radiation level for this area.

3.5 Summary of Stresses

Unlike a continuously operating system (i.e., component cooling water or service water) the RHR system remains in standby most of the time during normal plant operation. Hence, the predominant aging mechanisms due to operation, wear and erosion, are mitigated. The static components remain energized and active throughout their life and are subjected to both normal operational and environmental stress conditions. Table 3.3 summarizes some of the potential aging effects that are significant to RHR components. One important source of stress which contributes to RHR failures is testing. Since this system is categorized as a safety-system, components are more frequently tested to assure their operability. These tests may contribute high stress levels on the components causing premature failures. These insights were used as a baseline for evaluation of the results from data analysis discussed in the following sections.

Table 3.3 Aging Effects on RHR System Components

Stress Condition	Aging Effects	Mechanical	Electrical	Inst. & Control
Normal Operating Conditions	Erosion, wear, corrosion, crack, leakage	X		
	Clogging, blocking, reduced flow	X		
	Vibrations, misalignments, crack growth, loose or dislodged pieces	X	X	X
	Mechanical binding, distortion, rupture	X		
	Set point drift, out of calibration, loose connections		X	X
	Electrical shorts, grounds surface pittings, erratic signals/indicators		X	X
Normal Environment Conditions	Corrosion, cracks, surface damage (e.g., pitting)	X	X	X
	Burning, shorts, grounds		X	X
	Embrittlement, hardening	X	X	

4. CURRENT UTILITY PRACTICES

A survey of current utility practices regarding RHR operation and maintenance was performed. To conduct the survey, a form was developed by BNL and, with the co-operation of the Equipment Qualification Advisory Group, was forwarded to their member utilities. The results were complied by a contractor for the Electric Power Research Institute (EPRI) and released to BNL, along with copies of procedures for operating and testing the system. Nine BWR units responded to the survey and five units submitted information copies of their procedures.

The intent of this survey was to identify the type and amount of periodic testing, preventive maintenance, and corrective maintenance performed on the RHR system. This section evaluates the survey results and the procedure contents to determine if aging detection and mitigation methods are currently being implemented by some utilities. The results are presented for the major components - pump/motor, valves, heat exchangers, and instrumentation.

4.1 RHR Pumps

RHR pumps and motors are typically located in the lower levels of the secondary containment building. During power operation, they are in a de-energized state in an area accessible by plant personnel. Therefore, external equipment conditions can be routinely observed. However, the ability to inspect or maintain the pump and motor at any time is restricted by the technical specifications (tech specs). In accordance with the plant tech specs, an RHR pump may be removed from service during power operation for testing or maintenance for a specific time, as identified by the limiting condition for operation (LCO). This time can vary from 7 to 30 days depending upon the plant design and the operational state of other emergency systems. Similar restrictions would apply during shutdown conditions when decay heat removal or fuel pool cooling capability is required. As required by tech specs, pump capacity testing is periodically conducted. As a minimum, therefore, utilities operate the RHR pumps in a test configuration at least once per quarter to verify minimum flow and pressure parameters, as well as auto-start capability are maintained. Some plants take advantage of this testing to monitor other pump and motor characteristics. Some also perform preventive and predictive maintenance, as illustrated in Table 4.1 and described in the following paragraphs.

4.1.1 Periodic RHR Pump Testing

RHR pump testing is required on a quarterly basis in accordance with the standard tech specs.[14] The test requirement is for the pump capacity to be demonstrated through its ability to provide a certain flow at a given discharge pressure condition. It was observed by the review of several plant surveillance procedures that some plants record additional pump information during the conduct of this test, including bearing vibration and temperature, motor amps and voltage, and motor winding temperature.

Because the RHR pumps are normally in a standby mode, it is important that as much information as possible be obtained during the quarterly runs. Bearing degradation can be detected by increasing vibration and temperatures. Likewise,

motor and pump degradation may be apparent from increasing motor current or motor winding temperatures. Four of the plants responding to the survey felt that the in-service pump test was a key input for assessing the operational readiness of the RHR system.

4.1.2 RHR Pump Preventive Maintenance

The survey of current utility practices revealed that preventive maintenance (PM) of the RHR pumps consists primarily of inspection and lubrication. Some plants supplement these activities with periodic cleaning and alignment checks. Due to environmental qualification requirements, the plants indicated that bearings, seals, or gaskets are replaced at prescribed intervals.

The frequencies at which PM activities are performed are normally based on the equipment operating experience, plant configuration, and the impact of pump failure on plant risk. As noted in Table 4.1, pump inspection frequency varies from 18 months to 120 months, with an average inspection frequency of 56 months. Similarly, lubrication frequency varies from 18 months to 60 months, with an average of 29 months. In general, these PM activities appear minimal but may be justified based on the strengths of the periodic testing conducted. That is, as long as the quarterly tests monitor the major components of an RHR pump/motor set, and readings are monitored to detect any trends, then limited PM is necessary. However, minimal activities such as cleaning and lubrication should be periodically conducted regardless of whether the pump is in standby or continuously operated.

4.1.3 RHR Pump Corrective Maintenance

The survey requested the utilities to summarize the corrective maintenance performed on the RHR pumps. The 4 units responding to this question cited 33 events involving 7 different failure modes. Of these failure categories, the ones most critical to pump operation are pump seal replacement, motor overhaul, pump overhaul, and bearing replacement. It is expected that corrective maintenance conducted to repair oil leaks, replace suction pressure gauges, or repair the seal oil cooler were necessary to mitigate a degrading condition. However, it is likely that the pump would still have performed its safety function despite the existence of these problems. The number of critical corrective maintenance events from the 4 respondents is, therefore, estimated to be 24 events.

4.2 RHR Valves

Upon receiving a LOCA signal, RHR valves must transfer from their standby status to the LPCI injection mode flow configuration. When operating in the shutdown cooling mode, valves must realign automatically under certain conditions to assure containment integrity. Periodic testing, preventive maintenance, and corrective maintenance are performed to assure valve integrity to support these design requirements.

Table 4.1 RHR Pump Utility Practices (9 units surveyed)

Activity	Units	Parameter Measured/Task	Frequency (months)
Periodic Testing	8	Vibration, suction and discharge pressure,	(3)
		flow, pump ΔP	(3)
		Bearing temperature, motor winding temp., amps, and voltage	(3)
Preventive Maintenance	6	Inspection	2-(18) 3-(60) 1-(120)
	2	Cleaning	1-(3) 1-(18)
	5	Lubrication	2-(18) 2-(24) 1-(60)
	1	Alignment	(18)
	1	Replace bearings	(84)
	1	Replace seals and gaskets	(60)
Operational checks (when system is operating)	7	Pump discharge pressure	(1/shift)
	2	Motor amps, winding temperature	(1/shift)
		Task	Events
Corrective Maintenance	4	Replace bearings	8
		Replace suction pressure gauge	5
		Overhaul pump	5
		Overhaul motor	6
		Repair seal water cooler	2
		Repair oil leak	2
		Replace pump seal	4
		Replace oil seal	1
TREND Analysis	4	Pump inservice test results (ASME)	NOT AVAILABLE
	1	Pump ΔP, vibration, lube oil analysis	
	2	Maintenance work requests (MWRs) trended to identify recurring failures	

4.2.1 RHR Valve Periodic Testing

In accordance with the tech specs, periodic testing is required for key RHR valves. This testing is performed as specified in the ASME Boiler and Pressure Vessel Code, and supplemented by 10CFR part 50. The testing includes valve stroke time and valve seat leakage measurements. The standard technical specifications also require that each valve in the flow path be periodically checked to verify its correct position.

The survey results indicated that all units perform stroke time testing of valves, however, the frequency varies from monthly to every 18 months. Seat leakage measurement was reported by 6 of the units, but again the frequency varies from quarterly to every 2 years. Other periodic testing performed includes:

- relief valve setpoint verification (every 5 years)
- MOV signature analysis (every 18 months)
- position indicator function test (every 18 months)

The number of units performing these tests is identified in Table 4.2.

In regard to MOV signature analysis, it should be noted that as a result of problems with MOV switch settings, as identified in NRC Bulletin 85-03, utilities have been giving MOVs increased attention. Bulletin 85-03 did not originally include the RHR system valves in its scope, however, some utilities voluntarily extended their programs to cover these valves. NRC is currently processing a generic letter to require the extension of this program to all safety-related MOVs. The most common method of addressing these MOV problems is through a signature analysis program.

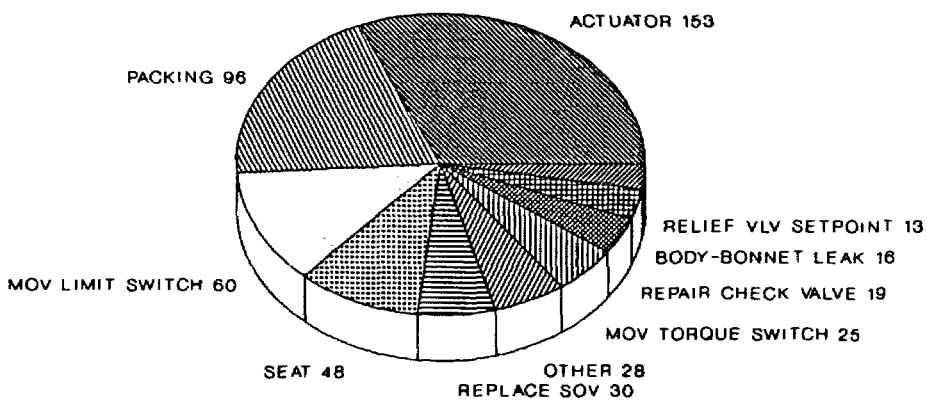


Figure 4.1 RHR Valve Corrective Maintenance Events

Table 4.2 RHR Valve Utility Practices
(9 units surveyed)

Activity	Units	Parameter Measured/Task	Frequency (months)
Periodic Testing	9	Stroke Time	1-(1) 5-(3) 3-(18)
		Check valve leakage	4-(3)
		Position indicator verification	1-(3) 3-(18)
		Seat leakage (Type C LLRT)	1-(3) 3-(18) 2-(24)
		Relief valve setpoint verification	3-(60)
		MOV signature analysis	1-(18)
Preventive Maintenance	9	MOVs - inspection, lubrication	1-(12) 4-(18) 3-(24) 1-(48)
	4	- megger	3-(24) 1-(48)
	4	- functional checks	1-(18) 2-(24) 1-(48)
	3	Check valves	(18)
	5	Air operated valves - replace parts	1-(48) 1-(84) 3-"As Req'd."
	2	Solenoid Operated Valves	1-(60) 1-(396)
Operational checks	2	Valve line-up verification	(1/shift)

Table 4.2 RHR Valve Utility Practices
(9 units surveyed)
(Cont'd.)

Activity	Units	Tasks	Events
Corrective Maintenance	4	Repair seat leakage	48
		Repair body-bonnet leak	16
		Adjust/replace packing	96
		Repair MOV actuator (mechanical)	58
		Repair AOV actuator (mechanical)	24
		Repair MOV actuator (electrical)	68
		Repair MOV torque switch	25
		Adjust position limit switches	60
		Replace actuator	9
		Modify handwheel to prevent loosening	10
		Adjust relief valve setpoint	13
		Replace solenoid valve	30
		Repair check valve	19
TREND Analysis	4	Pump inservice tests are trended (ASME)	Not Available
	1	MWRs trended to identify recurring failures	Not Available

4.2.2 RHR Valve Preventive Maintenance

The survey of utilities revealed that valve preventive maintenance (PM) is focused on MOVs, and consists primarily of inspection and lubrication. The frequency of this activity ranges from annually to every 4 years, with the average being every 2 years. Four utilities also periodically megger the motors on the MOV.

Air operated valves (AOVs), solenoid operated valves (SOVs), check valves and manual valves also receive PM by some of the units surveyed. Inspection and lubrication of manual and check valves are cited, while AOV and SOV PM is based on EQ requirements for parts replacement, although the specific parts with limited service lives were not provided.

4.2.3 RHR Valve Corrective Maintenance

The four units responding to this portion of the survey indicated that 488 corrective maintenance (CM) tasks had been completed, as illustrated in Figure 4.1 and Table 4.2. As expected, MOVs had the largest number of CM tasks due to their large population. Packing adjustments and replacements, actuator problems, and MOV torque switch/limit switch repairs dominated the activities. In the data base analysis, it was generally assumed that valve packing and seat leakage failures were age related. It was noted in the procedures reviewed that some plants require that the valve be stroked following packing, torque switch, or limit switch adjustments. This post work testing is a recommended method for assuring that maintenance was properly performed, and that the valve meets its tech spec requirements. The data obtained from this testing should be input to the inservice inspection program.

4.3 RHR Heat Exchangers

The RHR heat exchangers are sized to accommodate the decay heat removal requirements of the RHR shutdown cooling system. They are typically bypassed when the RHR system is operating in the LPCI mode.

Relatively little information on the heat exchangers was obtained from this survey. This passive component performs an important function in removing decay heat and the potential for a degraded shutdown cooling capability is conceivable considering flow blockage of the tube sheet and fouling of the tubes. A number of activities are available for detecting and mitigating aging degradation of this kind. The survey indicated that some of these activities are performed by several of the survey respondents (Table 4.3).

In-service testing, such as a heat balance, is a standard non-intrusive method for determining heat exchanger capacity. Design calculations for the heat exchanger assume a certain amount of fouling. The heat balance will determine if the design assumptions remain valid. Other periodic tests such as a hydrostatic test or leak test performed by two units verify the integrity of the pressure boundary interfaces. Finally, in-service non-destructive examination performed by one unit provides information regarding crack formation and tube wall thinning.

The preventive maintenance activities specified include inspection, tube cleaning, and a periodic replacement of gaskets.

When the RHR system is operated in the shutdown cooling mode, two units reported that they regularly monitor the differential pressure across the heat exchanger. This is a relatively easy reading to obtain and should be taken whenever the system is placed in service. Corrective maintenance was minimal and primarily involved tube cleaning.

4.4 RHR Electrical, Control, and Instrumentation

The RHR system contains a large amount of electrical, control, and instrumentation (ECI) equipment to perform the following functions:

- monitor system performance
- automatically start the system in the LPCI mode
- automatically isolate the system when in the shutdown cooling mode.

The instrumentation and controls associated with system operation are not directly required to be tested per the tech specs. However, to support system capacity testing, which is required, some routine calibration of the flow and pressure devices employed must be demonstrated. Instrumentation associated with LPCI injection or shutdown cooling isolation are required to be functionally tested and calibrated on a regular basis per the tech specs. Other than the testing required by tech specs, the utilities responding to the survey indicated that additional periodic testing and preventive maintenance were also performed to detect or mitigate degradation, primarily setpoint drift.

4.4.1 Periodic Testing

The standard tech specs for BWRs require that isolation actuation instrumentation channels and the LPCI actuation instrumentation be demonstrated operable by performing channel checks, channel functional tests, and channel calibrations at regular intervals. In general, the channel checks, which consist of comparing similar instruments to ascertain reasonable accuracy, are performed daily. Functional tests, which involve a determination that the logic is operable, is performed monthly. Actual calibration of the channel to verify accuracy is conducted on a quarterly basis or on an 18 month interval, depending upon the type of instrumentation used.

The required periodic testing is comprehensive for the instrumentation, but there are other electrical and control components associated with the system. Three units perform response testing of circuit breakers, two perform relay functional tests, and one cited a monthly logic functional test. With the circuit breakers and relays in a normally de-energized condition for the LPCI mode, it is important that they periodically be exercised to verify proper operation.

4.4.2 Preventive Maintenance (PM)

Twelve different PM activities were listed by the responding units. All utilities stated that some PM was performed on ECI equipment. The more commonly performed PM are listed in Table 4.4 and include cleaning, inspecting, lubrication, and calibration, with the first three activities associated primarily with electrical components and the latter with instrumentation. Meggering motors is also a common PM activity. Calibrations include a power supply, transmitters, an ammeter, and pressure switches.

Table 4.3 RHR Heat Exchanger Practices
(9 units surveyed)

Activity	Units	Parameter Measured/Task	Frequency (months)
Periodic Testing	1	Hydrostatic Test	(12)
	1	Heat Balance	(18)
	1	Leak Test	(12)
	1	Inservice NDE	(24)
Preventive Maintenance	3	Inspection	2-(12) 1-(18)
	1	Tube Cleaning	(12)
	1	Replace Gaskets	(240)
Operational Activity	2	Monitor ΔP across heat	1-(1/shift) 1-(12)
		Tasks	Events
Corrective Maintenance	4	Clean tubes	12
		Recoat water box	3
		Repair hand hole leak	1
TREND Analysis	2	Heat exchanger ΔP, flow, heat balance, and heat transfer coefficients	Not Available

Cleaning and inspection activities vary from annually to every 4 years with an average of 2 years. Calibration activities appear to revolve around the refueling cycle and are typically every 18 months.

4.4.3 Corrective Maintenance

Despite the seemingly high level of periodic testing and preventive maintenance performed on ECI equipment, failures still occur. The corrective maintenance activities summarized in the survey are illustrated in Figure 4.2. They are dominated by loop calibrations for instrumentation (due to set point drift), and cable splice replacement and breaker repairs for electrical components.

In addition to setpoint drift concerns, other age-related degradation of RHR ECI equipment is apparent from the following activities:

- replace transmitter O-rings
- tighten loose connections
- clean contacts.

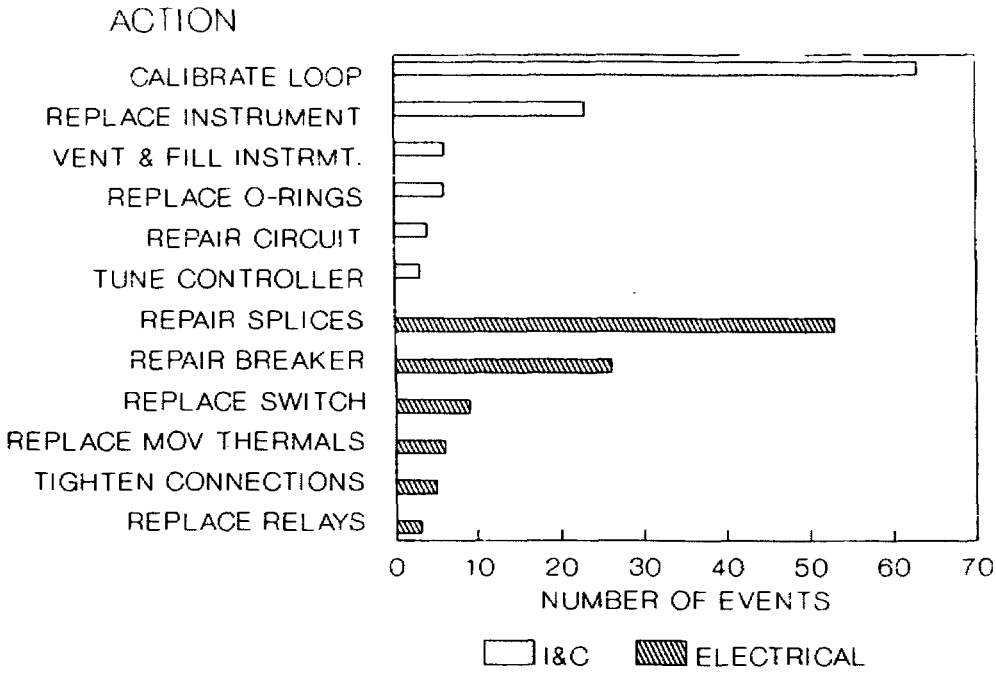


Figure 4.2 Corrective Maintenance on Electrical, Control and Instrumentation Equipment

Table 4.4 Electrical, Control, and Instrumentation (ECI) Utility Practices (9 units surveyed)

Activity	Units	Parameter Measured/Task	Frequency (months)
Periodic Testing	1	Logic functional check	(1)
	3	Circuit breaker response	2-(24) 1-(48)
	2	Relay functional test	1-(12) 1-(24)
Preventive Maintenance	7	Clean and inspect	2-(12) 4-(24) 1-(48)
	5	Transmitter calibration	(18)

Table 4.4 Electrical, Control, and Instrumentation (ECI)
Utility Practices
(9 units surveyed)
(Cont'd.)

Activity	Units	Tasks	Events
Preventive Maintenance (Cont'd)	2	Pressure switch calibration	(24)
		Power supply calibration	(60)
	5	Lubricate electrical components	1-(12) 3-(24) 1-(48)
	5	Megger motors	2-(12) 3-(24)
Corrective Maintenance	4	Instrumentation & Control	
		Calibrate instrument loops	63
		Replace instrument	23
		Vent and fill instrument	6
		Repair circuit	5
		Replace transmitter O-rings	6
		Tune controller	3
	4	Electrical	
		Replace MOV thermal overloads	6
		Replace cable splices (EQ)	53
		Repair breaker	26
		Replace panel switch	9
		Tighten loose connections	5
		Replace relay	3
TREND Analysis	2	MWRs trended to identify recurring failures	Not Available

4.5 Other RHR Equipment

Table 4.5 identifies utility practices involving other RHR equipment. These include periodic testing, such as hydrostatic tests to ascertain piping integrity. Overall system logic response time tests were indicated by 8 of the 9 units. This would generally involve simulating a system perturbation and verifying that all equipment, including pumps and valves, operate within design parameters.

The other system equipment identified in the survey are snubbers and pipe hangers. As illustrated in Figure 4.3, the number of corrective maintenance tasks associated with this equipment is relatively high. However, it is likely that the system operational effects were minimal as a result of these events.

Table 4.5 RHR System Utility Practices - Other Items
(9 units surveyed)

Activity	Units	Parameter Measured/Task	Frequency (months)
Periodic Testing	4	System leak check	1-(3) 3-(18)
	8	System logic response time	7-(18) 1-(6)
		Hydrostatic test	1-(96) 6-(120)
Operational Activities	6	Monitor keep fill operability	3-(1/shift) 4-(1)
		Tasks	**Events**
Corrective Maintenance	4	Repair snubber	22
		Repair pipe hanger	28
		Stroke test snubber	19
		Align snubber	6
		Clean steam trap	2

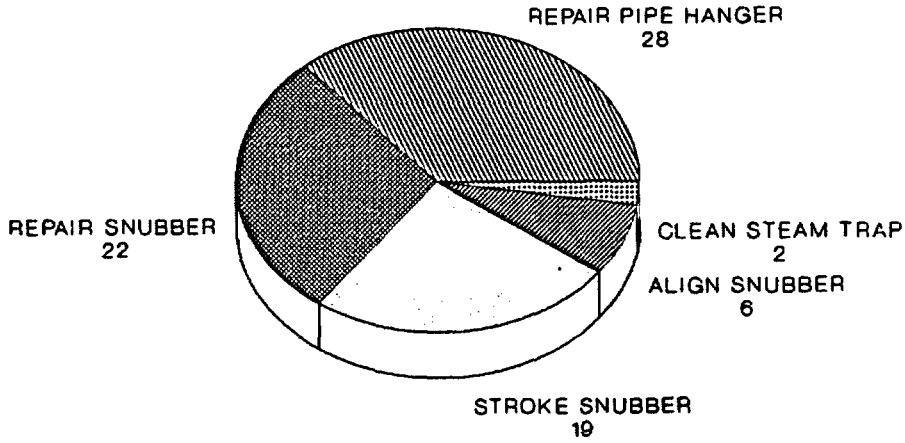


Figure 4.3 Corrective Maintenance on Miscellaneous RHR Equipment

4.6 Summary

Based on the survey of nine BWR units and the review of selected procedures from five of these units, the following observations are made:

1. Activities associated with determining component and system operational readiness are primarily related to requirements identified in the tech specs. These are pump and valve testing, instrumentation and logic calibrations, and system line-up verification checks.

2. The number of corrective maintenance tasks involving valves is significant. The survey indicated that a diagnostic system for MOVs is being implemented on a limited basis by at least one of the units. It is probable, however, that the benefits of this type of monitoring has not yet been realized.

3. The survey respondents indicated that the three key parameters monitored or tests conducted to assure operational readiness of the system are the:

   - In-service inspection pump test,
   - Logic response test, and
   - Valve stroke test.

4. The components requiring the most corrective maintenance are valves followed by electrical equipment and instrumentation and controls. A summary of the corrective maintenance events for each type of component is presented in Figure 4.4.

5. The predominant failure mode for valves is leakage (internal and external), followed by actuator malfunction. For electrical, control, and instrumentation equipment, the predominant failure mode is calibration drift.

6. From the corrective maintenance tasks reported, it is evident that aging degradation is present in RHR systems, and contributes to failures. Therefore, current maintenance and monitoring practices are not completely successful in detecting all aging degradation.

7. There is some diversity between utilities in the frequencies used and the parameters monitored in current test and maintenance activities. The relative advantages and disadvantages of differing practices in regard to their ability to monitor aging should be addressed in future work.

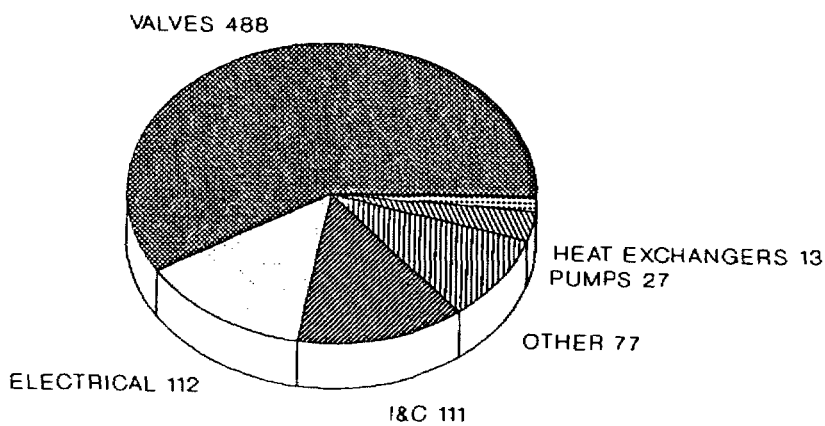


Figure 4.4 Summary of RHR Corrective Maintenance Events

5. EVALUATION OF RHR OPERATING DATA

The aging analysis of RHR systems included an evaluation of past operating experience from various national data bases. This section briefly discusses the data sources used and presents the results.

5.1 Data Bases

5.1.1 Descriptions and Limitations

The data bases used include the Nuclear Plant Reliability Data System (NPRDS), the Licensee Event Reports (LERs), and the Nuclear Power Experience (NPE). The information obtained from each of the sources was reviewed and analyzed to characterize aging effects on RHR systems. The data analyzed included all RHR components and operating modes.

The national data bases have several virtues that make them suitable as sources of aging information. They contain a large amount of data representing a broad cross-section of nuclear power plants. The data is available, although sometimes difficult to obtain. Much of the data includes sufficient information to identify basic failure characteristics, such as the component failed and the reason for failure. With proper review and evaluation, the data can also be used to identify prevailing trends.

Although a great deal of useful information is available from the data bases, there are limitations and weaknesses to it which must be recognized. In general, the data bases do not contain a complete record of all failures. This is partly due to the nature of the data bases and the failures required to be reported. The result is that failure frequencies determined directly from the data base information will probably be lower than actual. However, it must be noted that a large cross-section of plants is represented in the data bases. Using the data for analyzing failure characteristics, such as causes, modes and mechanisms, should not be severely affected by this deficiency. Using the data bases for evaluating aging effects is, therefore, a valid use of the data.

An additional concern with the data base information is the inconsistency in 1) the interpretation of codes used to report events, and 2) the understanding of the events associated with the failure. For example, when a failure is reported, the failed component may be incorrectly identified or the effect of the failure on system performance may not be consistent with other interpretations. This can be attributed to several reasons including a lack of standardized definitions, terminology and reportability for the data bases, as well as differences in experience and knowledge between personnel filing the reports. This is a valid concern in using data base information. However, its effect on analysis results can be mitigated by 1) performing a thorough review of the data, and 2) validating the results by comparison with actual plant data, as was done for this analysis. By performing an independent review using consistent definitions and interpretations, the data base information can provide meaningful results. The results should then be compared with findings from actual plant data to ensure that erroneous trends or failure characteristics are not identified by the data base. Uncertainties in data base results can be addressed by formal uncertainty analyses or by sensitivity studies.

5.1.2 Methods of Analysis

The information obtained from the NPRDS data base was the most extensive and complete; consequently, the majority of the effort spent on data analysis focused on this data. A total of 1673 failure records related to RHR systems were obtained from the NPRDS covering the time period from 1974 through 1986. These were individually reviewed by a team of engineers and then encoded into a computerized data base developed by BNL (utilizing d-BASE III software) specifically to sort and count large amounts of data for the NPAR program.

As part of the NPRDS data review, each failure record was categorized as to whether or not it was related to aging. Since the determinations found in the data records were inconsistent, a definition of "aging related" was established based on the NPAR definition presented in NUREG-1144[9]. This was applied to each event. The following two criteria had to be met in order for a failure to be considered aging-related:

1. The failure must be the result of cumulative changes with passage of time which, if unchecked, may result in loss of function and impairment of safety. Factors causing aging can include:

   - natural internal chemical or physical processes during operation,
   - external stresses (e.g., radiation, humidity) caused by the storage or operating environment,
   - service wear, including changes in dimensions and/or relative positions of individual parts or sub-assemblies caused by operational cycling,
   - excessive testing, and
   - improper installation, application, or maintenance.

2. The component must have been in service for at least 6 months before the failure (to eliminate infant mortality failures).

After all the data were encoded and entered into the BNL data base, the records were checked to verify that they were entered correctly and that the code interpretations were consistent. The data also were checked to verify that the components reported were in the RHR system boundaries defined in Section 2. Once the data base was complete, the data were sorted in various ways to obtain the information for this analysis. The database findings were then checked against actual plant data (discussed in Section 6) to verify the results.

A search of the Sequence Coding and Search System (SCSS) revealed 401 LERs related to RHR systems during the time period from 1980 to 1988. Each event was reviewed and categorized as to whether or not it was aging related. The failure cause, the effect of the failure on RHR performance and the component failed were also identified. The data were then sorted to identify predominant failure characteristics.

The Nuclear Power Experience (NPE) data bank for BWR RHR events was also examined for this study. A total of 241 events were extracted from the data bank and reviewed. The events covered the period from October, 1979 to October, 1987. The information obtained was sorted to identify the predominant components failing and the circumstances leading to failure.

5.2 Dominant Failure Trends

5.2.1 Aging Fraction

One of the primary concerns of this study is to determine if aging degradation is occurring in RHR systems and if it contributes to failures. To accomplish this, the failure records were first reviewed to identify which were related to aging. To make this identification, the NPAR definition of aging was used. This is a broad definition including many causes, as discussed in NUREG-1144, Rev. 1[9]. The data were then sorted to determine the fraction that were aging related. Results from the NPRDS data (Figure 5.1) show that the RHR system is susceptible to aging-related degradation, with 69% of the failures falling into this category. The LER data also shows a large aging fraction, with 50% of the RHR failures being related to aging (Figure 5.2). These results indicate that aging is a concern for RHR systems and should be properly monitored and controlled.

Comparing Figures 5.1 and 5.2, it is seen that the aging fraction obtained from the LER data is smaller than that obtained from the NPRDS data. This is due to the large number of reports in LERs dealing with human errors. These include administrative events such as failures to perform various tests. Events of this type are not reportable to NPRDS, however, they are reported as LERs and are clearly not aging-related.

5.2.2 Failure Detection

As previously discussed, the RHR system has several modes of operation. The most frequently configured modes are LPCI and SDC. Whenever the reactor is at power, the RHR system is aligned for LPCI and is maintained in a standby condition. The SDC mode is used whenever the reactor must be brought to shutdown conditions.

Since the RHR system is most often aligned for LPCI operation, it is expected that most failures will be detected while in this mode. This is verified by the data (Figure 5.3) which indicate that 47% of all RHR failures are detected while in the LPCI mode. As shown in Figure 5.4, testing (42%) was the most common method of detecting failures. Operational abnormalities (27%) and inspections (23%) were the next most frequently used failure detection methods. Operational abnormalities include events such as a valve failing to open on demand or radiation levels exceeding specified limits. The inspections include planned as well as unplanned inspections, for example, where an operator performing a system walk down might notice water leaking from a valve packing.

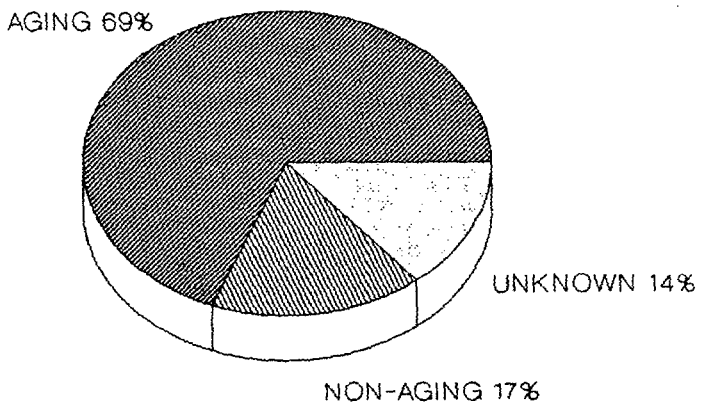


Figure 5.1 Aging Fraction - NPRDS Data

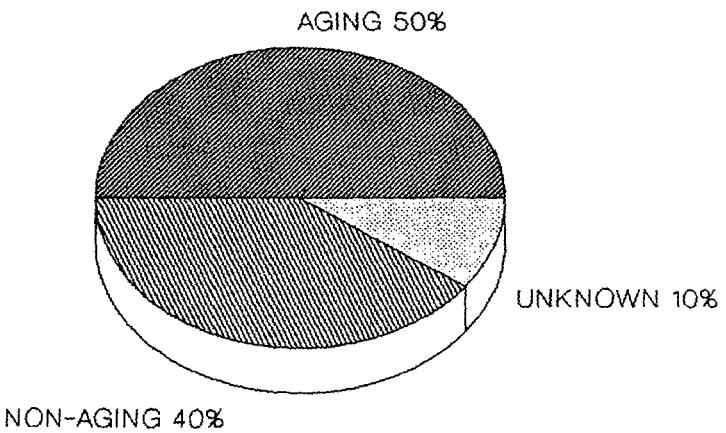


Figure 5.2 Aging Fraction - LER Data

It is significant that an emergency core cooling system such as RHR has 27% of its failures detected by operational abnormalities. It would appear that this would significantly reduce the reliability of such a system. However, it must be pointed out that those failures detected by this method occurred either while in the shutdown cooling mode or while realigning the system from or to the LPCI mode. No failures were detected by operational abnormalities such that LPCI was called for and it could not be provided. The fact that this many failures are detected by operational abnormalities, nevertheless, is important since it indicates that some failures will not be detected by current practices until the component or system is called upon to operate.

Including tests, inspections, maintenance and alarms, 73% of all RHR failures are detected by current monitoring techniques. Since 27% are not found until performance is affected, the potential exists for improvement in failure detection methods. However, the effect of the failure on system performance must also be considered. Improved monitoring methods would be justified only if system performance is degraded to an unacceptable level by the failure. Further work on this subject will be done in the second phase of the RHR system study.

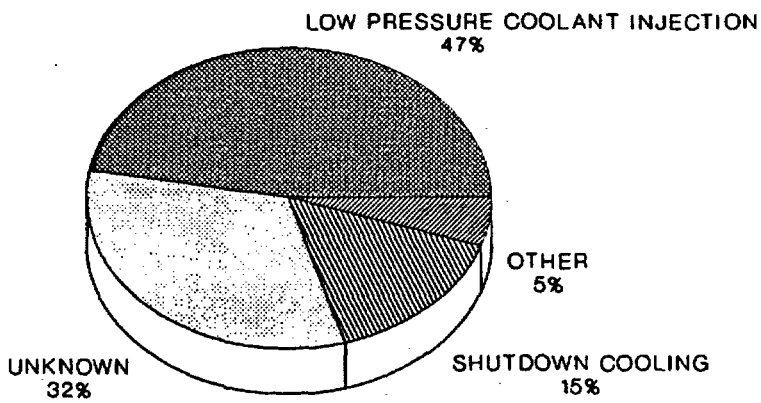


Figure 5.3 RHR Operating Mode During Failure Detection

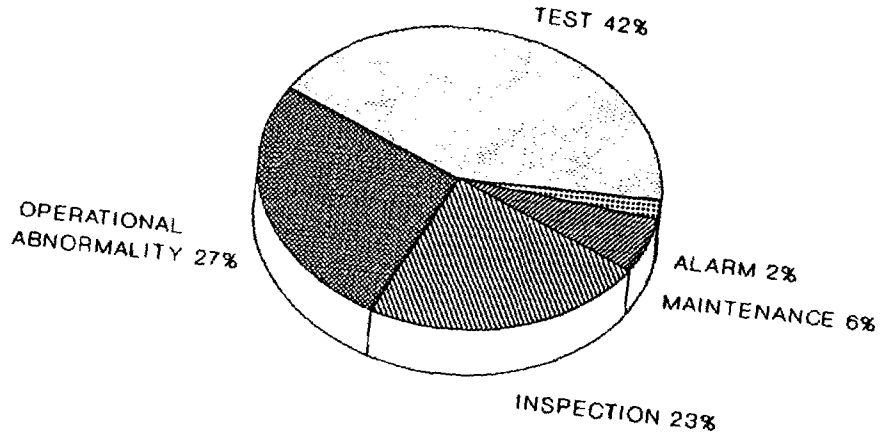


Figure 5.4 RHR Failure Detection Methods

5.2.3 Effects of Failure

The effect of each RHR failure on system performance was also determined from the data. The NPRDS results (Figure 5.5) show that over half of all failures result in degraded operation of the system. This implies that the system can still perform its function, however, the failure will eventually have to be corrected. If left uncorrected, the failure would get progressively worse until there was a complete loss of function or an impairment of safety. Results from a review of the LER data (Figure 5.6) were consistent with the NPRDS findings indicating 33% of the events resulted in degraded operation.

Figures 5.5 and 5.6 also show that approximately one-fourth of the RHR failures result in a loss of redundancy. This would occur, for example, if one loop of RHR used for shutdown cooling became inoperable. All plants have multiple means of shutdown cooling, however, from a PRA standpoint a loss of redundancy in the system would result in an increase in unavailability and a decrease in reliability. Loss of redundancy is, therefore, a significant failure effect.

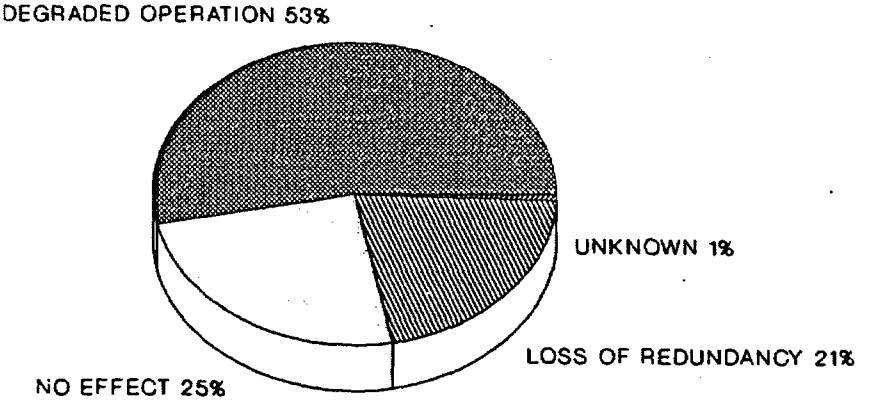


Figure 5.5 Effect of Failure on System Performance - NPRDS

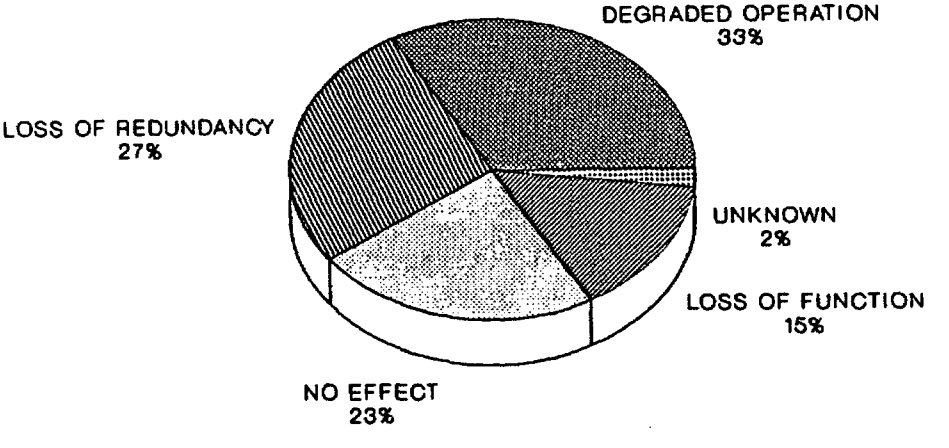


Figure 5.6 Effect of Failure on System Performance - LER

As previously mentioned, no failures were reported in which complete loss of LPCI resulted. However, as shown in Figure 5.6, complete loss of the shutdown cooling mode of RHR did result from some of the failures. This typically involved a failure of the instrumentation controlling the inboard or outboard isolation valves leading from the recirculation line to the RHR system, or failure of the valves themselves. Since the typical RHR design has all shutdown cooling flow passing through one set of isolation valves, failure of either valve can cause loss of flow to all shutdown cooling loops.

The effect of RHR failure at the plant level was also examined using the LER records. There were a significant number of events in which engineered safety features (ESF) were challenged or automatic scrams occurred (Figure 5.7). Several of these events were due to mechanical failures. For example, in one event leakage past the primary and redundant containment isolation valves between the reactor water recirculation system and the RHR system was indicated. The event was classified as an unusual event emergency and a controlled shutdown was initiated. While shutting down, an unplanned automatic reactor scram occurred. In a separate incident, while attempting to establish shutdown cooling flow, a check valve for filling the RHR system would not open. An alternate fill path was established, however, this caused a level transient in the reactor vessel which led to an initiation of the reactor protection system and automatic engineered safety features.

Automatic scrams have also been caused by human error involving the RHR system. For example, in one event an improper valve line up during a fill and vent operation of the RHR system caused reactor water to be diverted from the reactor vessel to the suppression pool. As a result, a reactor scram occurred on low reactor vessel water level.

These effects are important since they can increase plant risk. If an ESF is actuated there is a possibility that it will not be reset properly or it may be damaged during operation. In either case, it would not be available for use the next time called upon. Automatic plant scrams also contribute to increased risk since plant shutdown requires the proper operation of many different controls and components which increases the chance of failure. In addition, each scram can impose pressure transients on the reactor vessel which can reduce its remaining life.

Although most RHR failures do not result in any radiological release, there are some that do. From the LER data reviewed, thirteen events (3%) were found where a release occurred. As shown in Figure 5.8, six events resulted in a release to containment while seven events resulted in a release to the environment. Of the seven releases to the environment, six were due to heat exchanger leaks while one was due to a valve packing leak; all were aging related. The heat exchanger failures are typically caused by deterioration of gaskets or corrosion of tubes and weld areas. The corrosion can occur on the tube side, where stagnant water is present much of the time, or on the service water side.

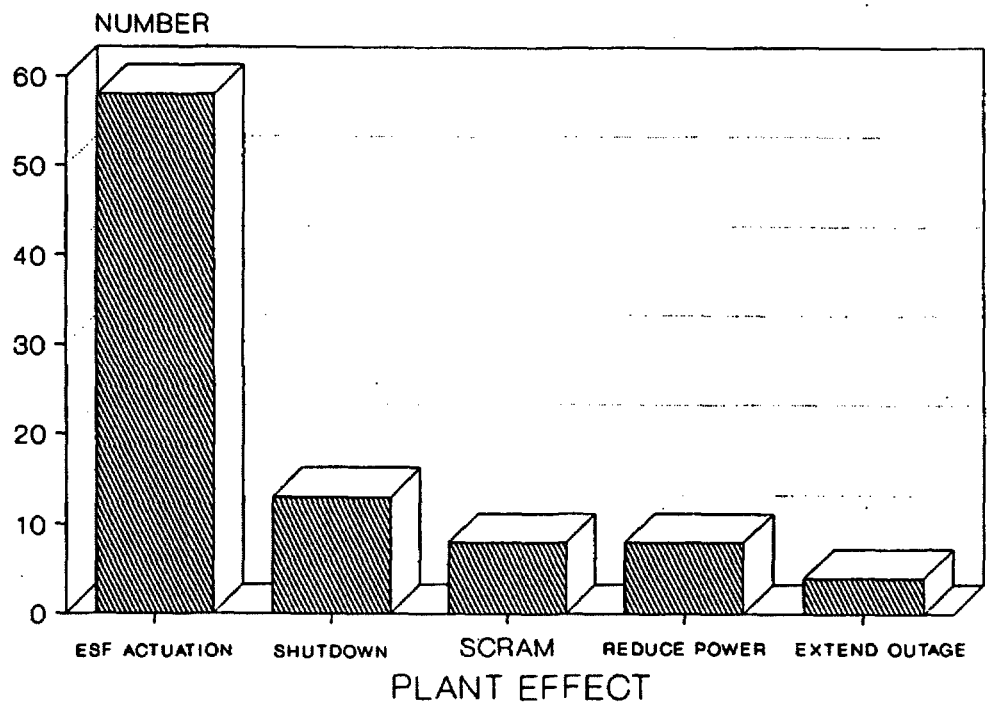


Figure 5.7 RHR Plant Level Failure Effects

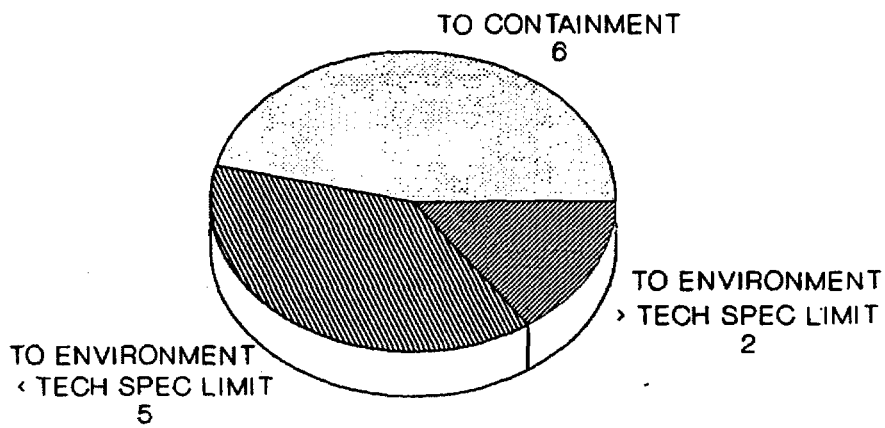


Figure 5.8 Types of Radiological Release Due to RHR Failure

From these results, it is seen that RHR failures can have an effect on plant risk and result in the release of radioactivity to the environment. It is, therefore, important to minimize such failures. Since such a large fraction are related to aging, one method of doing this is to monitor and control aging degradation.

5.2.4 Causes of Failure

The cause of failure is defined here to be the general condition or event which resulted in component failure. For this analysis, RHR failures were sorted into three general cause categories; "normal service", "human error" or "other". As examples of their application, a valve failure where seat leakage has resulted from erosion while in operation would be classified as being caused by "normal service." However, a valve failure where the valve cannot operate because the packing was incorrectly installed would be classified as being caused by "human error." Failure causes classified as "other" include failures of other components and systems outside RHR boundaries, manufacturing and design errors, operation in a harsh environment, and operation during accidents requiring service outside normal limits.

The causes of RHR failures as a function of plant age are shown in Figure 5.9. Normal service is the predominant cause of failure for all plant ages. Since normal service is directly related to aging failures, this is consistent with the large aging fractions seen previously. Analysis of the LER data (Figure 5.10) shows similar results with normal service accounting for 57% of the failures.

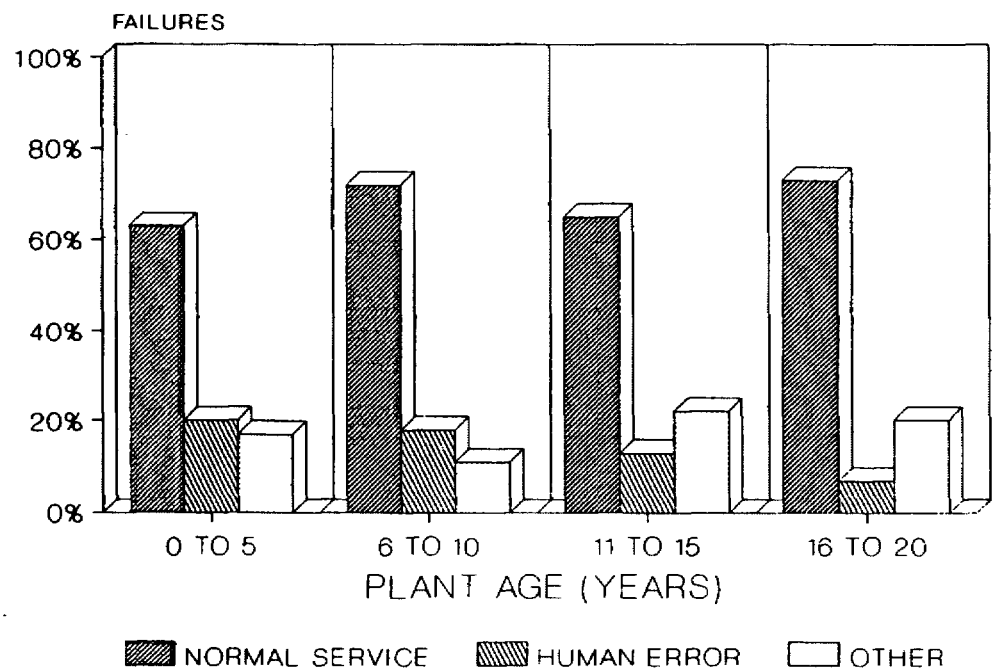


Figure 5.9 RHR Failure Causes Versus Plant Age - NPRDS

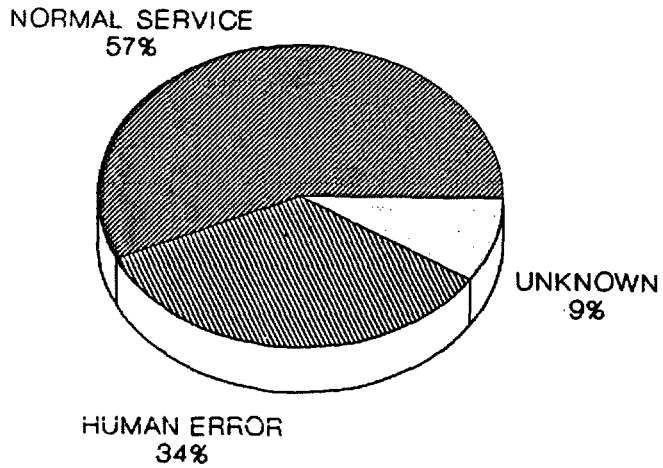


Figure 5.10 RHR Failure Causes - LER

Examination of Figure 5.9 shows that on the average, the relative percentage of failures caused by normal service remains fairly constant with age. This indicates that the testing and maintenance practices currently employed may be effectively controlling any growth in aging related failures. It should be noted that this plot represents a gross estimation of time-dependent aging effects. Although no dramatic increases with age are evident, it does not eliminate the possibility that failure rates for some specific components can increase with age.

As shown in Figure 5.9, human error was the second largest cause of failure in early years. This includes errors in the application and installation of the systems and components as well as errors in their operation and maintenance. To identify areas where improvements may be made, the human error failures were categorized into types (Figure 5.11). Problems related to maintenance were the predominant type of human error (89 events) followed by installation errors (67 events) and operational/procedural errors (42 events). These results show that if efforts are made to mitigate human errors they should be concentrated in the area of maintenance and installation.

It is noted from Figure 5.9 that failures related to human error tend to decrease with plant age. In early years they account for 20% of the failures while in later years they account for only 7%. This may be attributable to personnel becoming more familiar with operating and maintenance procedures and are performing them more effectively (i.e., learning-curve effect).

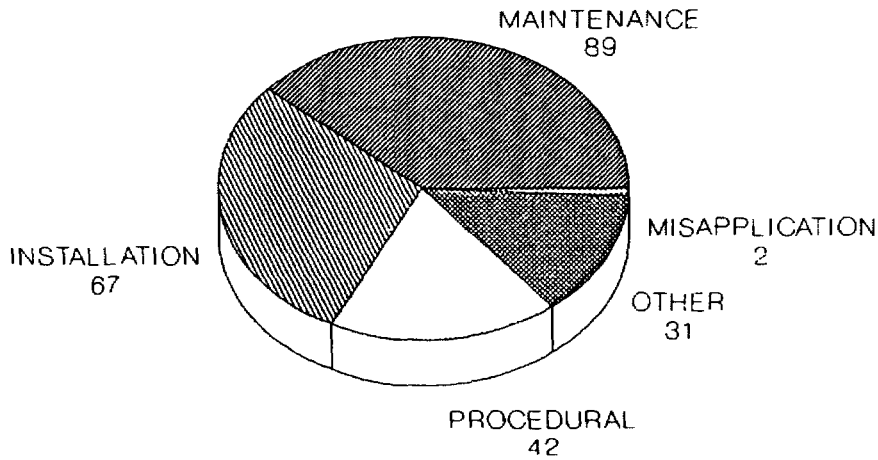


Figure 5.11 Types of Human Error

5.2.5 Modes of Failure

A failure mode is defined to be the manner in which a component fails. For a pump, common failure modes include failure to start or failure to run, while for valves they include failure to open/close or leakage. The failure modes were identified from the data since they are useful in assessing surveillance and monitoring methods.

The failure modes for the RHR system are quite diverse (Figure 5.12). The predominant modes of failure include leakage (26%), loss of function (21%) and wrong signal (15%). Leakage includes internal and external leakage of valves, along with leakage of pump seals, piping and pipe fittings. The loss of function failure mode is at the component level and includes, for example, failure of an instrument to operate or failure of a pump to run. The wrong signal failure mode includes, for example, a position switch indicating a valve is closed when it is actually open, or a pressure transmitter indicating the incorrect pressure.

Figure 5.12 shows that a number of different failure modes can occur in the RHR system. Failure modes classified as "other" include disengaged, engaged, opened, closed, overloaded, ruptured, plugged and excessive noise or vibration. To be able to detect all failures, many different monitoring techniques would be required. For example, visual inspections would be useful for detecting external leakage from components but it may not be able to identify instruments

which are giving wrong indications. A good surveillance and monitoring plan should, therefore, be diverse and include sufficient tests and inspections to cover all the significant failure modes of the important components.

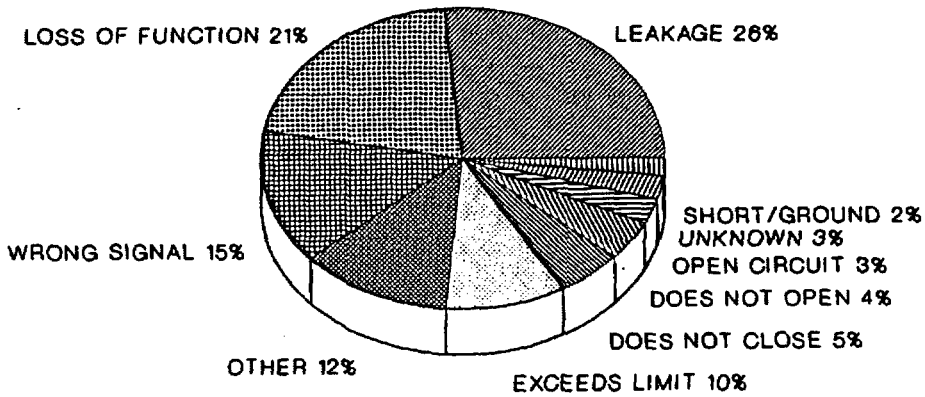


Figure 5.12 RHR Failure Modes

5.2.6 Mechanisms of Failure

A failure mechanism is the physical, chemical or other process by which a component or system degrades or fails. For example, if a heat exchanger is found to have a tube leak because the tube wall is corroded, the failure mechanism would be corrosion. Since the RHR system has standby modes as well as operational modes, a number of different failure mechanisms are expected to be present. This is verified by the data, as shown in Figure 5.13.

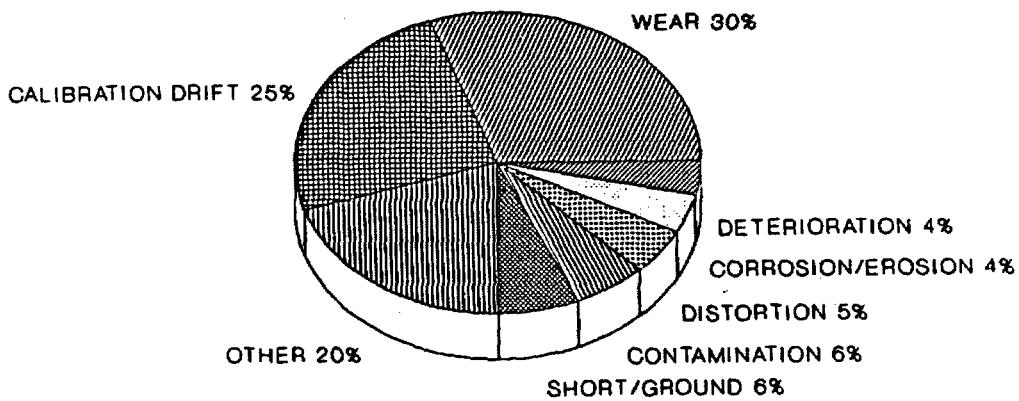


Figure 5.13 RHR Failure Mechanisms

The predominant RHR failure mechanisms are wear (30%) and calibration drift (26%). Wear represents an exposure to stresses encountered during operation, as described in Section 3, which result in some portion of the component being worn away. An example would be a valve packing which is worn down due to friction. Wear is associated with the aging phenomenon. Calibration drift is a mechanism whereby instruments or other calibrated components exceed their set point tolerances. This is expected to be a large contributor to failure for the RHR system since it is heavily instrumented and includes various interlocks.

The failure mechanisms classified as "other" include embrittlement, fatigue, fracture, vibration and those that are unknown. Failure mechanisms in the "contamination" category include events where a foreign material was introduced into the system or component causing a buildup or blockage. Those classified as "deterioration" include failures where a material of construction, for example, insulation or gaskets, is broken down physically by the environment to a point where it can no longer perform its function.

5.3 Component Level Failure Analysis

5.3.1 Predominant Component Failures

The various data sources were reviewed to determine the number of failures attributed to each of the various components in the RHR system. The NPRDS data (Figure 5.14) indicate that valves are the most frequently failed component followed by instrumentation/controls and supports. The dominance of valve failures can be attributed to the large population typically available in RHR systems. In particular, motor-operated valves (MOVs) are the predominant type of valve failing (Figure 5.15). Normalization of the failure data was performed to account for component populations and results are discussed later in this section. It should be noted that MOVs are the subject of ongoing reliability improvement methods[21].

Instrumentation and controls were the second most frequently failed components. This is consistent with previous findings which showed calibration/set point drift to be a significant failure mechanism. A breakdown of the instrumentation failures (Figure 5.16) shows switches to be the most frequently failed type. Piping supports also contributed a significant number of failures.

The fraction of failures related to aging for each component is also shown in Figure 5.14. All components were found to have a high aging fraction indicating that aging degradation is present and contributes to the majority of failures.

Analysis of the LER data (Figure 5.17) and the NPE data (Figure 5.18) gave similar results showing valves to be the component most frequently failed in the RHR system.

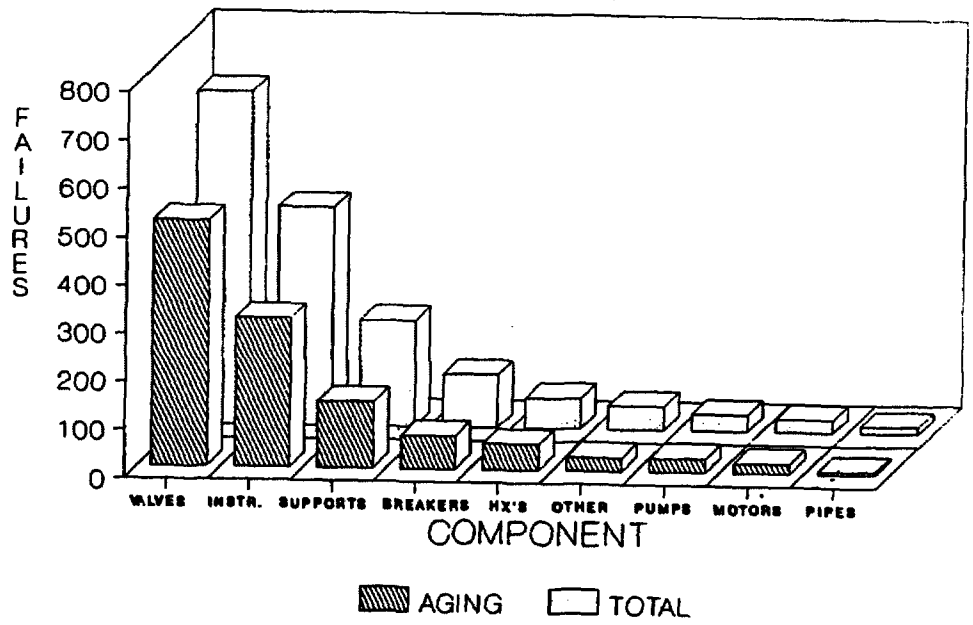


Figure 5.14 Failures Per Component and Aging Fraction - NPRDS

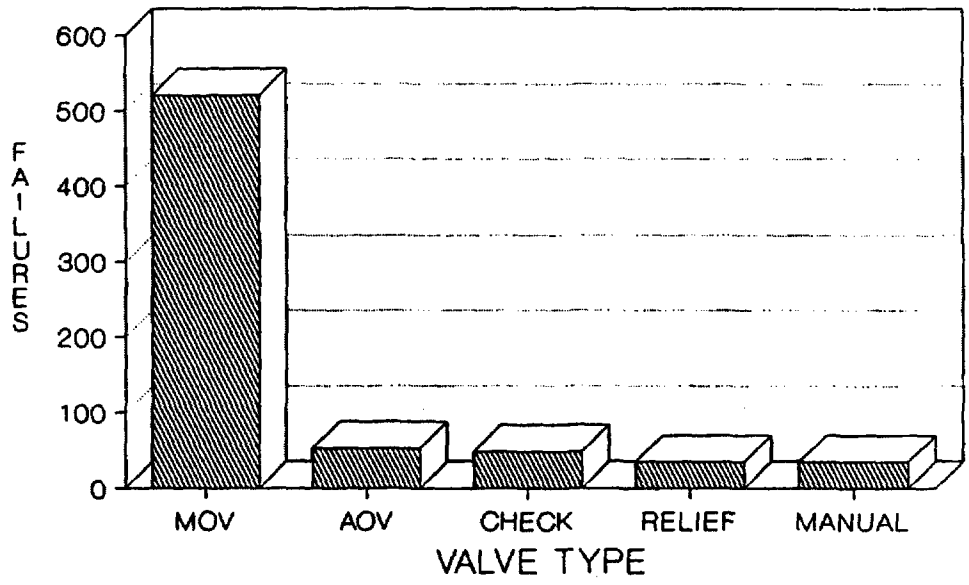


Figure 5.15 Failures Per Valve Type

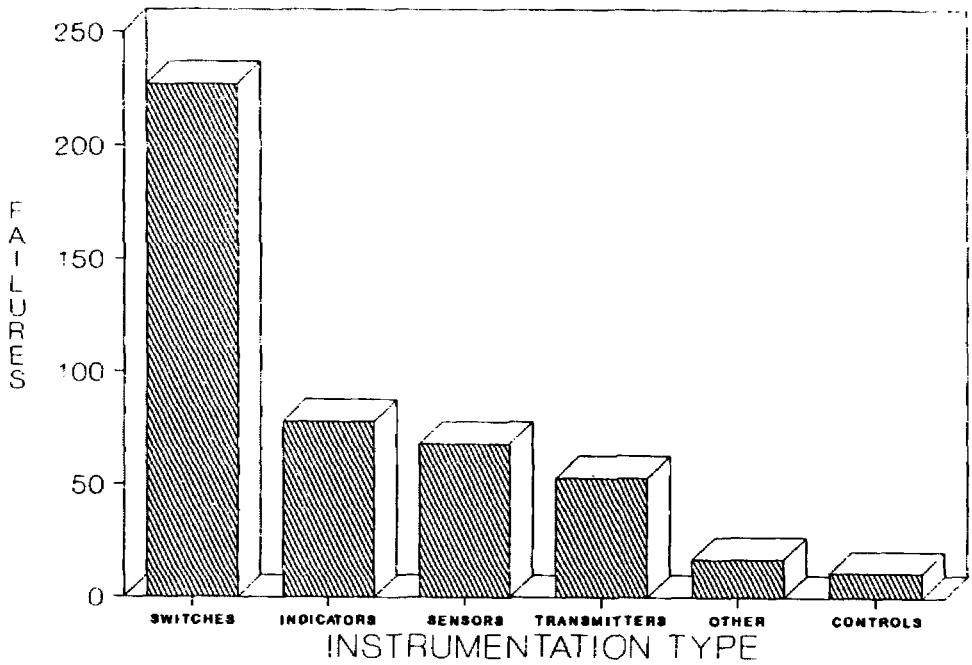


Figure 5.16 Failures Per Instrument Type

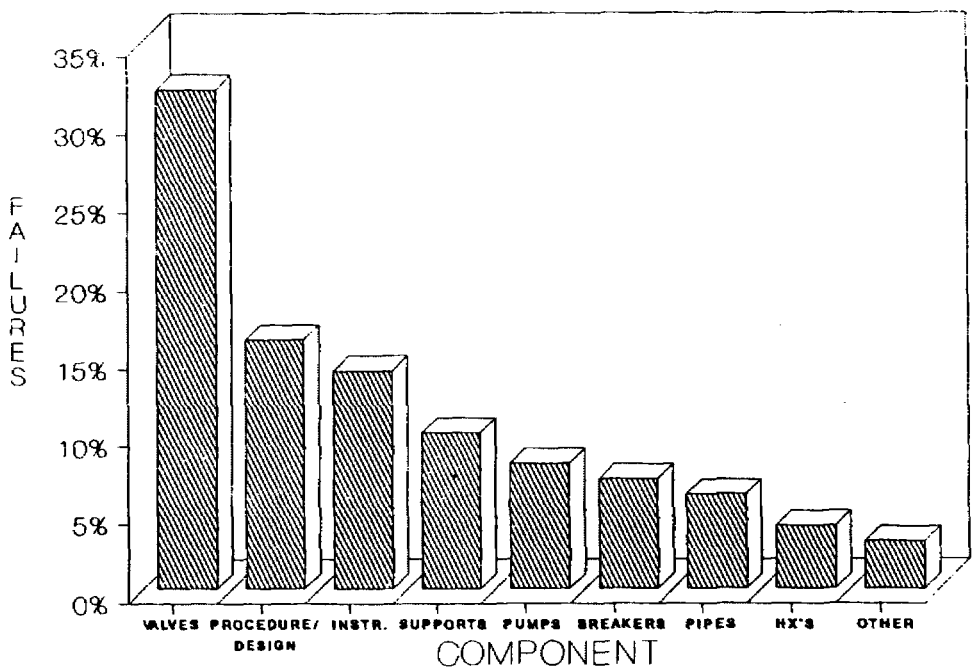


Figure 5.17 Failures Per Component - LER

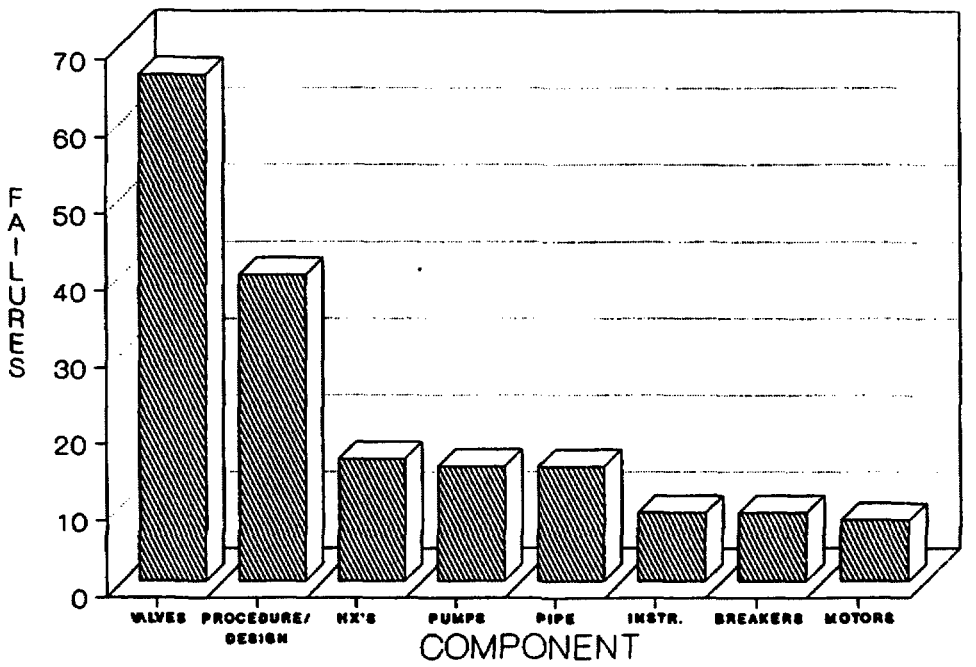


Figure 5.18 Failures Per Component - NPE

The NPE event descriptions for motor-operated valves revealed three areas of concern (Figure 5.19). Ten MOV failures resulted from motor misoperation, including five in which the injection valve motor windings and rotor were found severely burned. In three cases, thermal overload protection was found to be inadequate. One case was attributed to aging based on the ten year age of the motor.

Thirteen events were associated with MOV binding with some due to design problems such as undersized washer spring packs, as well as mechanical problems associated with the motor operator. Loosening of components, such as locknuts and the motor pinion gear, as well as mechanical binding due to drive wear are some of the age-related mechanical problems discussed.

Ten MOV failures involved limit switch or torque switch contact problems which resulted in system degradation. The effect of an isolation valve not closing on demand or an injection valve not opening was noted due to this type of aging problem. These problems are symptomatic of the larger programmatic problem with MOVs in general, which is currently beginning to be addressed by utilities through MOV signature analysis programs. As previously mentioned in Section 4, NRC first addressed these problems in Bulletin 85-03, and is currently processing a generic letter to further improve MOV operability.

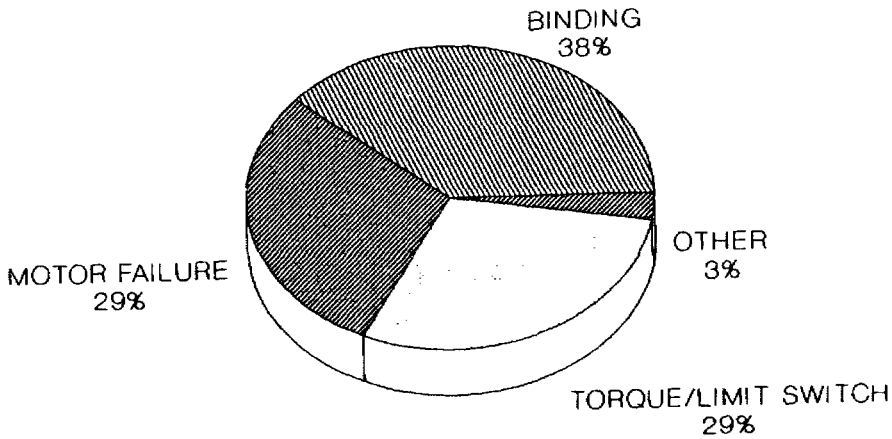


Figure 5.19 MOV Failure Causes - NPE

The second most frequent problem area identified in the LER and NPE data involved procedural or design errors. This differs from the results of the NPRDS data, however, it can be attributed to the data source. The LER data and the NPE data (which are based on LERs) include a large number of programmatic type events dealing with human errors, procedural deficiencies and design problems. These events are required to be reported as LERs, however, they are not reportable to the NPRDS data base. Consequently, events of this nature do not appear in the NPRDS data.

The programmatic events in the NPE data base include a diversity of issues. Of major significance for the NPAR study are the large number (15) of modifications that have been made on the system to preclude recurring failures. These include changes in operating philosophy, as well as hardware modifications. Examples of recommended modifications include adding stiffeners to the heat exchanger loops, a new pump seal design, and a new short shaft pump design. Awareness of these changes is necessary to avoid placing too much emphasis on those age-related failures which have already been corrected.

Other programmatic issues based on operating experience include operator actions, water hammer, vibration, and excessive testing of certain components. A breakdown of the programmatic events from the NPE data is shown in Figure 5.20.

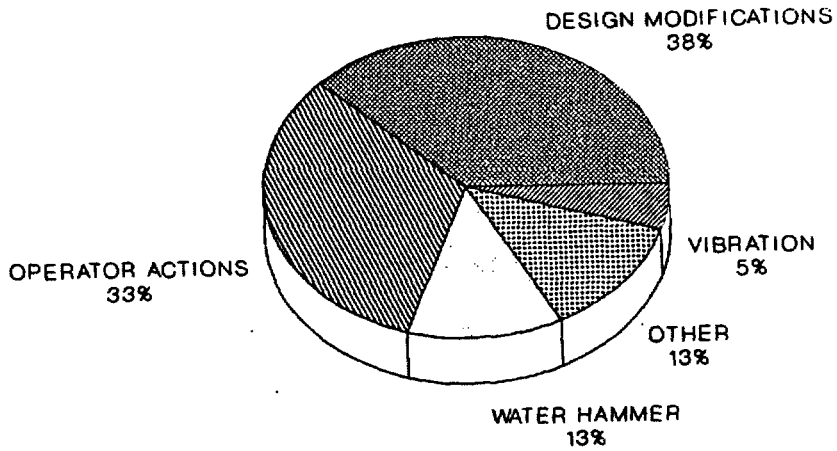


Figure 5.20 Programmatic Events Reported - NPE

Review of the NPE data also showed twenty-seven events recorded which dealt with aging problems associated with the RHR Pump/Motor. Many RHR pumps employ a seal cooling subsystem. Seven problems cited dealt with inadequate flow through the seal cooler heat exchanger which resulted in pump inoperability. One event described an inadequate design in which a common-mode over pressurization failure of all four RHR pump coolers could have occurred under emergency conditions, rendering all pumps inoperable.

Eight events dealt with RHR motor aging-related problems ranging from bearing wearout to winding failures. Of a generic nature in the review of these events were two information notices referenced; IN 86-39, "Failure of RHR Pump Motors and Pump Internal," and IN 87-30, "Cracking of Surge Ring Brackets in Large GE Motors."

Seven events addressed pump seal and wear ring degradation that has occurred. IN 86-39 mentioned above also describes problems with pumps manufactured by Bingham - Williamette which are used for RHR service. It is suspected that intergrannular stress corrosion cracking (IGSCC) had contributed to the wear ring failures.

Sixteen events described in NPE addressed heat exchanger problems including leakage and pluggage which could affect operation of the shutdown cooling system. One report discussed an Alert condition which was declared at one plant due to heat exchanger tube leakage which resulted in an unmonitored radioactive

release. Other age-related failures at another plant led the utility to replace the heat exchanger with a different design. For LPCI operation, the heat exchanger is generally bypassed.

The twenty-six events described dealing with valves (non-MOV) were mechanical in nature and caused by vibration, wear, or material incompatibility. IE Bulletin 80-21 addressing valve body material and IE Information Notice 83-70 describing vibration-induced valve failures are two examples of valve problems experienced which have affected RHR.

Only nine events associated with RHR logic were noted in the NPE data base, with several due to spurious signals or setpoint drift. This area may be addressed more thoroughly in the LER or NPRDS data base. However, it is evident that the effects of instrumentation aging have not been significant.

While piping is typically considered to be a static component in a system analysis, it is investigated here because of problems that have been experienced with reactor recirculation system piping. Piping events reported in the NPE data source included drain line fatigue failures, flange leaks caused by thermal growth, and potential large diameter piping defects. This latter issue was deemed generic and circulated in Information Notice 84-63, "Defective RHR Replacement Piping." A second potentially generic piping concern is related to cracks found in a number of RHR piping welds which appeared to be caused by intergranular stress corrosion cracking.

It should be noted that all of the previous figures showing failures per component represent unnormalized data which shows only the relative frequency of failure for the various components. This information does not indicate the importance of each component nor the significance of its failure. For example, depending on the particular system design and the operating mode it is in, failure of a pump could be much more critical than a valve failure even though pump failures are shown to be less frequent. Component importances are discussed in more detail in Section 7 of this report.

5.3.2 Component Test Frequencies

Since the RHR is a safety-related system, various components within it are required to be tested on a periodic basis. Component testing is used to verify functional ability and readiness for operation and is particularly important for safety-related components. However, testing has disadvantages. It can be very time consuming and costly, depending on the component or system to be tested. Too frequent testing can also lead to premature wearout of components. In addition, more frequent testing increases the potential for human error in not restoring the system or component to its normal status. It is important, therefore, to choose the optimum frequency for component tests.

Using the RHR system failure data, the check-test frequency and functional test frequency of the various components were examined. A check-test is an inspection performed during normal operation of the component to verify the component is operating properly. No special procedures are required for check-tests. A

functional test is one in which the component is taken out of service and operated specifically to verify performance of its design function. It is usually done according to a formal procedure.

Figure 5.21 shows that the majority of the components that failed were either check-tested very frequently (at least once per month) or they were not check-tested at all. In Figure 5.22, the check-test frequency for specific components is shown. All components show the same basic trend, where most failures occur for those components which are frequently tested or not tested at all.

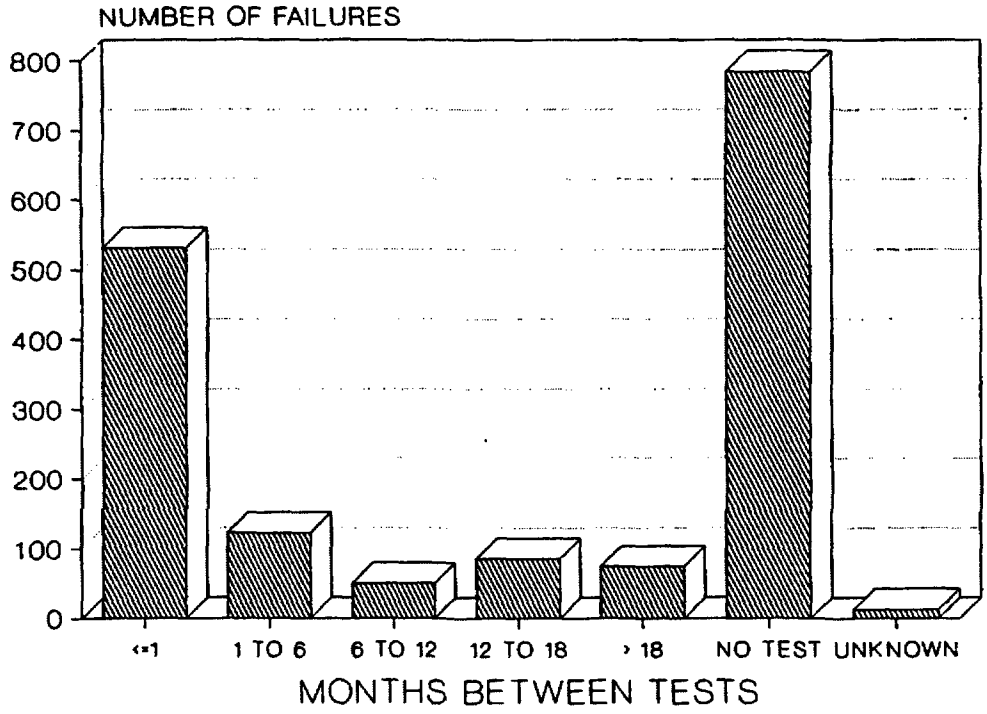


Figure 5.21 Check Test Frequency for RHR Components

To investigate check-test frequency further, the data were sorted to show the distribution of causes of failure and effects on the system for the two predominant frequencies. The resulting distributions were similar to those found previously, namely the predominant failure cause for each test frequency was normal service while the predominant effect was degraded operation. No correlation between check-test frequency and failure cause or effect was seen from these results.

From the check-test frequency results shown, there appears to be no correlation between check testing and component failures. One possible reason is that there is not much uniformity in RHR system monitoring programs between plants. Some plants may check components very frequently while others may not. An additional contributing factor is the diversity of components in the system. It is, therefore, believed that the data presented here reflect the distribution of checking frequencies performed in the plants rather than any correlation with failure rate. To draw any firm conclusions, the data should be normalized with information on which plants perform a particular type of checking.

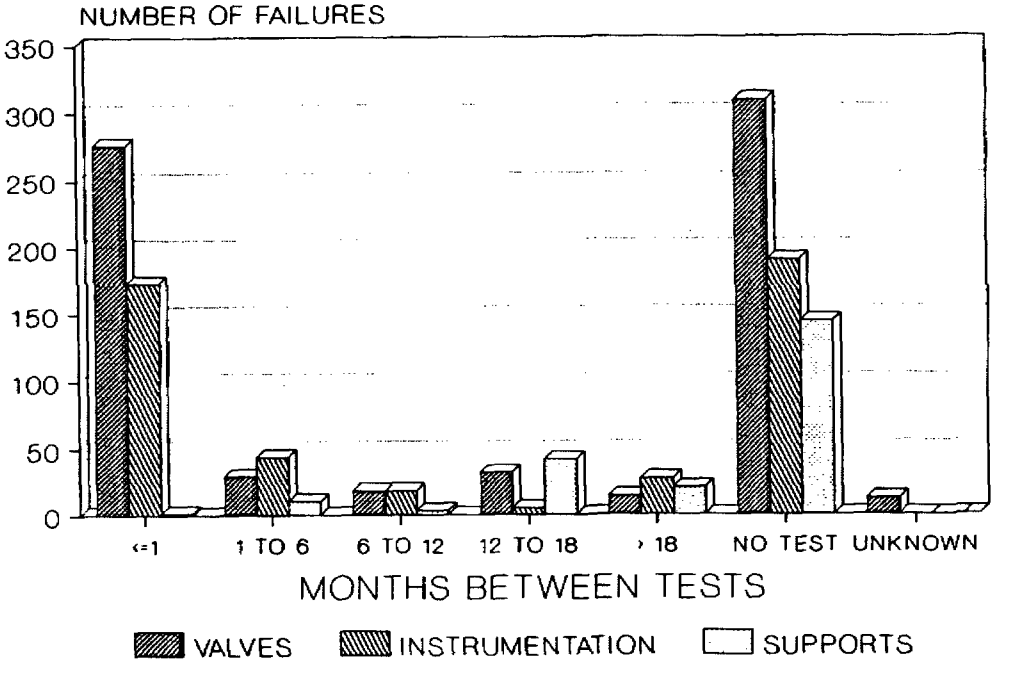


Figure 5.22 Check Test Frequency for Specific Components

Functional test frequency for RHR system components is shown in Figure 5.23. The predominant number of failures occur in components which are frequently functionally tested. The remainder of the failures are fairly evenly distributed among the other functional test frequencies. Functional test frequencies for specific components are shown in Figure 5.24.

As for check-test frequency, the distribution of failures for the functional test frequencies may be related to the distribution of testing frequencies at the plants. Since LPCI is a safety-related operating mode of RHR, it is required to have frequent functional tests. This would account for the large number of components, and thus failures, falling into this category. However, the frequent functional testing of components in the RHR system may be contributing to increased failures due to additional wear. If this is the case, the advantages and disadvantages of frequent testing should be compared to determine if a reduction in test frequency is warranted. This should be evaluated in more detail in the Phase II work.

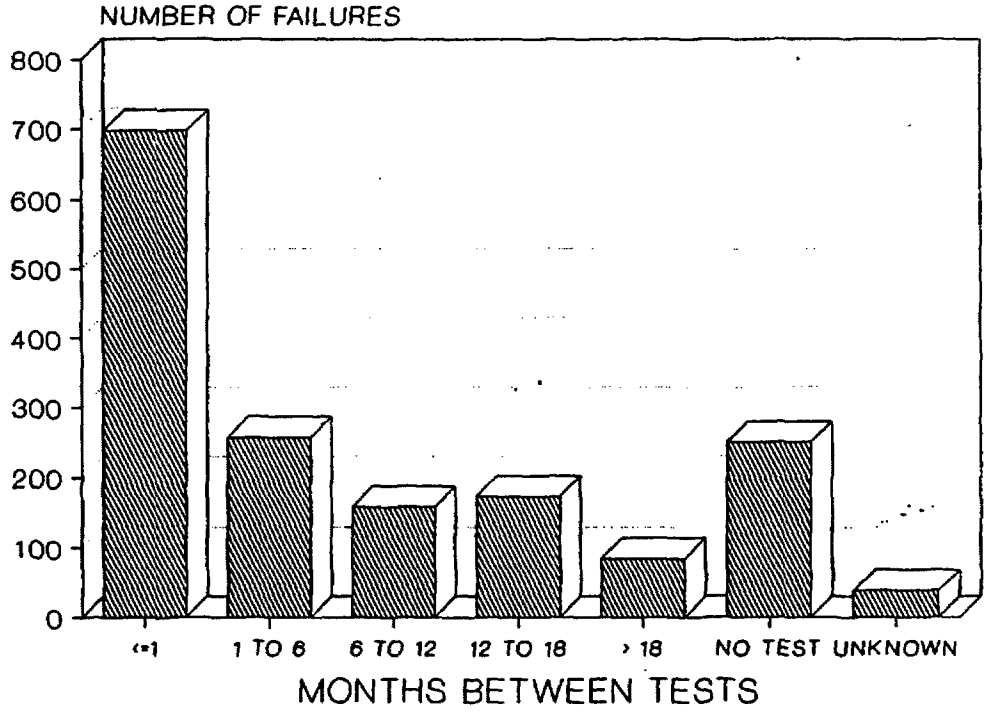


Figure 5.23 Functional Test Frequency for RHR Components

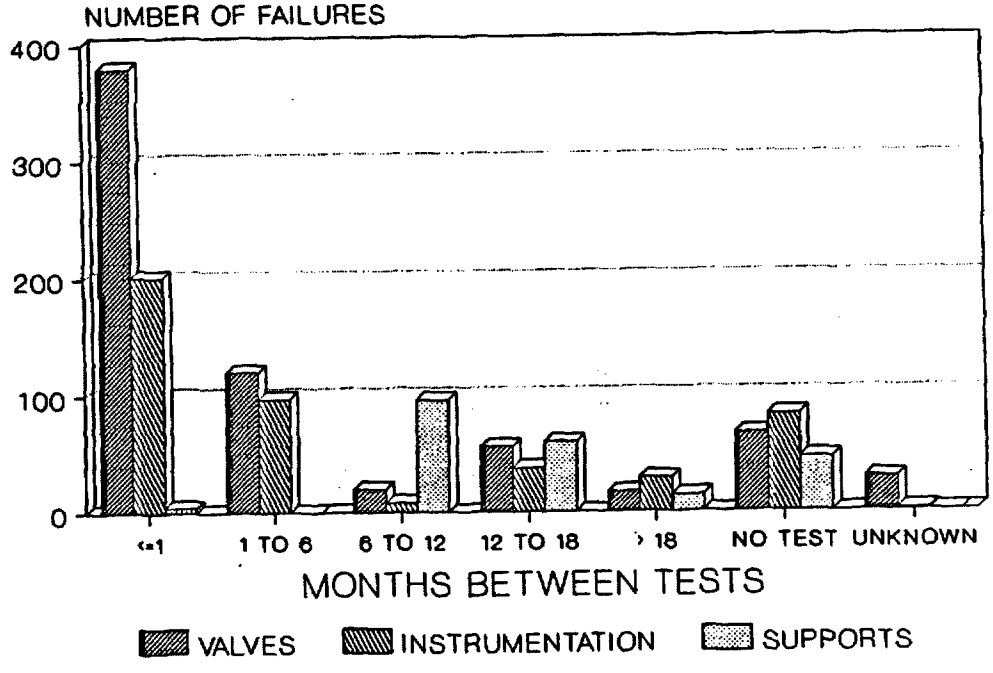


Figure 5.24 Functional Test Frequency for Specific Components

5.4 Normalized Failure Data

5.4.1 Aging Trends

To investigate possible aging trends from past operating experience, the NPRDS failure data for several components were normalized to account for component population. The data were first categorized into age groups for each specific component to determine the number of failures at each age. An estimate was then made of the component population for that age, which accounted for the plants reporting to NPRDS and the plant age during the reporting years (Appendix B). Examples of these sorts are shown in Figures 5.25, 5.26, and 5.27 for MOVs, pumps and heat exchangers, respectively.

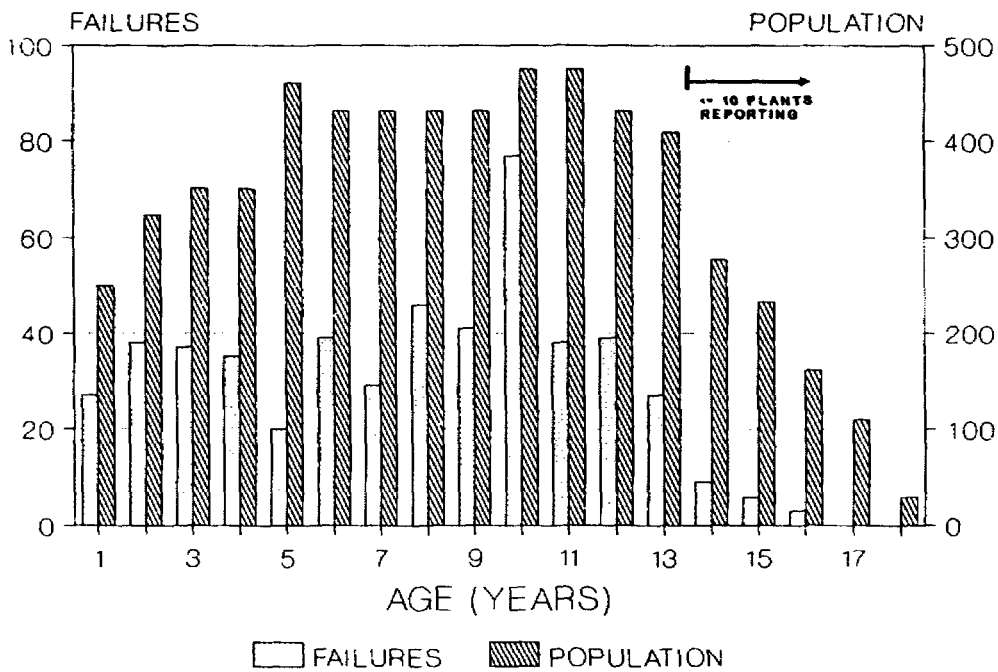


Figure 5.25 MOV Failures and Population Versus Age

As seen from these figures, component population drops off rapidly after age 13, since relatively few plants (less than 10) have been in operation that long. Consequently, there is a corresponding drop in total failures during this time period. For components in this age group, the decrease in total number of failures can be attributed to a decrease in units available to fail. This illustrates the need for normalizing the data to account for the effects of varying population.

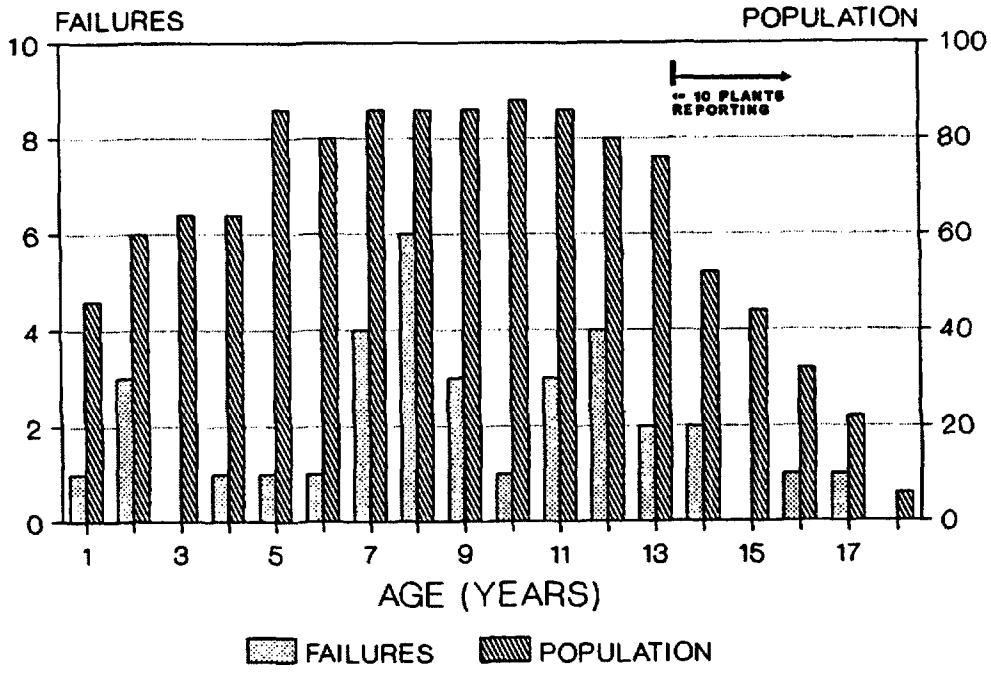


Figure 5.26 Pump Failures and Population Versus Age

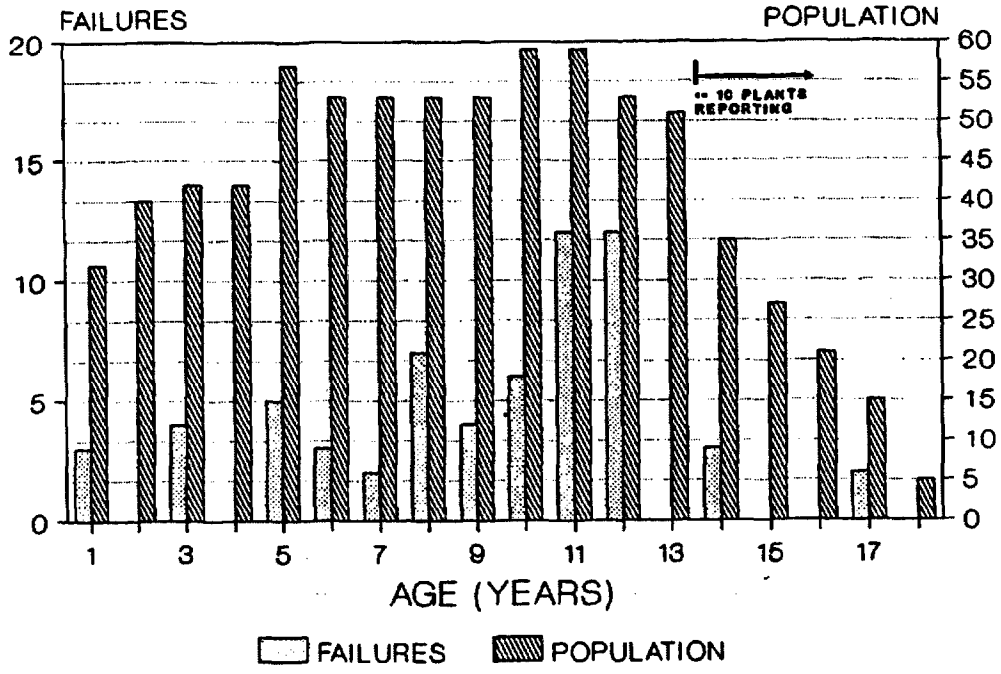


Figure 5.27 HX Failures and Population Versus Age

The data were sorted further to obtain the number of failures and the population as a function of age for each individual plant. The number of failures per component year was then calculated for each age by dividing the number of failures by the component population. Examples of the results for MOV failures is shown in Figure 5.28 for three plants.

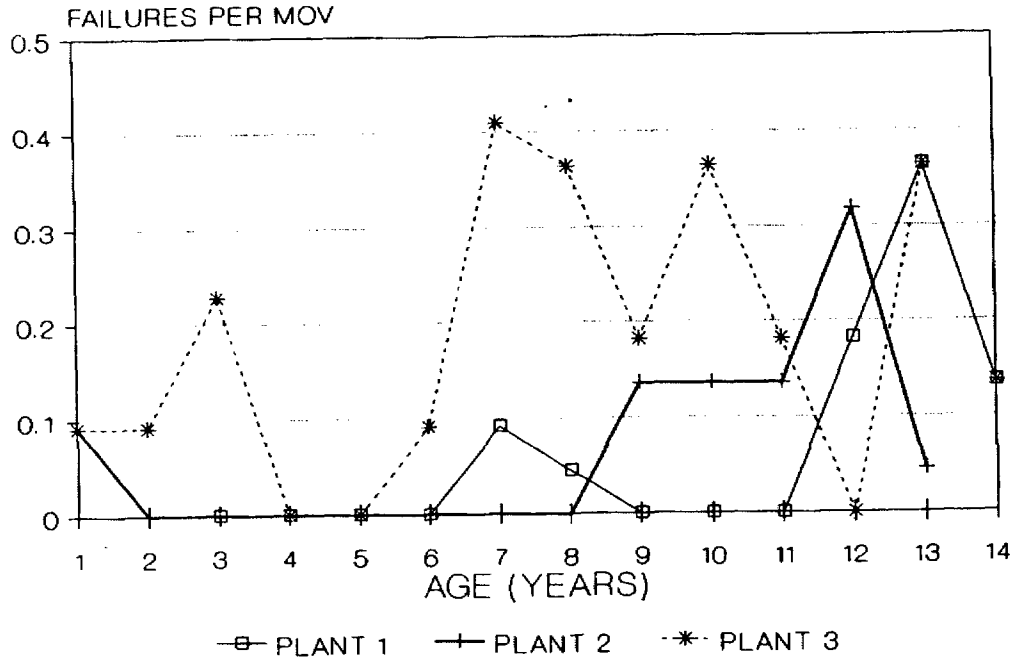


Figure 5.28 Plant Specific Normalized MOV Failures

The individual results for each plant were then compared on a year-by-year basis using a "Generalized Likelihood Ratio" statistical test (Appendix C). The null hypothesis for the test was that all plants have the same failure rate at a particular age. The failure data were compared statistically to determine if this null hypothesis is valid. If it is, the plant specific data can be combined to obtain a generic failure rate for all plants.

A typical set of results for MOVs is shown in Table 5.1. The "Test Statistic" parameter measures how closely the failure rates agree between plants. If each plant had exactly the same failure rate, the test statistic value would be zero. For values greater than the cutoff values shown, the significance of the test statistic is less than 0.05, indicating there is little confidence that failure rates are the same. The large test statistic values shown in Table 5.1 indicate that the null hypothesis tested is not valid, therefore, the data indicate that failure rates are not the same for all plants at a given age.

The fact that component failure rates are found to be different from plant to plant may be due to limitations in the data. However, it may be attributed to other factors including differences in manufacturers, as well as differences in maintenance and monitoring practices between plants. This result is important since it is common practice to use generic failure rates in risk calculations. In many cases it may be the best available information, however, it could impact estimates of risk.

Table 5.1 Statistical Comparison of Plant Failure Rates for MOV's

Null Hypothesis Tested: All plants have the same failure rate at a particular age.

MOV Age	Plants Compared	Test Statistic*	Cutoff Value
1	13	54.0037	21
2	18	47.8036	28
3	20	63.7206	30
4	17	49.3488	26
5	20	35.8872	30
6	19	37.5482	29
7	19	50.2789	29
8	19	47.0285	29
9	18	32.8891	28
10	18	49.6565	28
11	17	27.5836	26
12	17	37.5332	26
13	12	34.2547	20
14	9	12.0900	16
15	8	4.7487	14
16	5	5.4513	10
17	2	0.0000	5

* Test statistic measures how closely failure rates agree. If all rates are exactly the same the test statistic would equal zero. Values greater than cutoff indicate failure rates are not the same.

To investigate the possibility of aging trends, the data analysis proceeded on a plant-by-plant basis. For each component to be examined, the normalized failures for each plant were fitted to a straight line using a least-squares linear regression. An example for MOV's is shown in Figure 5.29 for three plants.

From the results of the linear regressions, the slope of the failure curve for each plant was examined to identify aging trends. Those curves exhibiting a positive slope were categorized as showing an increasing failure rate with age. Those curves showing a negative or zero slope were categorized as showing no aging effect on failure rate. Results of the regressions are presented in Table 5.2.

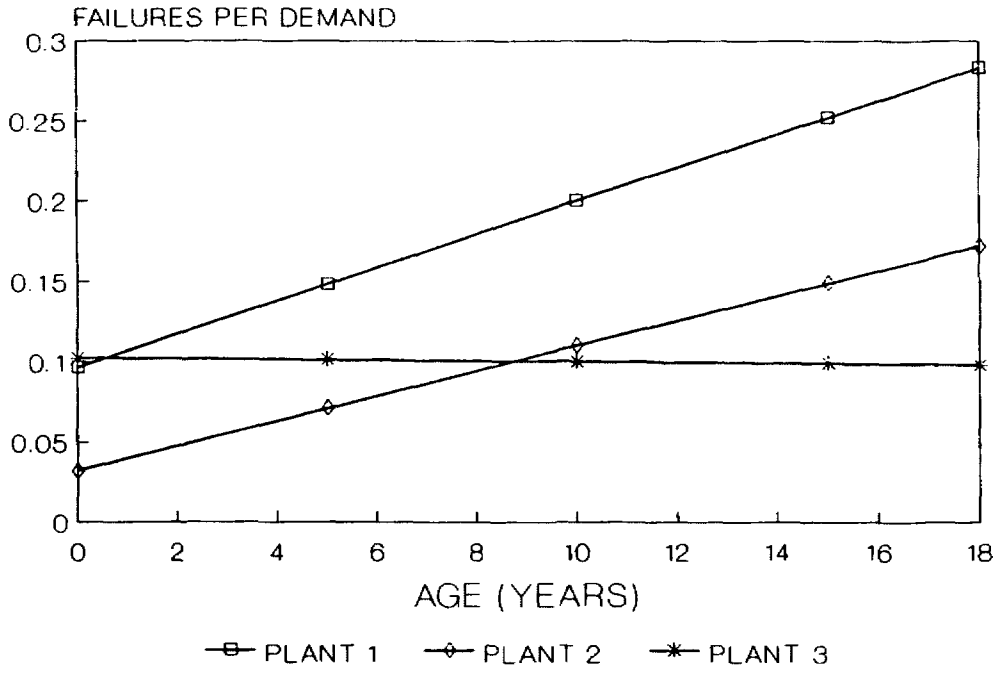


Figure 5.29 Linear Regression on Plant Specific Normalized MOV Failures

Table 5.2 Aging Impact on Component Failure Rates

	Number of Plants			
Component	Failure Rate Increasing	Failure Rate Decreasing	Failure Rate Constant	No Failures Reported*
MOV	15	11	1	4
PUMP	8	4	1	18
HX	12	6	1	12
PRESS. SWITCH	13	7	0	11
PRESS. SENSOR	8	8	0	15
LEVEL SWITCH	2	2	0	27
LEVEL SENSOR	3	5	0	23

* Plants did report failures for other components.

Comparison of the regressions did not provide any firm conclusions regarding failure rate trends. In most cases, the number of plants with increasing failure rates exceeded the number with decreasing failure rates indicating the potential for an overall increasing trend. However, the number of plants with no failures reported was also relatively high. This could be due to poor reporting for that particular component or because there actually were no failures for the components. The results do show, however, the wide variations in failure rates that exist between plants.

5.4.2 Time-Dependent Failure Rates

As previously discussed, plants may not have the same component failure rate. However, for purposes of comparison with commonly used values, time-dependent failure rate curves were generated from the data. These curves were calculated using data from all plants that reported to NPRDS for a minimum of 5 years. Data from plants with less than 5 years reporting time were not used since these data could be influenced by short term variations in plant performance which would bias the results. The data were used together with estimates of yearly operating hours or yearly demands to calculate failure rates as a function of age. It should be recognized that results from this type of pooling are probably strongly dependent upon the plants used in the sample.

To generate the time-dependent curves, an average failure rate was first calculated for all plants. This was used as the y-axis intercept for each curve. Next, linear regressions were performed for each plant. The resulting slopes were then averaged to obtain a nominal slope. The intercept and slope thus obtained were used to construct the time-dependent failure rate curves. Minimum and maximum bounds were found by calculating the standard deviation for the average intercept and average slope. A standard error was then calculated for each by dividing the standard deviation by the square-root of N-1, where N is the number of points averaged.

The time-dependent failure rate curve for MOVs is shown in Figure 5.30. The mode is failure-to-transfer. As shown, good agreement with other commonly used sources, such as WASH-1400, was found for the initial zero age failure rate. However, it should be recognized that the failure rates calculated from the NPRDS data are probably lower than actual since not all failures are reported to the data base. It should also be recognized that this failure rate is based on any reported failure of an MOV to transfer on demand, and does not account for specific operating conditions. Under certain operating conditions the failure rate could be significantly higher, as reported in NUREG/CR-5140[20] for MOVs operating under high differential pressures. Currently, efforts are underway (through Bulletin 85-03) to address this issue.

Figure 5.30 also shows that the MOV failure rate increases with age. The slope corresponds to an increase of 11% per year. This could result in a factor of four increase in failure rate over a 40 year life which could lead to an increase in system unavailability and possibly an increase in plant risk. This is evaluated in Section 7 of this report.

The time-dependent failure rate curves for pump fail-to-run and heat exchanger leakage are presented in Figures 5.31 and 5.32. As for MOVs, the failure rate is seen to increase with age for both these components. The pump increase corresponds to 17% per year while the heat exchanger failure rate increases by 8% per year. By way of comparison, previous work on the component cooling water system found failure rates increasing by as much as 30% per year. The increases seen here for mechanical RHR components is, therefore, considered to be moderate.

The electrical components examined in the RHR system included switches and sensors. Figures 5.33 and 5.34 show the time-dependent failure rates calculated for pressure switches and pressure sensors, respectively. As shown, the pressure switch failure rate curve showed only a slight increase with age (3% per year). The pressure sensor curve showed no increase with age. The failure rates for level switches and level sensors both showed no increase with age, as shown in Figures 5.35 and 5.36, respectively. The electrical RHR components examined were, therefore, judged to show little or no impact due to aging.

For those curves showing an increase with age, the slope of the line represents the aging acceleration rate. As shown on Figures 5.30 and 5.31, initial values are in good agreement with other commonly used failure rates. However, after 20 years the failure rate may increase to several times its initial value. This could lead to an increase in system unavailability and possibly an increase in plant risk. The effect of these increasing failure rates was examined and is discussed in Section 7.

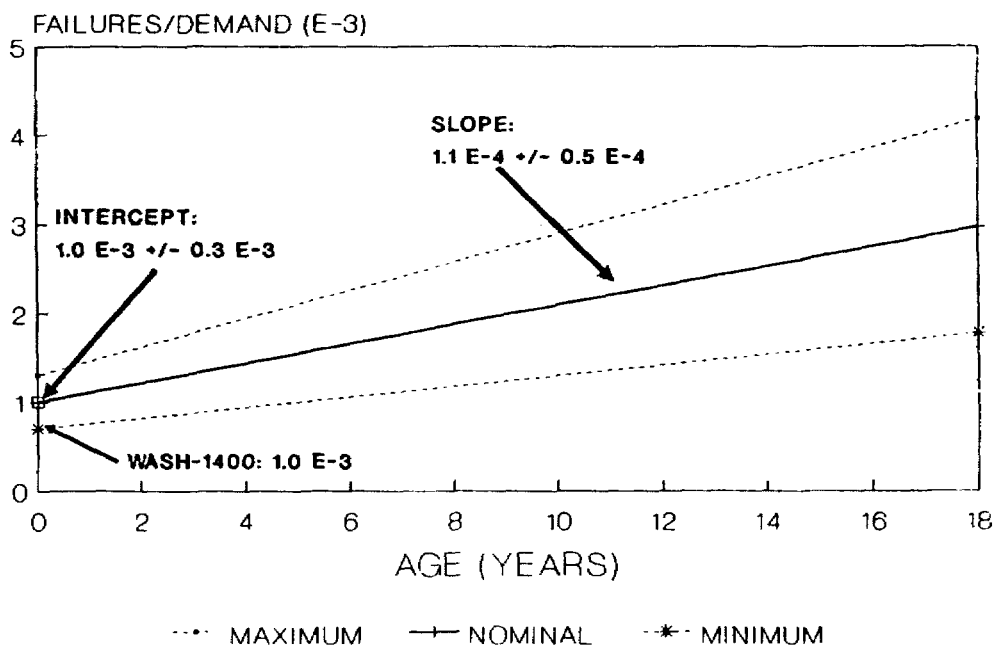


Figure 5.30 MOV Failure to Transfer Failure Rate Versus Age

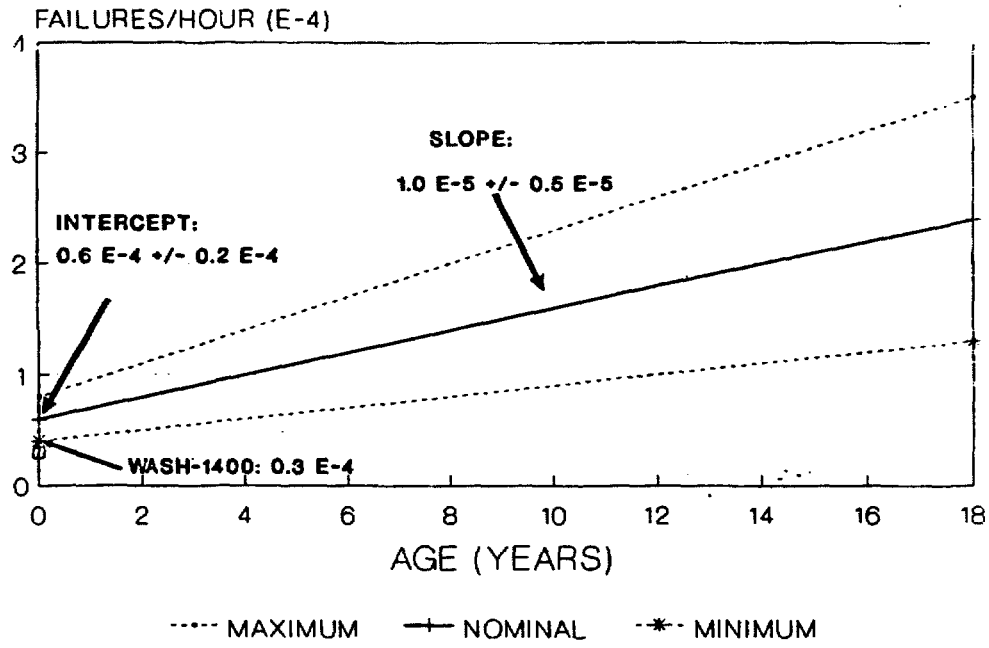


Figure 5.31 Pump Fail-to-Run Failure Rate Versus Age

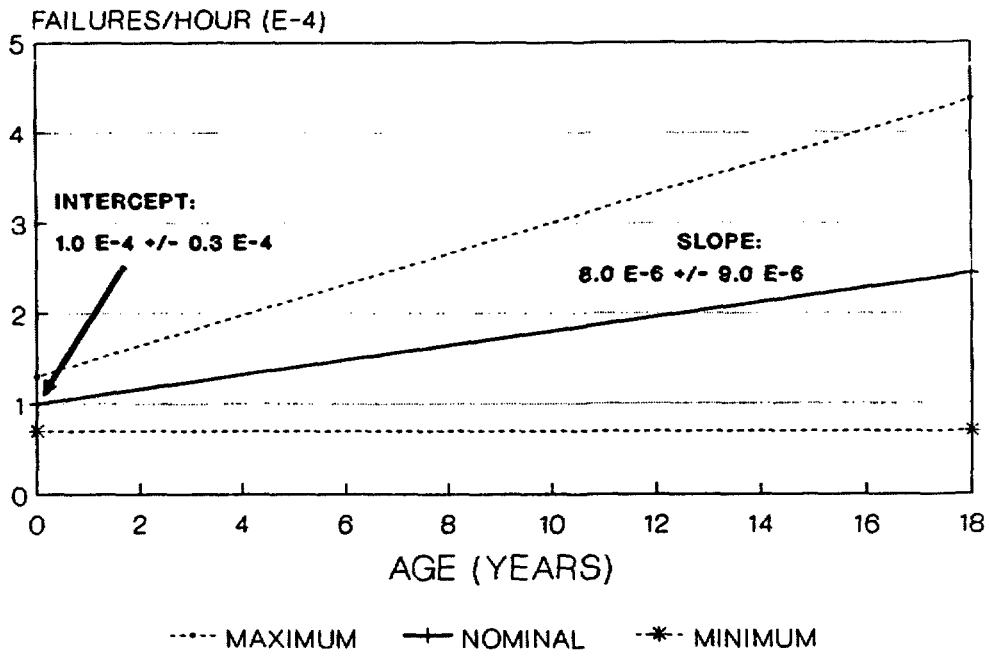


Figure 5.32 Heat Exchanger Leakage Failure Rate Versus Age

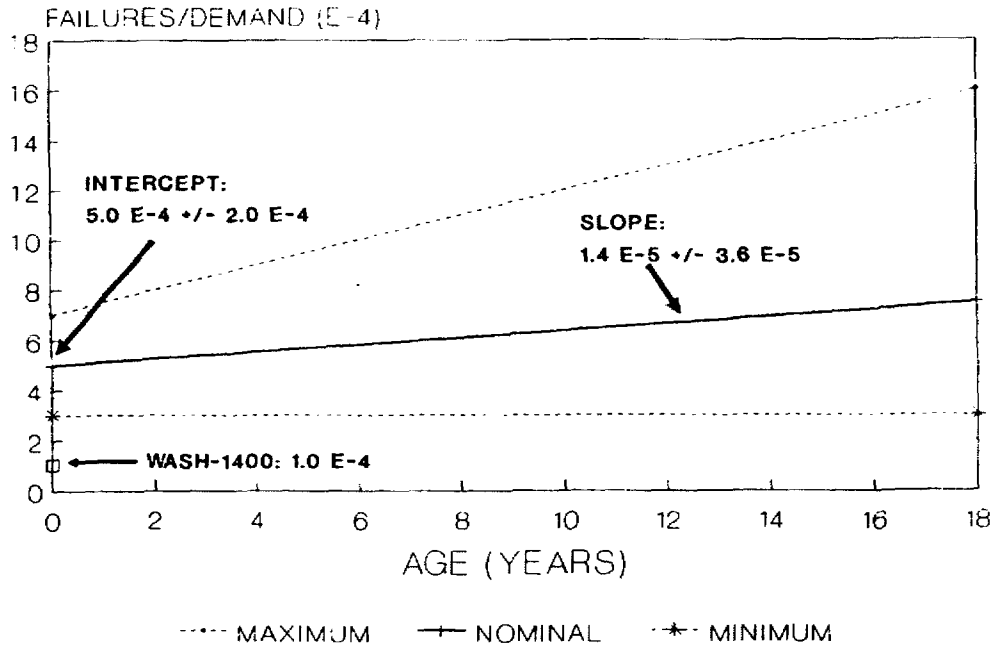


Figure 5.33 Pressure Switch Loss of Function Failure Rate Versus Age

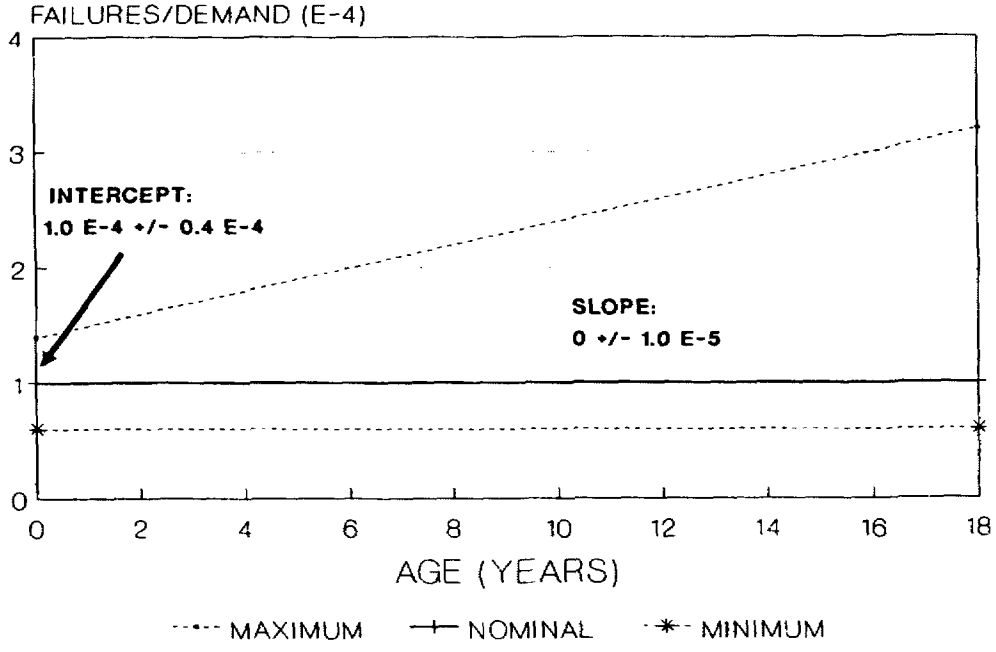


Figure 5.34 Pressure Sensor Loss of Function Failure Rate Versus Age

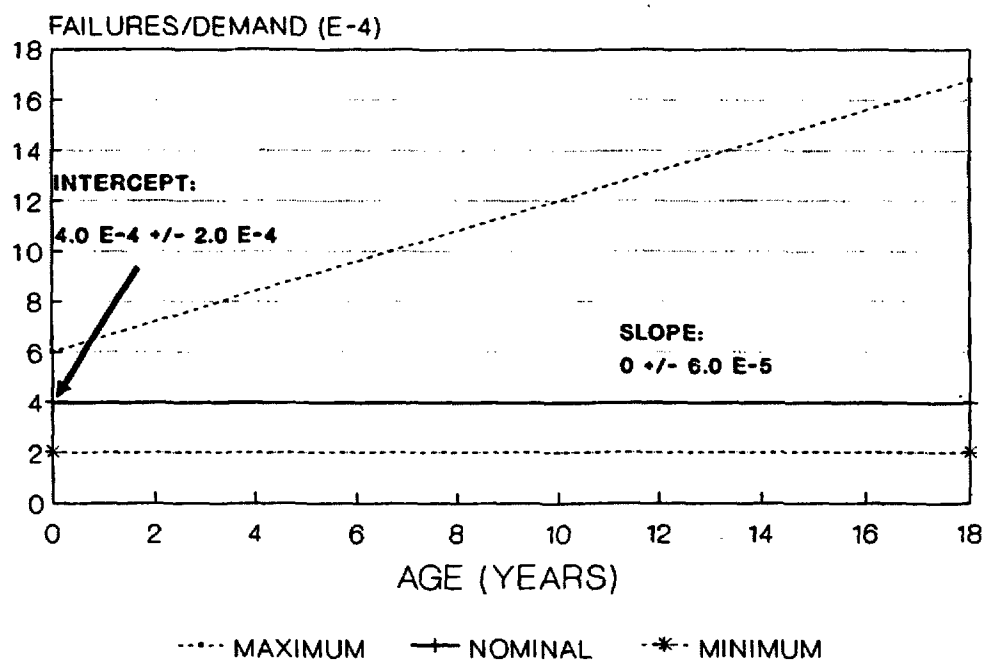


Figure 5.35 Level Switch Loss of Function Failure Rate Versus Age

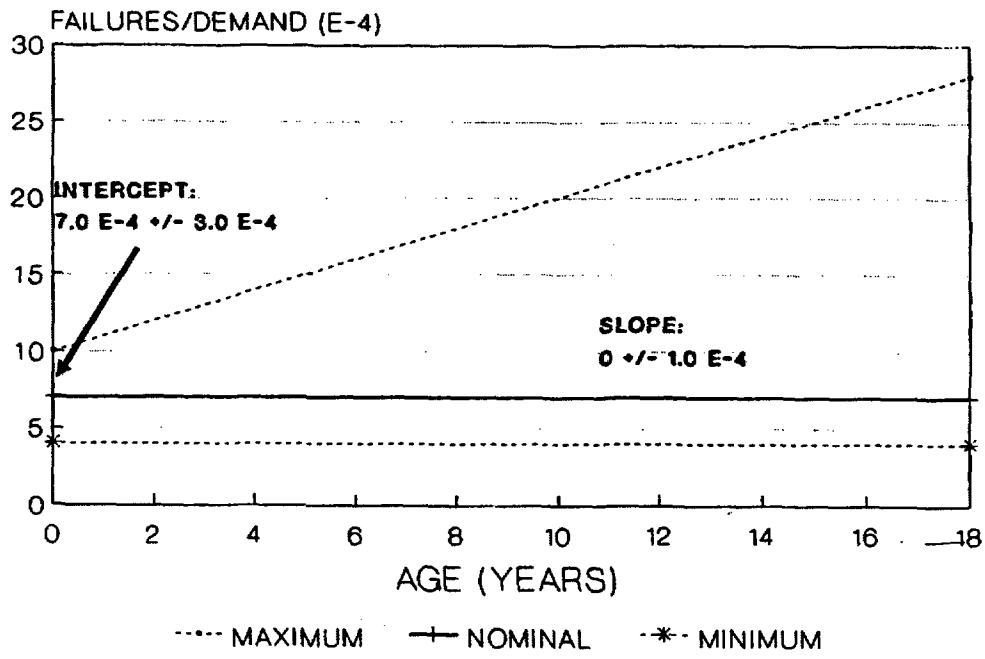


Figure 5.36 Level Sensor Loss of Function Failure Rate Versus Age

It should be recognized that the method used to generate the failure rate curves presented here includes various assumptions. In grouping all the failure data from various plants, it is assumed that the individual plant failure rates are part of the same normal distribution. From the statistical analysis discussed previously, it is seen that this may not always be correct. The data indicate that failure rates may be different between plants, possibly due to differences in maintenance, monitoring and inspection practices. The results presented here are industry averages which are intended for purposes of comparison with existing generic failure rates and for identifying time-dependent trends caused by aging. Use of these failure rates for other applications should be performed with caution.

Since RHR is predominantly a standby system, the impact of aging on the various components may be affected by their relatively low operating hours. To examine this possibility, a comparison of aging acceleration rates was made between two RHR components and components from Component Cooling Water (CCW) systems [14]. Since CCW is a continuously operating system its components accumulate significantly more operating time than RHR components. Results of the comparison are shown in Table 5.3

Table 5.3 Aging Acceleration Rate Comparison for Continuously Operating and Standby Systems

	Aging Acceleration Rate (Failures/hr-yr)	
Component	RHR	CCW
PUMP	1×10^{-5}	3×10^{-5}
HX	8×10^{-6}	6×10^{-7}

Pumps in a standby system show a lower aging acceleration rate than pumps in a continuously operating system. This is expected since the predominant failure mechanism for pumps was found to be wear, which is directly proportional to operating time. For heat exchangers, which are passive components, the predominant failure mechanism is corrosion.

The aging acceleration rate for heat exchangers was found to be higher in the standby system. This can be partially attributed to inaccuracies in the data, however, it may also be due to standby operating characteristics. Some aging mechanisms may be more active under standby conditions; for example localized corrosion processes leading to leaks in a heat exchanger are more likely to occur when stagnant water remains in the unit over a long period of time than when water is continuously flowing through the system.

5.5 Summary of Data Analysis Findings

Review and analysis of past RHR operating experience and failure data has shown that aging degradation does occur in the RHR system and does lead to failures. An aging fraction of 69% was found from the NPRDS failure data, which is typical of the percentage of failures related to aging found in other sources.

The data show that 65% of the failures are detected by current tests and inspections. However, approximately 27% of the failures are not detected until some operational abnormality occurs. This indicates that improvements to current methods may be possible.

The dominant cause of failure was found to be normal service, while the dominant failure mechanisms were wear and calibration drift. Wear was associated mainly with mechanical components while calibration drift involved electrical components. The predominant failure mode was leakage followed by loss of function and wrong signal. These results were consistent among the various data bases.

In evaluating the effect of failure on RHR performance it was found that over 50% resulted in degraded system operation, while approximately 20% resulted in a loss of redundancy. Loss of redundancy is significant since it increases the probability that the system can become unavailable. Other significant effects of RHR failures include loss of shutdown cooling capability, radiological releases, reactor scrams and actuation of engineered safety features.

At the component level it was found that valves were the component failing most frequently followed by instrumentation/controls, and supports. The predominant type of valve failing was MOVs while the predominant type of instrumentation/control was switches. All component failure were found to have a large aging fraction; typically 60% to 80%.

The failure data were used to calculate time-dependent failure rates for several RHR components. It was found that mechanical components showed a low to medium increase in failure rate with age, with the increases ranging from 8% to 17% per year. Electrical components (switches and sensors) showed little or no increase in failure rate with age, with increases typically 0% to 3% per year.

6. PLANT SPECIFIC FAILURE DATA ANALYSIS

6.1 Background

This section of the report discusses results from the review of plant specific data from Millstone Unit 1. This information was reviewed to identify aging trends and problems encountered in the plant for comparison with the data base results discussed previously. The purpose of this review of plant specific data is to validate the data base findings and identify any deficiencies in the data bases. Millstone 1 was chosen since it is a BWR and has a computerized maintenance record keeping system which allows data to be easily retrieved and ensures consistency and completeness.

6.2 Millstone RHR Data Analysis

6.2.1 System Description

The Millstone Unit 1 RHR system is composed of two separate systems; the LPCI/Containment Cooling system and the Reactor Shutdown Cooling System.

The LPCI/Containment Cooling System has two loops with two pumps and one heat exchanger per loop (Figure 6.1). There is a cross-connect line between the loops with an MOV used for isolation. The heat exchangers are used for containment cooling and are bypassed for LPCI operation. The system has a total of 23 motor operated valves and 2 air operated valves. Stop-check valves are also used in the pump discharge lines to act both as check valves and as stop valves. For the portion of the system inside containment, check valves are equipped with operators to allow exercising and testing during normal plant operation. The system also includes various gate and butterfly valves, which are used for maintenance purposes and are normally locked open.

The LPCI/Containment Cooling system is heavily instrumented. Pressure is measured at the inlet and discharge of each pump, while flow is measured in both of the injection lines. Numerous alarms and interlocks are used to activate and monitor LPCI operation, as well as to restrict use of the containment cooling function of the system. Electrical power for the system is available from the normal auxiliary power supply, an emergency diesel generator, or a gas turbine generator.

Testing of the LPCI system includes pre-operational testing and periodic surveillance testing. A pre-operational test is performed prior to plant startup to assure proper functioning and operation of all instrumentation, pumps, heat exchangers and valves. In addition, reference pressures, temperatures and flows are recorded for future use as base points for tests performed during plant operation. Once the plant has become operational, periodic tests are performed to demonstrate operability of pumps, valves, and control circuits.

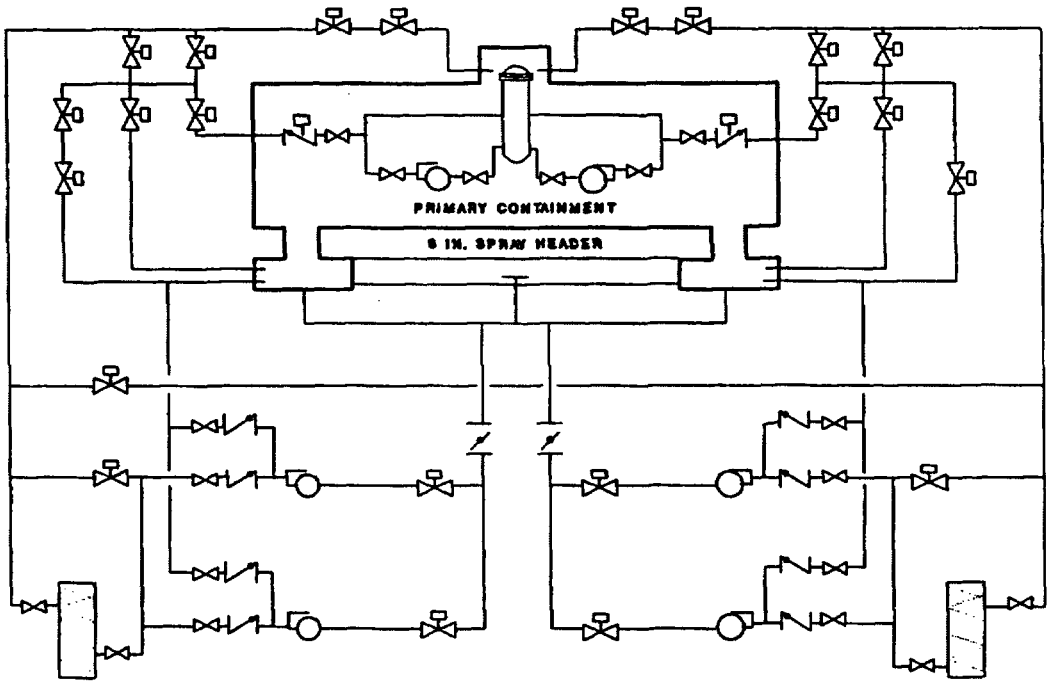


Figure 6.1 Millstone 1 LPCI/Containment Cooling System

A summary of the LPCI system configuration is shown in Table 6.1.

Table 6.1 Summary of Millstone 1 LPCI System Design.

Number of Pumps	Design Coolant Flow	Pressure Range	Electrical Power
4(33%)	7500 gpm @ 165 psi	0 to 235 psi	Normal Aux. Power or Emerg.
	15,000 gpm @ 0 psi		Diesel Gen. or Gas Turbine Gen.

The shutdown cooling system includes two loops with one pump and one heat exchanger per loop (Figure 6.2). The system has a total of six motor operated valves along with check valves at each pump discharge. The pumps are horizontal, centrifugal pumps with mechanical seals and have a design flow of 2900 gpm. The heat exchangers are U-tube type and are designed for 1250 psig and 350°F on the tube side. Cooling water is supplied to the heat exchangers from the reactor building closed cooling water system.

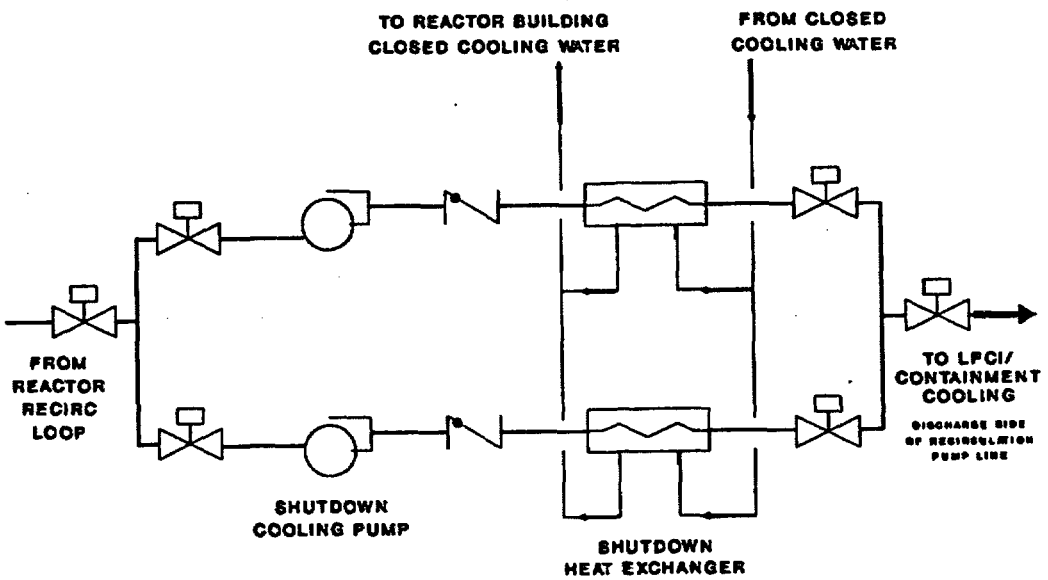


Figure 6.2 Millstone 1 Shutdown Cooling System

Instrumentation is provided to measure pump inlet temperature and pressure, as well as pump discharge pressure and flow. In addition, switches and interlocks are provided to restrict pump and isolation valve operation.

6.2.2 Failure Characteristics

Data from Millstone 1 were obtained from a recently instituted computerized maintenance monitoring system. This system includes records of all maintenance activities from 1984. Each record identifies the component and the work performed, and categorizes the maintenance as preventive, corrective or other type. For this analysis, all corrective maintenance records were retrieved and reviewed for the LPCI/Containment Cooling and shutdown cooling systems. Data for the period January, 1984 to May, 1988 were obtained. This corresponds to a plant

age of 14 to 18 years old, with the fourteenth and eighteenth years only partially represented. During this period a total of 139 corrective maintenance items were recorded for the LPCI system and 21 for SDC.

As was done for the national data base records, each Millstone record was reviewed to determine if the failure was aging-related and to identify the aging characteristics. As shown in Figure 6.3, a high aging fraction was found for both the LPCI and SDC events, which is consistent with the data base findings.

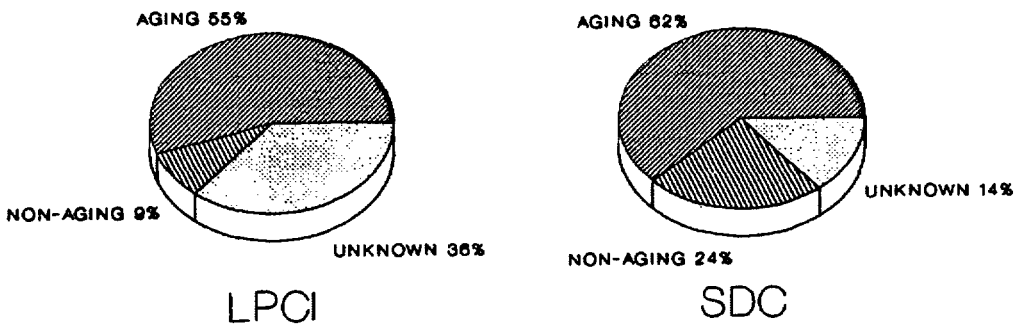


Figure 6.3 Aging Fraction for Millstone 1 RHR Data

The corrective maintenance items recorded for the LPCI and SDC systems typically involved adjusting valve limit switches to allow complete opening or closing, repacking of valves, or calibration of instruments. Several items involved the retightening of bolts on valve operators and heat exchanger waterbox covers. Many involved normal inspections of equipment which resulted in the detection and replacement of worn parts; for example, several records were found where valve operators were inspected and new switch box cover gaskets were installed as a result. The majority of these events resulted in degraded operation of the system, as shown in Figure 6.4. However, many of the events were minor and had no effect on system performance.

Each of the Millstone records were categorized as to the cause of failure. The predominant cause of failure was identified as normal service (Figure 6.5). This is consistent with findings from the national data base.

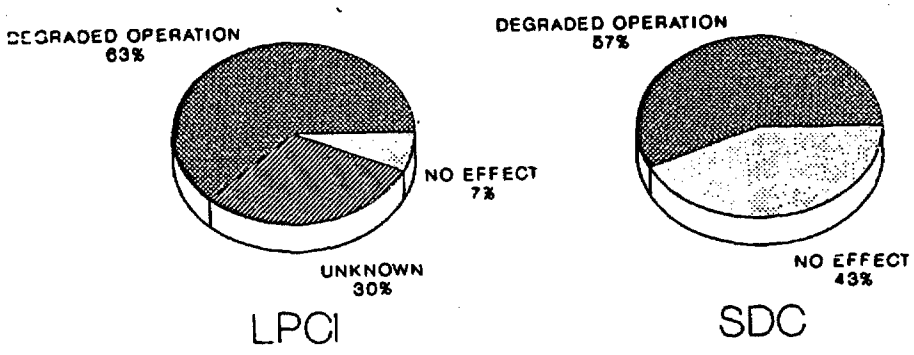


Figure 6.4 Failure Effect - Millstone Data

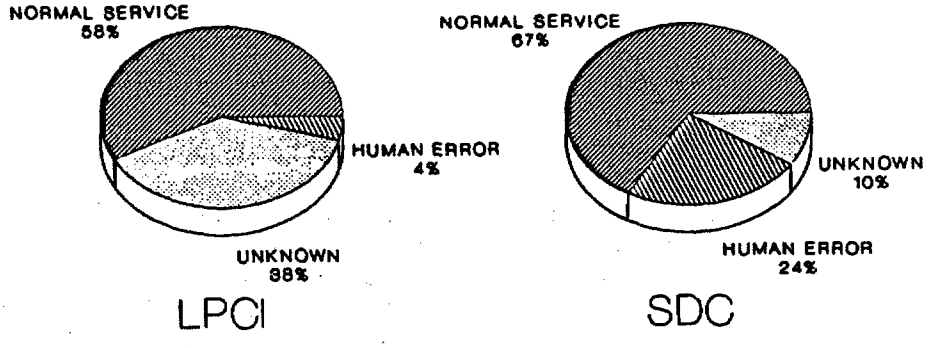


Figure 6.5 Failure Cause-Millstone Data

6.2.3 Component Failures

The components most frequently failed were identified and are shown in Figure 6.6. Again, results were consistent with date base findings. Valves were the most frequently failed component followed by instrumentation. This is expected since these results are not normalized and these components have the highest population, however, the large aging fraction discussed previously indicates that many of these failures are due to aging degradation.

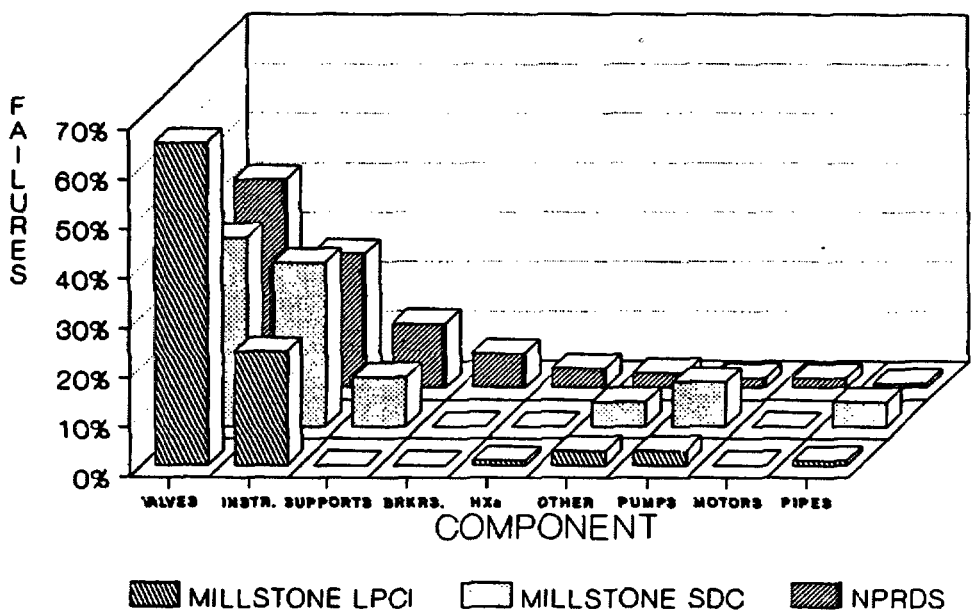


Figure 6.6 Component Failures - Millstone Data

To compare the plant specific data with the data base findings, the MOV data were normalized to account for population and a failure rate was calculated. Since complete data were not available for ages 14 and 18, only data for ages 15, 16 and 17 were used. The total number of failures-to-transfer for these three years were used together with an estimate of MOV demands (Appendix B) to calculate a failure rate for MOV's. The failure rate was calculated to be 5×10^{-3} failures/demand. The result is shown in Figure 6.7 together with the failure rate calculated from the NPRDS data.

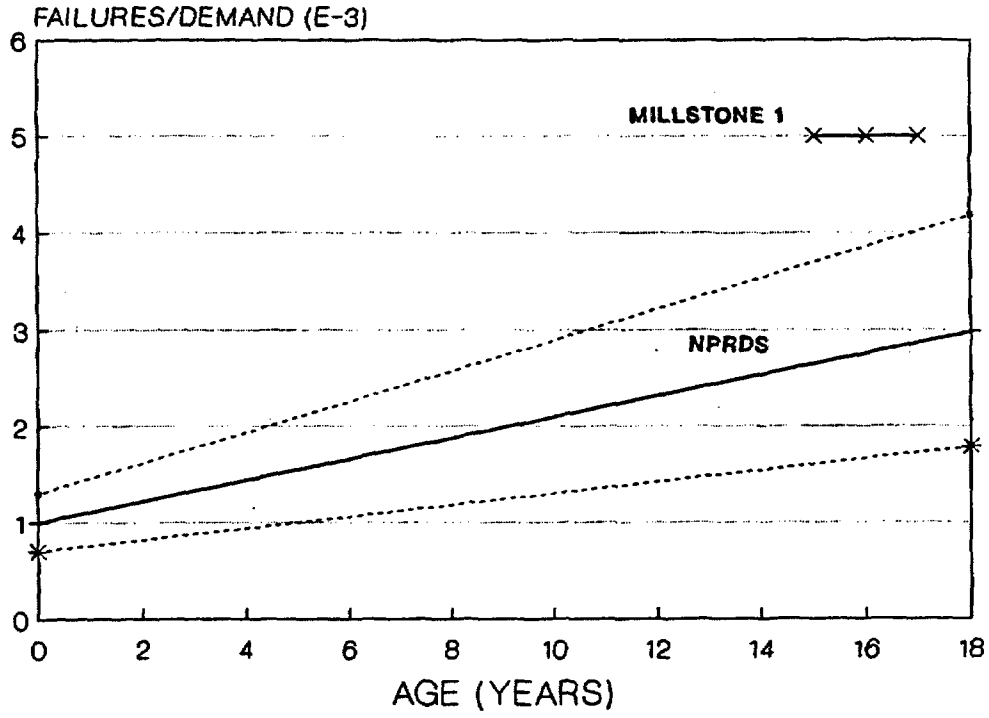


Figure 6.7 MOV Failure-to-Transfer Failure Rate for Millstone 1

Comparison of the MOV failure rates shows relatively good (order of magnitude) agreement between the plant specific data and the generic curve generated from the data base reports. The plant data indicates a slightly higher failure rate in this case. This can be attributed to incomplete reporting of events to NPRDS which results in under estimates of failure rates. This illustrates the differences that could exit between plant specific data and generic failure rates.

Failure rates were also calculated from the Millstone data for heat exchangers and pressure switches. During the three year period for which data were available, one heat exchanger failure occurred. This involved a leak in the waterbox cover which was repaired by tightening the head bolts. The failure rate calculated for this event, assuming the same amount of operating time used in the failure rates calculated from NPRDS data, is 2.0×10^{-4} failures/hr (Figure 6.8). Two pressure switch failures were found in the Millstone data for this time period. All involved a loss of function of the instrument requiring repair or replacement. A failure rate of 1×10^{-3} failures/demand was calculated for the pressure switches (Figure 6.9). As shown in Figures 6.8 and 6.9, good agreement was found between the Millstone failure rates and those calculated from the NPRDS data.

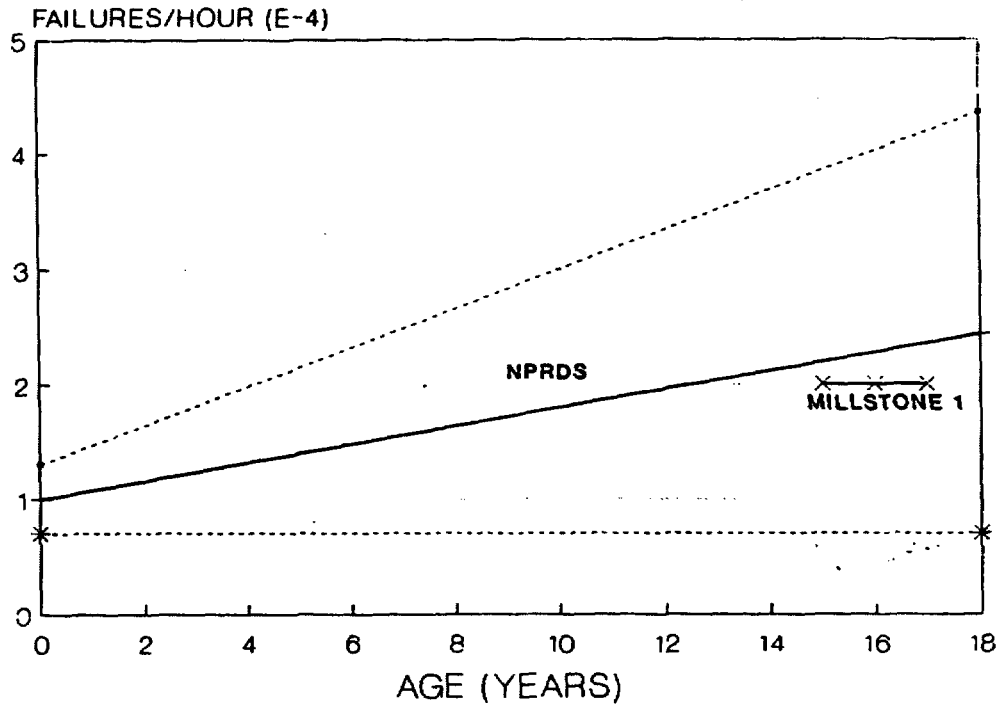


Figure 6.8 Heat Exchanger Leakage Failure Rate for Millstone 1

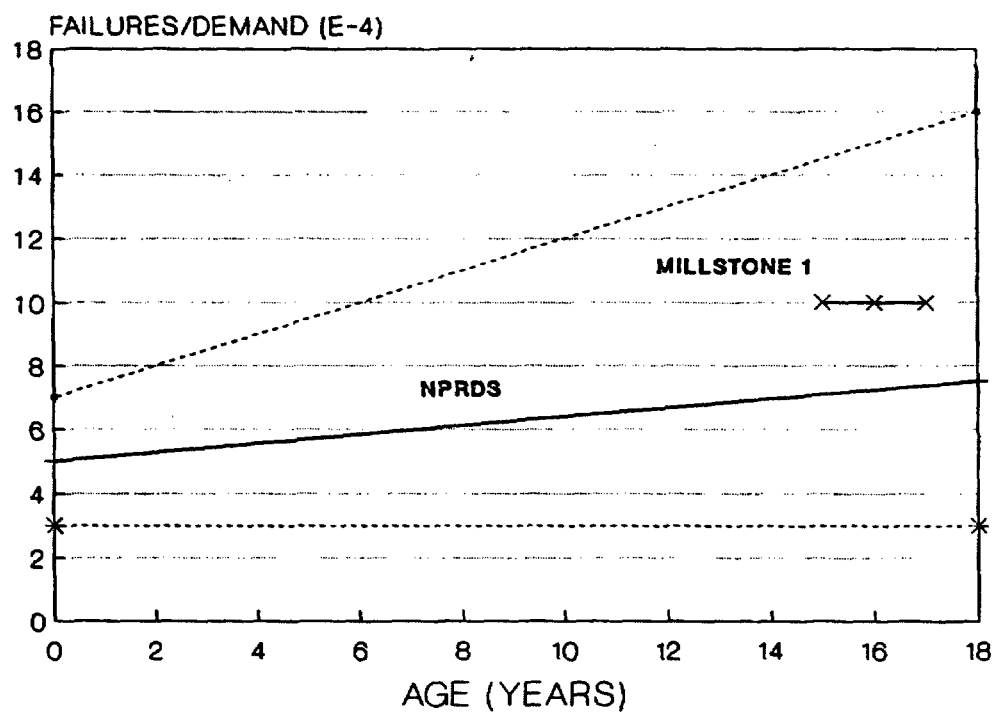


Figure 6.9 Pressure Switch Loss of Function Failure Rate for Millstone 1

6.3 Summary of Findings

Comparison of the Millstone data analysis results with those from the national data bases has shown that the findings are consistent between the data sources. The aging characteristics identified from the Millstone data show a high aging fraction for the events reported, which resulted in degraded system operation. The predominant cause of failure was found to be normal service with the components most frequently failed being MOV's and instrumentation. In addition, failure rates calculated for several components were in good agreement with the national data base findings. This plant specific data, therefore, serves as validation that the data base findings are representative of aging characteristics in BWR RHR systems.

7. PRA MODEL OF THE RHR SYSTEM

7.1 Overview

To supplement the deterministic work performed for this study a probabilistic analysis was also performed. In a probabilistic risk assessment (PRA), component failure rates are used as a basis to calculate system unavailability and component importances. The calculations can also be extended to the plant level to provide estimates of core melt frequency. A limitation of current PRA techniques is that time-dependent aging effects are not addressed.

The purpose of this analysis was to examine how the effects of aging are reflected in predictions of system unavailability and component importances. To do this, time-dependent component failure rates were calculated from the data and used as input to a PRA model.

7.2 PRAAGE Computer Program for Aging Assessment

To perform the probabilistic study, a plant with an existing PRA was chosen and its RHR system model was used. For this study, Peach Bottom Unit 2 (PB-2) was selected since its RHR system design is fairly common and it is an older plant. The complete Peach Bottom PRA study is documented in NUREG-1150[17].

To implement the PB-2 RHR PRA model and perform time-dependent PRA calculations, a computer program was developed by BNL. The program, PRAAGE-1988, performs PRA calculations for various plant ages to predict the effect of aging on system unavailability and component importance. The required input to the program includes time-dependent failure rates for the various system components, which for this study were obtained from the NPRDS data analysis presented in Section 5. The PRAAGE program is discussed in detail in Appendix D.

The following subsections discuss the PB-2 PRA model used for this analysis and present the results obtained from the PRAAGE program.

7.3 PRA Model of the Peach Bottom RHR in the Shutdown Cooling Mode (SDC)

7.3.1 Description of the SDC Mode

The function of the SDC mode of RHR is to remove decay heat during accidents in which the reactor vessel integrity is maintained.

The PB-2 SDC system is a two-loop system consisting of motor-operated valves (MOV) and electric motor driven pumps. There are two pump/heat exchanger trains per loop, with each pump rated at 10,000 gpm at a head of 20 psid. Cooling water flow to the heat exchanger is required for the SDC mode. The SDC system takes suction from one recirculation line. Figure 7.1 shows this as line PS-31, as well as other major components and pipe segment definitions used in the fault tree.

The SDC system is manually initiated and controlled because a fast response is not required. The primary success criterion for the SDC system is the injection of flow from any one of the two pump/heat exchanger trains to the reactor vessel.

Most of the SDC system is located in the reactor building. Local Access to the SDC system could be achieved by containment venting. Room cooling failure is assumed to fail the SDC pumps in four hours.

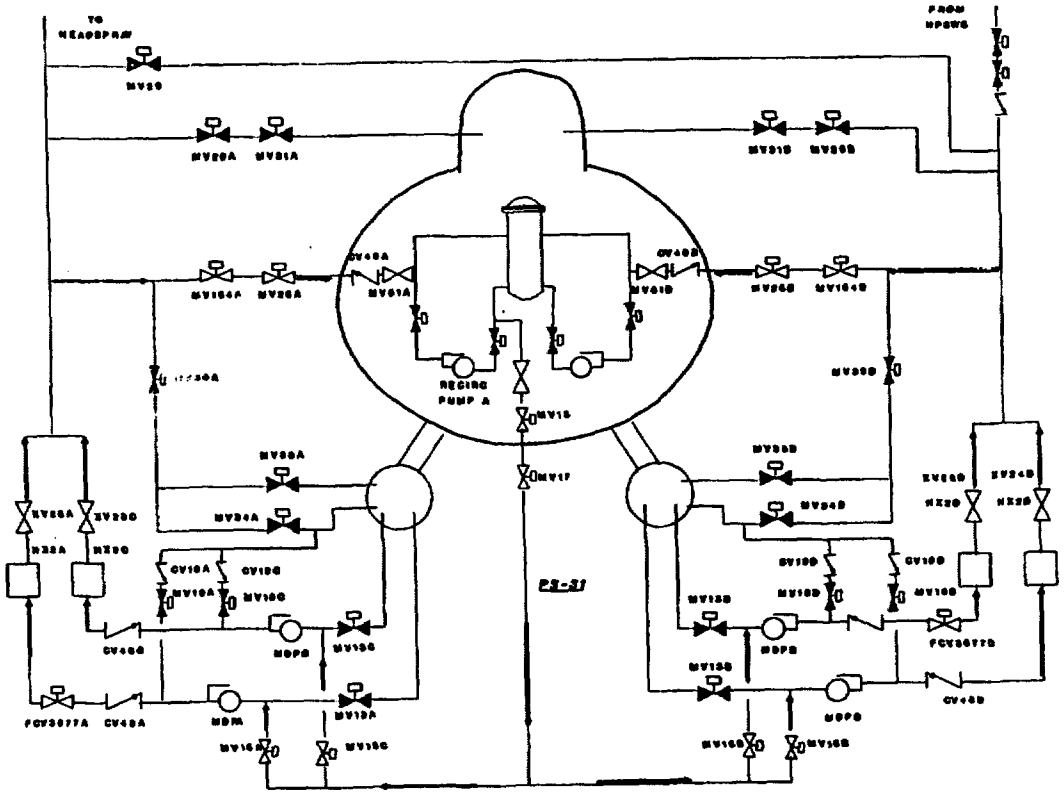


Figure 7.1 Simplified Schematic of the Residual Heat Removal System in the Shutdown Cooling mode

7.3.2 SDC Interfaces and Dependencies

Each SDC pump is powered from a separate 4160 Vac bus with control and actuation power being supplied by a separate 125 Vdc bus. All pumps require pump cooling. A simplified dependency diagram of the SDC is shown in NUREG/CR-4450[17].

Each loop's normally closed injection valve receives motive power from one of two 480 Vac sources. The loop A injection valve sources are either 480 Vac/A or 480 Vac/C and the loop B injection valve sources are either 480 Vac/B or 480 Vac/D.

Many components of the SDC system are shared with the different modes of the RHR system. These commonalities are: 1) the RHR pumps are common to the SDC, SPC, CS and LPCI modes, 2) Loops A and B injection valves are common to the SDC, LPCI and HPSW injection modes; and 3) heat exchanger cooling is common to the CS, SDC and SPC modes. The two SDC suction valves are common to all four SDC pumps. Valve MV17 requires 250 Vac/B and valve MV18 requires 250Vac/A. If either of these valves fails to open, the SDC fails completely. Each pump's suppression pool suction valve and the SDC cooling suction valve are interlocked to assure that one valve is fully closed before the other can open.

SDC is initiated after emergency core injection is successful and reactor pressure is low. If an injection signal subsequently occurs, the RHR system will automatically be realigned to the LPCI mode. SDC cannot be initiated if any of the following conditions exists: 1) reactor pressure greater than 225 psig, 2) high drywell pressure or 3) low reactor water level.

7.3.3 SDC Test and Maintenance

The SDC surveillance test requirements are: 1) pump operability - once per month, 2) MOV operability - once per month, 3) pump capacity test - quarterly, 4) simulated automatic actuation test - once/operating cycle, and 5) logic functional test - semiannually.

7.3.4 SDC Technical Specifications

Because of the sharing of equipment, the LPCI mode affects the SDC mode. If any one LPCI pump is found to be inoperable, operation may continue for 7 days provided that the remaining LPCI components and both loops of the LPCS system are operable.

7.3.5 Logic Model

The SDC system was modeled by fault trees, as presented in NUREG/CR-4450[17] Vol. 4, Appendix A. The work reported here used the fault tree cutsets obtained from Sandia National Laboratory, as well as the quantification data which was used in the initial testing of PRAAGE. As field data, and especially time-dependent field data were obtained, these initial data were replaced.

The fault tree modeling contains the major active components and most passive components. Components within a pipe segment were grouped to form a single

basic event in the fault tree. Generally a pipe segment is the pipe run between junctions. A segment was defined to contain components and piping having common dependencies and/or terminating at the containment penetration. Pipe ruptures were considered to be negligible. Only piping with a diameter greater than or equal to one-third of the main system piping diameter were considered to be a diversion path. Two human errors were incorporated in the fault tree model: miscalibration of various sensors and failure of manual initiation.

7.4 PRA Model of the Peach Bottom RHR in the LPCI Mode

7.4.1 Description of the LPCI Mode

The function of the LPCI system is to provide a makeup coolant source to the reactor vessel during accidents in which system pressure is low compared with operating pressure. This low pressure may be achieved by using the ADS system to reduce the pressure to LPCI operability. The LPCI mode is one of four modes of the RHR system and consequently shares components with other modes.

The PB-2 LPCI system is a two-loop system consisting of motor-operated valves (MOV) and electric motor driven pumps. There are two pump/heat exchanger trains per loop, with each pump rated at 10,000 gpm at a head of 450 psid. The heat exchanger bypass line is opened during LPCI operation. Cooling water flow to the heat exchangers is not required for the LPCI mode. The LPCI supply is the suppression pool, as shown in Figure 7.2.

The LPCI mode is automatically initiated and controlled. Operator intervention is only required to manually start the system given an auto-start failure, and to stop the system or control flow during an ATWS, if so required. The success criterion for the LPCI system is the injection of coolant from any one pump to the reactor vessel.

Most of the LPCI system is located in the reactor building. Local access could be affected by either containment venting or failure. If room cooling fails for 4 hours, it is assumed that the LPCI pumps fail.

7.4.2 LPCI Interfaces and Dependencies

Each LPCI pump is powered from a separate 4160 Vac bus with control and actuation power being supplied by a separate 125 Vdc bus. All pumps require cooling. Each loop's normally closed injection valve can receive motive power from one of two 480 Vac sources. The loop A injection valve sources are either 480 Vac/A or 480 Vac/C, and the loop B injection valve sources are either 480 Vac/B or 480 Vac/D, as indicated in the simplified dependency diagram shown in NUREG/CR-4550.

Many components of the LPCI system are shared with the different modes of the RHR system. These commonalities are: 1) the RHR pumps are common to the LPCI, SPC, CS and SDC modes, 2) the suppression pool suction valve for each pump train is common to the LPCI, SDC and CS modes, and 3) loops A and B injection valves are common to the LPCI, SDC and high pressure service water injection modes.

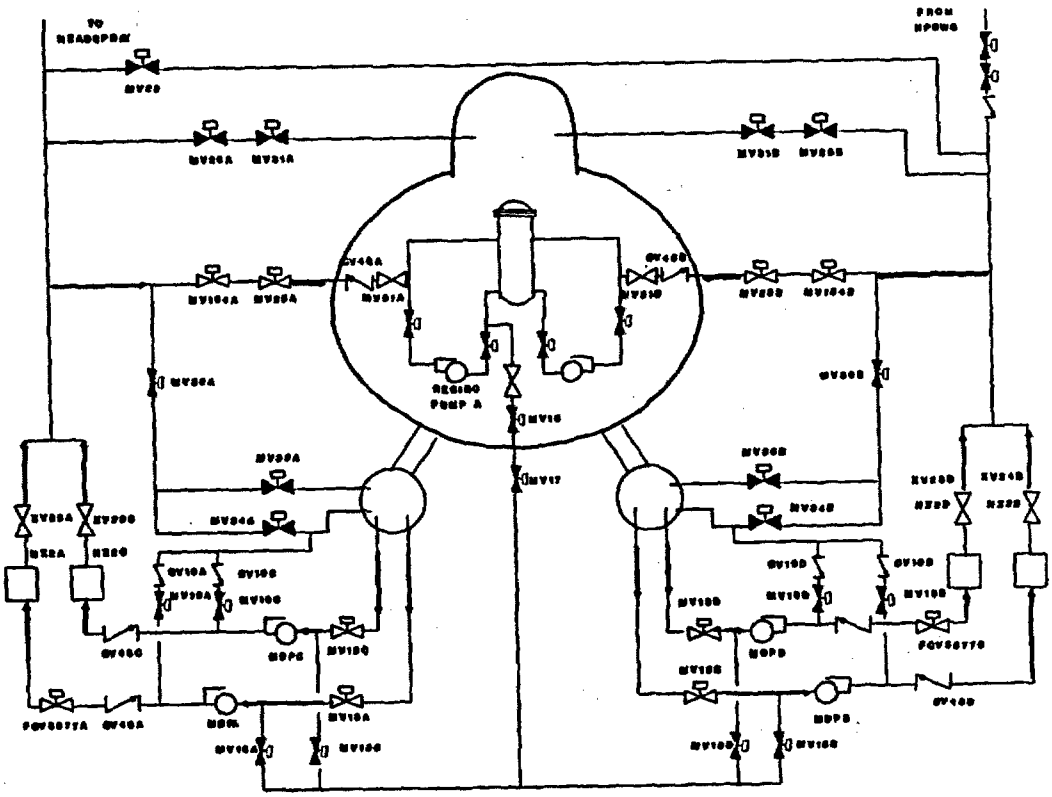


Figure 7.2 Simplified Schematic of the Residual Heat Removal System in the Low Pressure Coolant Injection Mode

Upon receipt of a LPCI injection signal, start signals are sent to all pumps. Loops A and B injection valves are subsequently demanded to open when reactor pressure is low enough and the test return valves are demanded to close. The LPCI system is automatically initiated on the receipt of either a low-low reactor water level (378 inches above vessel zero), or a combination of high drywell pressure (2 psig) and low reactor pressure (450 psig). All actuation sensors are shared with the LPCS system.

LPCI actuation and control circuitry is divided into two divisions. Division A is associated with the actuation and control of components in loop A, and division B serves a similar purpose for loop B. Each LPCI pump and loop injection valve receives an actuation signal from both divisions. Although the LPCI system has no isolation signals, there are permissives which will prevent the operation of certain components. LPCI pumps are demanded to stop or prevented from starting if the suppression pool suction valve or any of three SDC suction valves is not fully open. Loops A and B injection valves are prohibited from opening unless a low reactor pressure permissive (@450 psig) is received.

7.4.3 LPCI Test and Maintenance

The LPCI surveillance test requirements are: 1) pump operability - monthly, 2) MOV operability - monthly, 3) pump capacity - quarterly, 4) simulated automatic actuation - once each operating cycle, and 5) logic system functional - semiannually.

7.4.4 LPCI Technical Specifications

If any one LPCI pump is inoperable, operation may continue for seven days provided that the remaining LPCI components and both loops of the LPCS are operable.

7.4.5 Logic Model

The LPCI system for the injection of coolant into the reactor vessel was modeled by fault trees, as presented in NUREG/CR-4550 Vol. 4, Appendix A. The work reported here obtained the fault tree cutsets from Sandia National Laboratory as well as the quantification data which was used in the initial testing of PRAAGE. As field data and especially time-dependent field data were obtained, these initial data were replaced.

The fault tree modeling contains the major active components and most passive components. Components within a pipe segment were grouped to form a single basic event in the fault tree. Generally a pipe segment is the pipe run between junctions. A segment was defined to contain components and piping having common dependencies and/or terminating at the containment penetration. Pipe ruptures were considered to be negligible. Only piping with a diameter greater than or equal to one-third of the main system piping diameter were considered to be a diversion path. Two human errors were incorporated in the fault tree model: miscalibration of various sensors and failure of manual initiation.

7.5 Basecase PRAAGE Results

The PRAAGE computer model has been used to calculate system unavailability and relative importance of various RHR system components in both the SDC and LPCI modes of operation. The relative importance of each component is a measure of its contribution to the unavailability of the system as compared to other components in the system (e.g., if the relative importance of a component is 50%, half of the unavailability of the system would be due to failure of that particular component). The relative component importances are calculated based

on a normalized inspection importance which is the components inspection importance divided by the sum of the inspection importances for all components in the system. Additional detail on component importances can be found in Reference 15.

Using time-dependent failure rates in a PRA model, the effects of aging on component relative importance can be examined. If the failure rate of a component increases more rapidly with age than other components, it is possible that the relative importance of that component can change. The effects of aging could, therefore, make a component with an initially low relative importance become more important as the system ages. This may indicate that the component should receive increased attention in later years to mitigate increases in system unavailability. The basecase PRAAGE results discussed in this section have been obtained using the following three sets of failure frequency data:

1. Constant failure frequencies obtained from the Peach Bottom Unit-2 (PB-2) PRA;

2. Constant failure frequencies obtained from the NPRDS data review described in Section 5,

3. Time-dependent failure frequencies developed from the NPRDS data review described in Section 5.

The flexibility and efficiency of the PRAAGE model has permitted analyzing both the LPCI and SDC modes of operation with all three sets of data.

7.5.1 LPCI Mode

As discussed in Section 7.4, the PB-2 PRA provides a model for the RHR system during the LPCI mode of operation. This model has been used with PRAAGE to determine the system unavailability and the relative importance of various components in the system. The initial calculations were performed with the failure rates and PRA modeling as discussed in NUREG/CR-4550. A summary of the failure rates used in the calculations is given in Table 7.1.

Table 7.1 Summary of Generic Failure Rates - PB-2 PRA

MOVs: Fail to operate	3.8×10^{-3}/demand
MOVs: Limit switch failure	3.8×10^{-4}/demand
MOVs: Out for maintenance	3×10^{-4}/demand
RHR Pump: Fail to start	3.2×10^{-3}/demand
RHR Pump: Fail to run	2.0×10^{-5}/hr
RHR Pump: Out for maintenance	7.0×10^{-4}/demand
Air Compressor: Loss of function	7.0×10^{-4}/demand
Pressure & Level Sensors: Miscalibration	1.0×10^{-4}/demand
Level Sensor: Loss of function	2.5×10^{-3}/demand
Pressure Sensor: Loss of function	2.5×10^{-3}/demand
ESW: Actuation	1.0×10^{-6}/hr
ESW: Pipe segment faults	4.0×10^{-5} to 3.0×10^{-4}/demand

Table 7.2 shows the importance ranking of system and components for the LPCI mode of operation in terms of the percentage contribution to total system unavailability. These results indicate that the dominant contribution to unavailability is from failures of the emergency service water (ESW) system. The total system unavailability for this case is $4x10^{-4}$, a comparable result was not available from the PB-2 PRA. However, an independent calculation with the SETS computer program[18] was performed to provide a verification of the PRAAGE result. The two calculations were in excellent agreement providing a benchmark on the accuracy of the PRAAGE model.

Table 7.2 LPCI Mode System and Component Importance - PB-2 PRA Data

System/Component	Percent Contribution
ESW System: All modes	53.0
Pressure/Level Sensor: Miscalibration	18.0
MOVs: Fail to transfer	7.9
MOVs: Out for maintenance	5.0
Pressure Sensor: Loss of function	4.1
Ventilation System: All failure modes	3.9
Diesel Generator: Fail to start	1.3
AC Power: All modes	0.9
Level Sensors: Loss of function	0.4
Pipe Segment Fault	0.3
Others	5.0

The ESW system and several other systems and components modeled in the PB-2 PRA are considered outside of the boundary that has been defined for the RHR NPAR study (Section 1). In addition, it is of interest to analyze the availability of the RHR/LPCI components in detail. Thus, for the analysis described in this section, the failure probability of the ESW system and other systems supporting RHR/LPCI operation have been set to zero. This assumption implies that these support systems operate perfectly and allows the PRAAGE calculation to rank the importance of the RHR/LPCI components. Other supporting systems and components that have been eliminated from the PRA calculations include ac and dc power systems, instrument air system, emergency heating and ventilating system, high pressure service water, and the diesel generators. In addition, the Peach Bottom RHR system has been designed to support both Unit 2 and 3. Thus, the PB-2 PRA model includes elements that refer to interactions from the other unit. These interactions have been eliminated for the analyses described in this report since it is considered desirable to isolate the results for a system supporting a single plant.

With the modifications to the PB-2 failure probabilities described above, the total system unavailability was recalculated using PRAAGE to be $1.5x10^{-4}$. In addition, the importance ranking of the RHR/LPCI components with the effects of the support systems removed is shown in Table 7.3.

Table 7.3 LPCI Mode Component Importance
PB-2 PRA Data with Support Systems Removed

Component	Percent Contribution
Sensor Miscalibration	48.0
MOVs: Fail to transfer	23.0
MOVs: Out for maintenance	15.0
Pressure Sensor: Loss of function	12.0
Pipe Segment Faults	1.0
Others	1.0

It should be noted that although the LPCI pumps perform an important function in the LPCI mode of operation, these pumps are not included in the ranking of important system components. These pumps make a very small contribution to the total system unavailability due to the redundancy in the system design. For example, the system includes four pumps; however, only one operational pump is required for mission success in the PRA analysis for the LPCI mode. Thus, system redundancy has sufficiently lowered the importance of the RHR pumps such that pumps are not an important contributor to the system unavailability and do not appear in the component ranking shown in Tables 7.2 and 7.3.

Since the effects of all supporting systems have been removed, Table 7.3 includes only components that are part of the RHR/LPCI system. The component groupings have been developed such that similar failure modes are combined in a particular group. For example, the sensor miscalibration grouping includes containment drywell pressure sensors, reactor vessel pressure sensors, and reactor vessel water level sensors. Each of these components have a failure probability of miscalibration of $1x10^{-4}$ and have been grouped together for this analysis. A summary of the components included in each group is given in Table 7.4.

Table 7.4 Description of PRAAGE Component Grouping - LPCI Mode

Component Grouping	Description
MOVs: Fail to transfer	2 RHR system MOVs
MOVs: Out for maintenance	8 RHR system MOVs
MOVs: Limit switch fails to indicate valve position	4 MOV limit switches
Pressure Sensors: Loss of function	4 reactor pressure sensors
Sensor Miscalibration	Reactor pressure, reactor water level, drywell pressure sensors
RHR Pump: Fails to run	4 RHR pumps
RHR Pump: Fails to start	4 RHR pumps
RHR Pump: Out for maintenance	4 RHR pumps
Pipe Segment Faults	2 sections of system piping

The importance ranking of the components shown in Tables 7.2 and 7.3 have been based on failure probabilities obtained from the PB-2 PRA. However, the analysis of NPRDS data described in Section 5 of this report has identified several component failure probabilities that differ from the NUREG/CR-4550 values. For the LPCI mode of operation, the NPRDS data analysis has identified failure probabilities for two component groups identified in Table 7.4: motor-operated valves and the pressure sensors associated with LPCI initiation. The PRAAGE model was updated with the values of failure probabilities for these components, as determined in Section 5. As shown in Table 7.5, the new probabilities are slightly reduced from the values used in the PB-2 PRA.

Table 7.5 LPCI Mode Failure Probabilities: BNL Data Analysis

		Failure Probability Per Demand		
Component	Failure Mode	PB-2 PRA Mean Value	BNL Analysis Nominal Value	BNL Analysis Maximum Value (1)
Motor-operated Valves	Fail to Transfer	3.8×10^{-3} (2)	1.0×10^{-3}	1.3×10^{-3}
Pressure Sensor	Loss of function	2.5×10^{-3} (3)	6.0×10^{-4}	8.4×10^{-4}

(1) Based on the uncertainty analysis described in Section 5.
(2) Generic value.
(3) Wash 1400, plant data.

The failure probability associated with the pressure sensor in the BNL analysis was determined by an arithmetic combination of the pressure sensor and pressure switch data given in Section 5. The component importances, calculated with the modified data are shown in Table 7.6. It should be pointed out that comparison of Tables 7.3 and 7.6 indicates a change in percent contribution for each of the components. This is due to the use of different failure rates for MOV failure to transfer and pressure sensor loss of function, as presented in Table 7.5. Since both of these failure rates were lowered from the original values used in the PB-2 analysis, the corresponding percent contribution to system unavailability due to that component was lowered. This in turn causes an increase in contribution from other components.

The component importances shown in Table 7.6 represent the initial starting point for examination of aging effects. By allowing the component failure rates to vary with time, changes in these values will indicate the effects of aging. The overall system unavailability for this initial case was 1.1×10^{-4}.

Table 7.6 LPCI Mode Component Importances:
PB-2 and BNL Data With Support Systems Removed

Component	Percent Contribution
Sensor Miscalibration	77.0
MOVs: Out for maintenance	13.0
MOVs: Fail to transfer	5.5
Pressure Sensors: Loss of function	1.1
Pipe Segment Faults	0.9
Others	2.5

The results, ranked by components, emphasize the importance of the sensor miscalibration on the LPCI mode system unavailability. As discussed previously, these sensors include reactor and containment drywell pressure and reactor water level and are the initiating devices for LPCI operation during a design basis accident. Thus, the importance indicated in Table 7.6 is reflected in the importance of these sensors in the operation of the system. Significant reductions in system unavailability are possible through reductions in the potential for miscalibration of these sensors. Improvements in training, procedures, and maintenance implementation and frequency may have the potential for reductions in the LPCI mode unavailability.

It should also be noted that Reference 19 describes a ranking of the important LPCI mode components based on PRA considerations independent of the results described in this report. The Reference 19 ranking of components includes miscalibration of pressure and water level sensors, and motor-operated valves (maintenance and failure to transfer) among the most important LPCI failure modes. Thus, similar rankings have been obtained from two independent analyses.

7.5.2 SDC Mode

The analysis of the shutdown cooling mode of RHR operation follows the same strategy as described for the LPCI mode. The system unavailability, utilizing the failure probabilities and PRA modeling described in NUREG/CR-4550, was calculated by PRAAGE to be 4.2×10^{-2}. The importance ranking of all SDC components and support systems is shown in Table 7.7.

As shown, selected plant operational conditions have the highest contribution to SDC mode unavailability. The predominant one is low reactor water level (i.e., water level below the shroud) which automatically isolates the RHR system and prevents operation in the SDC mode. Of the components included in this ranking, MOVs and pressure sensors make the largest contribution to the system unavailability. As noted, the percent contribution to system unavailability is different for the SDC mode as compared to the LPCI mode (Table 7.3). This is because the LPCI mode performs a different function than the SDC mode. Although some RHR components may be shared between modes, other components used to perform the specific mode function, along with the instrumentation needed to initiate and control operation can be very different. Components which are very important for one mode, therefore, may not be as important for the other mode. These component groups will be analyzed in detail in subsequent sections of this report.

Table 7.7 SDC Mode Systems and Component Importance
PB-2 PRA Data

System/Component	Percent Contribution
Low Reactor Water Level Conditions	46.0
MOVs: Fail to transfer	27.0
Pressure Sensors: Loss of function	12.0
MOVs: Out for maintenance	5.8
MOVs: Limit switch	1.8
Emergency Service Water System	1.5
Sensor Miscalibration	0.9
Others	5.0

In the SDC PRA model, the effects of support systems and interactions with the other units were removed by setting various failure probabilities to zero. The systems and components removed included AC and DC systems, emergency service water system, instrument air system, and the diesel generators. The calculated system unavailability for only SDC components based on the failure probabilities in the PB-2 PRA was $2x10^{-2}$. Table 7.8 ranks the components according to relative importance to the system unavailability.

Table 7.8 SDC Mode Component Importance - PB-2 PRA Model
With Support Systems Eliminated

Component	Percent Contribution
MOVs: Fail to transfer	56.0
Pressure Sensors: Loss of function	24.0
MOVs: Out for maintenance	12.0
MOVs: Limit switch	3.7
Sensors: Miscalibration	1.9
RHR Pumps: Fail to start	0.5
RHR Pumps: Fail to run	0.4
RHR Pumps: Out for maintenance	0.3
Others	1.2

The BNL analysis of NPRDS data described in Section 5 has determined the failure probability of four components that make contributions to the unavailability of the RHR system in the SDC mode of operation. These components are shown in Table 7.9 with the corresponding value of failure probability determined by the BNL analysis and the values from the PB-2 PRA.

Table 7.9 SDC Mode Failure Probabilities: BNL Data Analysis

Component	Failure Mode	Failure Probability Per Demand: PB-2 PRA Mean Value	BNL Analysis Nominal Value	BNL Analysis Maximum Value (1)
MOV	Fail to transfer	3.8×10^{-3}(2)	1.0×10^{-3}	1.3×10^{-3}
RHR Pump	Fail to run	2.0×10^{-5}(2,3)	6.0×10^{-5}(3)	8.0×10^{-5}(3)
Level Sensor	Loss of function	2.5×10^{-3}(4)	6.0×10^{-5}	1.6×10^{-3}
Pressure Sensor	Loss of function	2.5×10^{-3}(4)	6.0×10^{-4}	8.6×10^{-3}

1. Based on uncertainty analysis described in Section 5.
2. Generic value.
3. Failures per hour converted to a failure probability using a mission time of 40 hours.
4. WASH-1400, plant data.

Based on the BNL data from Table 7.9 and the PRAAGE model excluding support systems, the system unavailability was calculated to be 8.1×10^{-3}. This value is slightly reduced from the previous calculations since the failure probabilities for the more important components are lower than the PRA values. The component ranking is shown in Table 7.10.

Table 7.10 SDC Mode Component Importance-PB-2 PRA Model With Support System Eliminated and BNL Data

Component	Percent Contribution
MOVs: Fail to transfer	37.0
MOVs: Out for maintenance	30.0
Pressure Sensors: Loss of function	15.0
MOVs: Limit switch failure	9.2
Sensors: Miscalibration	4.8
Manual Valves: Plugged	2.1
RHR Pumps: Fail to start	1.0
RHR Pumps: Fail to run	0.7
RHR Pumps: Out for maintenance	0.6

The results presented in this table are based on a PRA model that only considers the basic components of the RHR system in the SDC mode; i.e., external effects from support systems, other plants, and plant operational conditions have been removed. Thus, this model is appropriate for the analysis on aging effects on RHR components. The combination of this model with the data developed in Section 5 should represent a best estimate analysis of the RHR availability. The effects of aging on these components will be discussed in the next section.

7.6 Time-Dependent Failure Rates

The detailed analysis of NPRDS data described in Section 5 has determined the failure probability as a function of time for several components that are important during the LPCI and SDC modes of operation. These time dependent probabilities are representative of the aging of the system components. The capabilities of the PRAAGE model have been used to determine the time dependent system unavailability and relative component importance for each mode of operation. Basically, the system unavailability is found by adding the cutset probabilities for a particular generic component to give the generic component's contribution to the unavailability. A more detailed discussion of the PRAAGE code is presented in NUREG/CR-5052[15].

For the LPCI mode, the data shown in Figures 5.30 (MOVs), 5.33 (pressure switches) and 5.34 (pressure sensors) have been used to predict the system unavailability over a 50 year time period. As discussed previously, the pressure switch and sensor data have been combined to represent the pressure sensor failure probability in the PRA model. The input data for PRAAGE is given in Table 7.11.

Table 7.11 LPCI Mode PRAAGE Input Data for Aging Analysis

Component	Initial Failure Probability	Aging Rate
MOVs: Fail to transfer	1.0×10^{-3}/demand	0.11 fractional change/year
Pressure Sensors: Loss of function	6.0×10^{-4}/demand	0.02 fractional change/year

Based on this data and the PRAAGE model with the effects of support systems eliminated, the system unavailability over a 50 year time period was calculated and is shown in Figure 7.3. These predictions are based on a linear extrapolation of the failure rates discussed above assuming that current trends continue and no action is taken to change the aging rates. This assumption does not account for improvements to current practices which could reduce the aging rate, nor does it account for rapid increases in aging rate which have not yet been seen due to limited service time. The results presented, therefore, are estimates of future unavailability which indicate what could potentially occur if future trends continue. It should be noted that data is not yet available to confirm these predictions.

The figure shows a moderate increase in unavailability to a value of approximately 1.9×10^{-4} after 50 years. This represents less than a two fold increase from the initial value at 1.1×10^{-4} at age zero. This result is representative of the moderate aging rates that have been identified for MOVs and pressure sensors. The relative importance of the various component groups defined for the LPCI mode (Table 7.4) are shown as a function of time in Figure 7.4. This figure clearly shows the effects of aging on the importance of the MOVs and the pressure sensors. These components show an increasing level of importance throughout the time period analyzed. The unavailability contribution

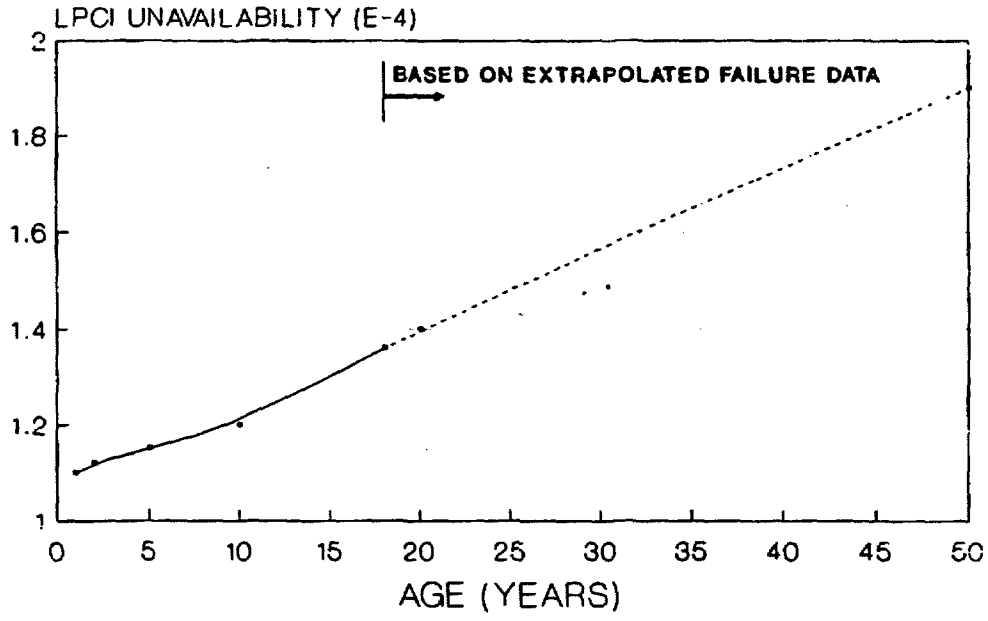


Figure 7.3 Time-Dependent LPCI Mode Unavailability Using BNL Nominal Aging Rates

of sensor miscalibration remains dominant because of the relatively high initial contribution to system unavailability. However, MOVs show the potential for becoming equally important in later years. This result indicates that additional emphasis on training and maintenance practices related to instrument calibration could reduce RHR system unavailability, and that increased attention to MOVs should be considered for extended life operation.

The SDC mode of operation has been analyzed in a similar manner. In this case, the analysis of NPRDS data determined that four components represented in the PRA model could be characterized by a time dependent failure probability. These components and the associated PRAAGE input parameters are shown in Table 7.12.

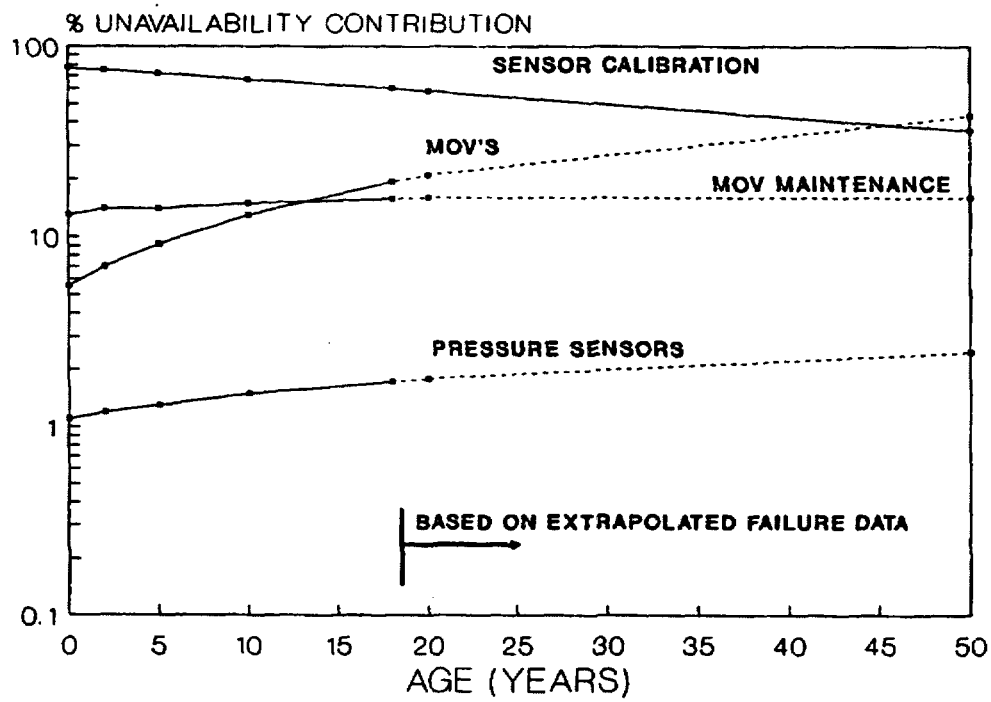


Figure 7.4 Time-Dependent LPCI Mode Component Improtances Using BNL Nominal Aging Rates

Table 7.12 SDC Mode: PRAAGE Input Data for Aging Analysis

Component	Initial Failure Probability	Aging Rate
MOVs: Failure to transfer	1.1×10^{-3}/demand	0.11 Fractional change/year
RHR Pumps: Fail to run	2.4×10^{-3}	0.16 Fractional change/year
Pressure Sensors: Loss of function	6.0×10^{-4}/demand	0.02 Fractional change/year
Level Sensors: Loss of function	1.1×10^{-3}/demand	0.

Figure 7.5, shows SDC system unavailability as a function of time based on the PRAAGE aging model. As for the LPCI mode predictions, these results are based on linear extrapolations of the failure rates calculated from the data and assume that current trends continue. The figure indicates a slow increase in unavailability to approximately 3.5 times the initial value over a 50 year time period. This result is the consequence of the moderate slope of the component failure probability data shown in Figure 5.31 and 5.37.

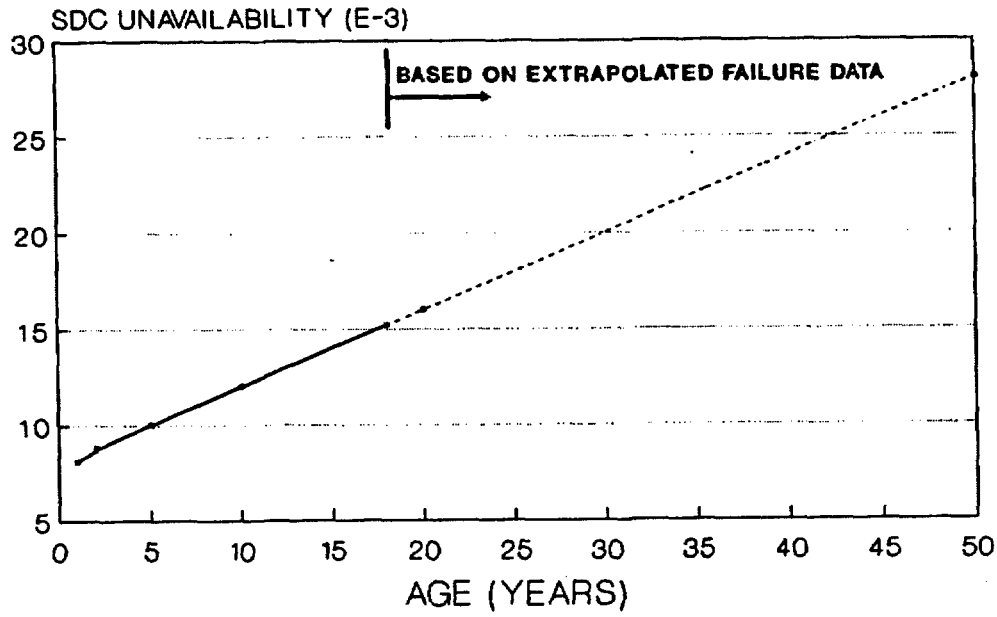


Figure 7.5 Time-Dependent SDC Mode Unavailability Using BNL Nominal Aging Rates

Figure 7.6 shows the relative importance versus time of the SDC components making the largest contribution to system unavailability. This result indicates that MOV failures dominate system unavailability over the entire period analyzed. The importance of MOVs increases from 37% to 70% in a 50 year time period. The largest percentage increase in relative importance is associated with the RHR pumps. Over the period analyzed, the importance of the RHR pumps increases from 0.7% to approximately 7% reflecting the effects of aging on the pump performance.

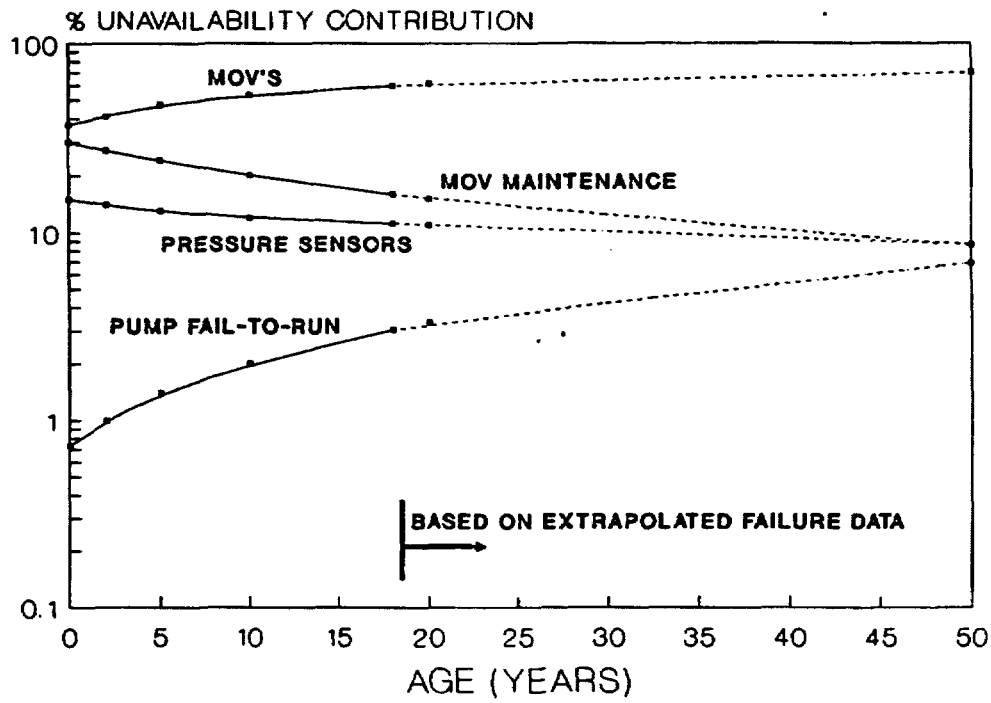


Figure 7.6 Time-Dependent SDC Mode Component Importance Using BNL Nominal Aging Rates

8. SENSITIVITY STUDIES

This section presents the results of an assessment of the effects of variations in the component failure probabilities and the component aging parameters that have been discussed in Section 7. The purpose of this analysis is to examine the sensitivity of results based on calculated time-dependent failure rates to uncertainties in the failure rates. The PRAAGE computer model has been used to determine the sensitivty of the base case results to statistically determined upper bounds on the failure frequencies for various components. The results are shown in terms of changes in the overall system unvailability for both the LPCI and SDC modes of operation.

8.1 LPCI Mode of Operation

The analysis of NPRDS data described in Section 5 has identified two components which are important to the LPCI mode of operation, that can be characterized by a time-dependent failure probability. Analysis results demonstrating the effects of the varying failure probabilities of motor-operated valves and pressure sensors on system unavailability have been presented in Section 7.6. However, the statistical analysis described in Section 5 has also determined an upper bound on the time-dependent failure probabilities for each component. The effects of these upper bounds on system unavailability in the LPCI mode of operation are discussed in this section.

Figures 5.30, 5.33, and 5.34 provide the maximum values for the initial failure probabilities and the maximum aging rate for the components important in the LPCI mode that have defined aging trends (MOVs, pressure switches and pressure sensors). As discussed previously, the pressure sensor and pressure switch data have been combined to represent the pressure sensors in the PRA analysis. The sensitivity of system unavailability to variations in the failure data for each component has been determined utilizing the PRAAGE model.

Figure 8.1 presents the results in terms of system unavailability calculated with MOVs and pressure sensors aging at the maximum rate compared with the nominal results that were described in Section 7.6. The figure shows a relatively small increase in system unavailability as each component ages at the maximum aging rates that have been determined for MOVs and pressure sensors.

The importance of each of the component groups that are defined in Table 7.4 has also been evaluated for each component aging at the maximum rate. Figure 8.2 shows the result for MOVs. This figure, when compared with Figure 7.4, shows the same general trends with a moderate increase in the unavailability contribution of MOVs. However, the order of ranking of important components is unchanged. This shows that the results are not very sensitive to the expected variations from the predicted aging rates.

A similar result has been obtained when the aging rate of pressure sensors was increased to the maximum value. The PRAAGE results are shown in Figure 8.3.

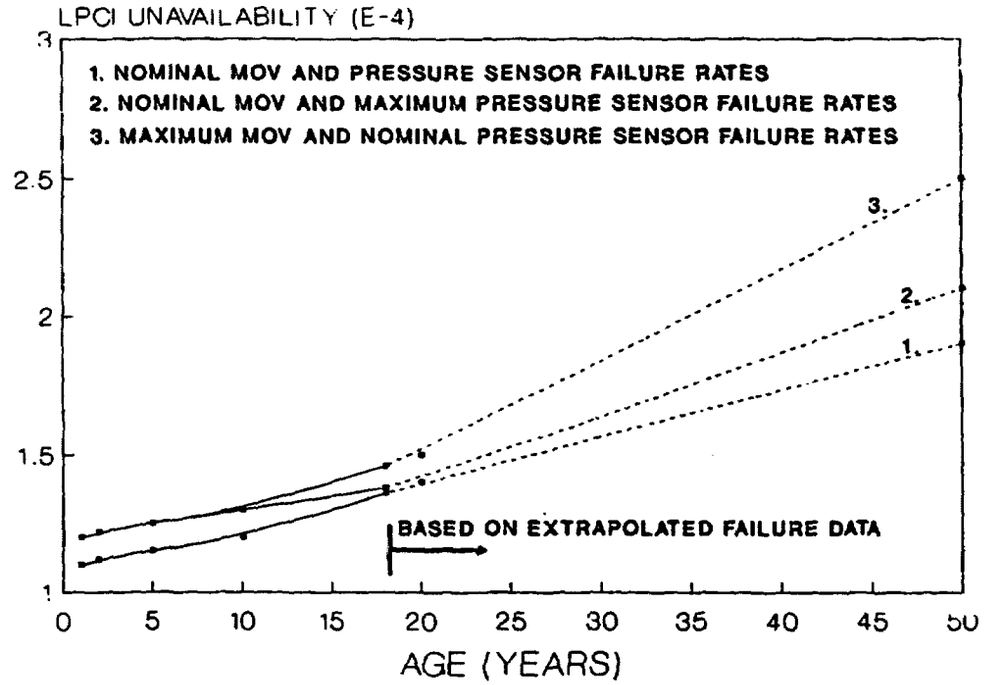


Figure 8.1 LPCI Mode Unavailability - Maximum MOV and Pressure Sensor Aging Rates

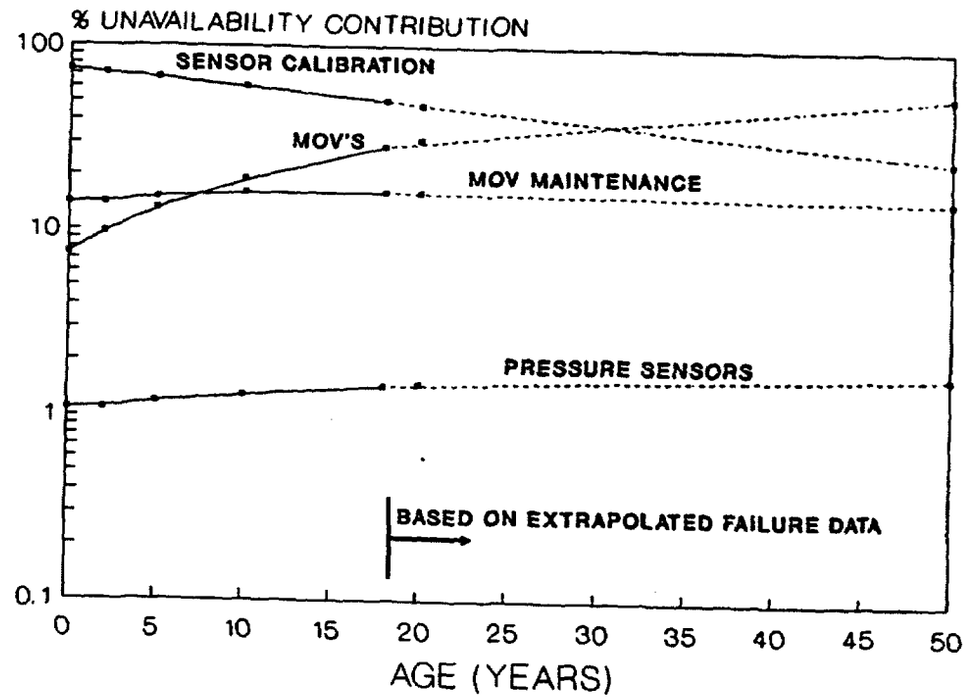


Figure 8.2 LPCI Component Importance - Maximum MOV Aging Rate

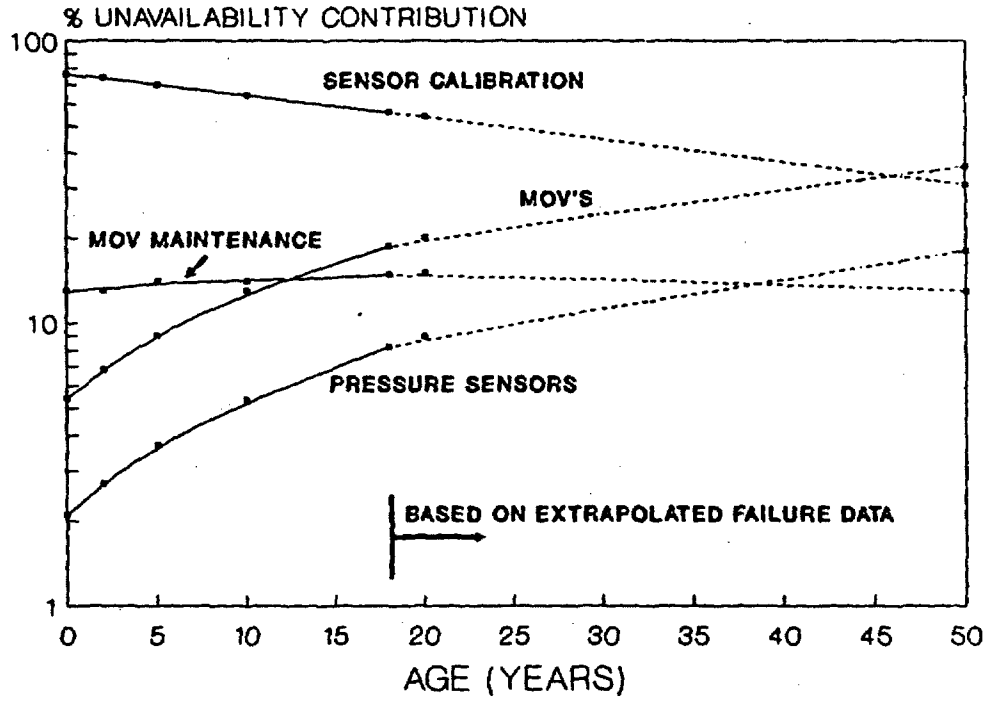


Figure 8.3 LPCI Component Importance - Maximum Pressure Sensor Aging Rate

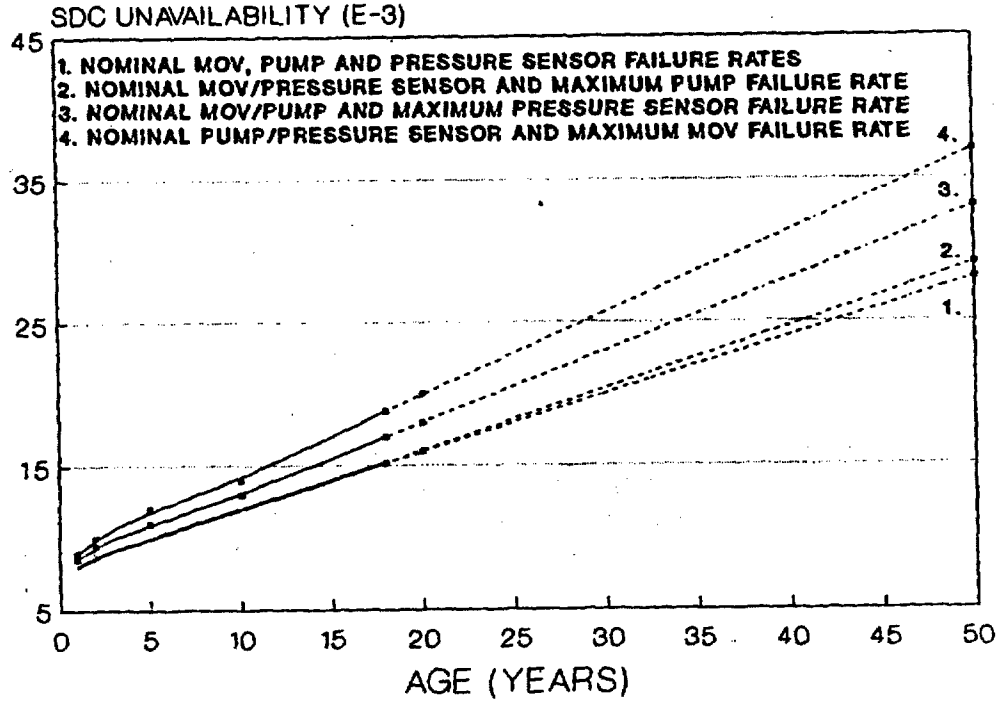


Figure 8.4 SDC Mode Unavailability - Maximum MOV, Pressure Sensor and RHR Pump Aging Rates

The increased importance of the pressure sensors is evident by comparison with the nominal results in Figure 7.4. In the nominal case, the contribution to unavailability of pressure sensors increased by a factor of 2.3 in a fifty year time period; for the extreme case (Figure 8.3), pressure sensor importance increased by a factor of 8.5 in the same time period. However, the accelerated aging of the pressure sensors does not change the overall component ranking from the nominal results. This indicates that, due to the dominance of the sensor miscalibration in the importance ranking, the results are relatively insensitive to changes in the aging models for motor-operated valves and pressure sensors.

8.2 SDC Mode of Operation

The failure data sensitivity analyses performed for the components important to SDC operation included an evaluation of the effect of three component failure rates on system unavailability. The PRAAGE model has been used to determine the change in system unavailability when MOVs, pressure sensors and the RHR pumps are aged at the maximum rate, as determined in Section 5. These results are shown in Figure 8.4.

As indicated in the figure, MOVs have the strongest sensitivity to system unavailability. Two reasons contribute to this sensitivity:

- MOVs, as a group, make the largest contribution to overall system unavailability in the SDC mode (Table 7.10); and
- MOVs have the second highest aging rate of the components analyzed in the SDC mode (Table 7.12).

The sensitivity of system unavailability for each component analyzed can be summarized in terms of the increase in unavailability over the fifty year time, as shown in Table 8.1.

Table 8.1 Summary of System Unavailability Increases - SDC Mode

Component	Unavailability Increase (From Initial Value)
MOVs: fail to transfer	4.1
Pressure sensor: loss of function	3.8
Pump: fail to run	3.6

The relative importance of each component group has also been determined for each of these sensitivity studies. As for the LPCI mode of operation, the relative ranking of each group does not change from the nominal case presented in Section 7.6. Figures 8.5 through 8.7 show the results graphically.

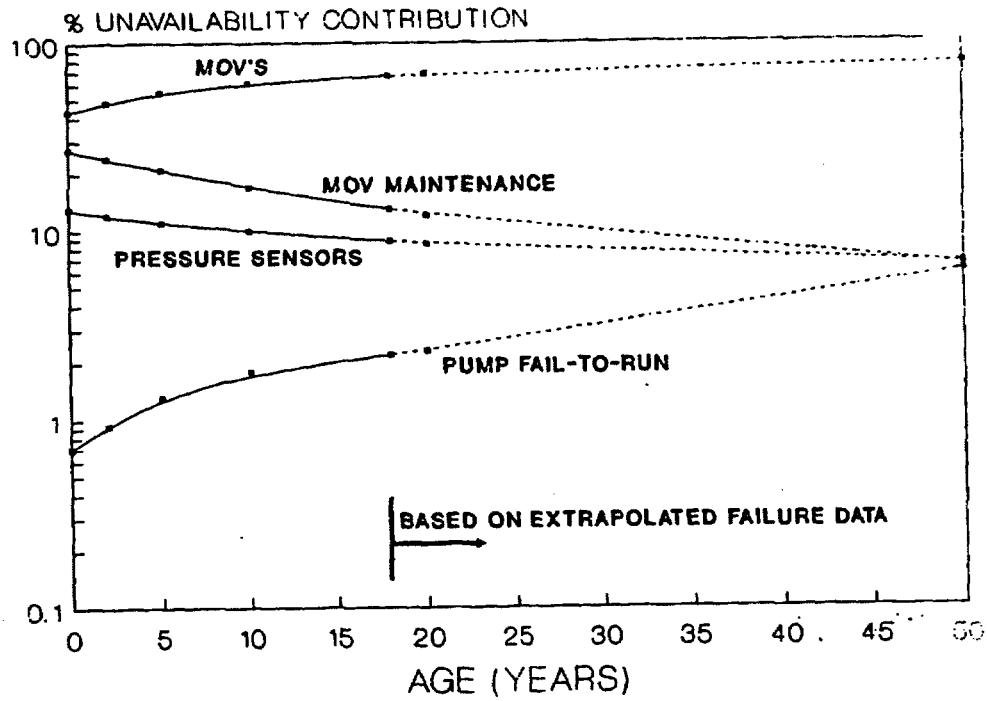


Figure 8.5 SDC Component Importance Ranking Maximum MOV Aging Rate

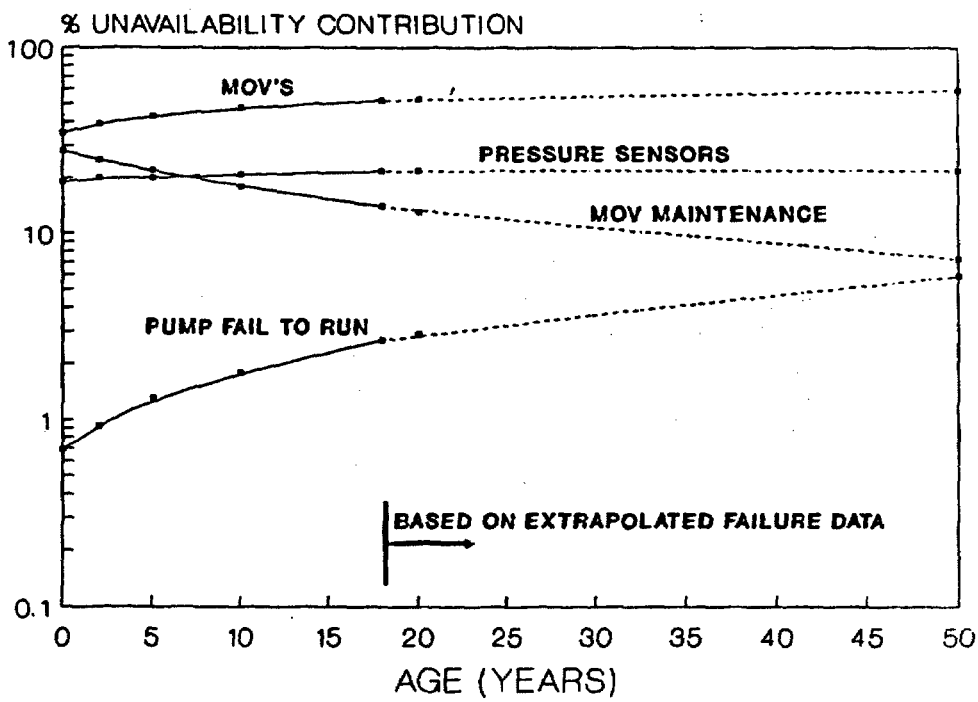


Figure 8.6 SDC Component Importance Ranking - Maximum Pressure Sensor Aging Rate

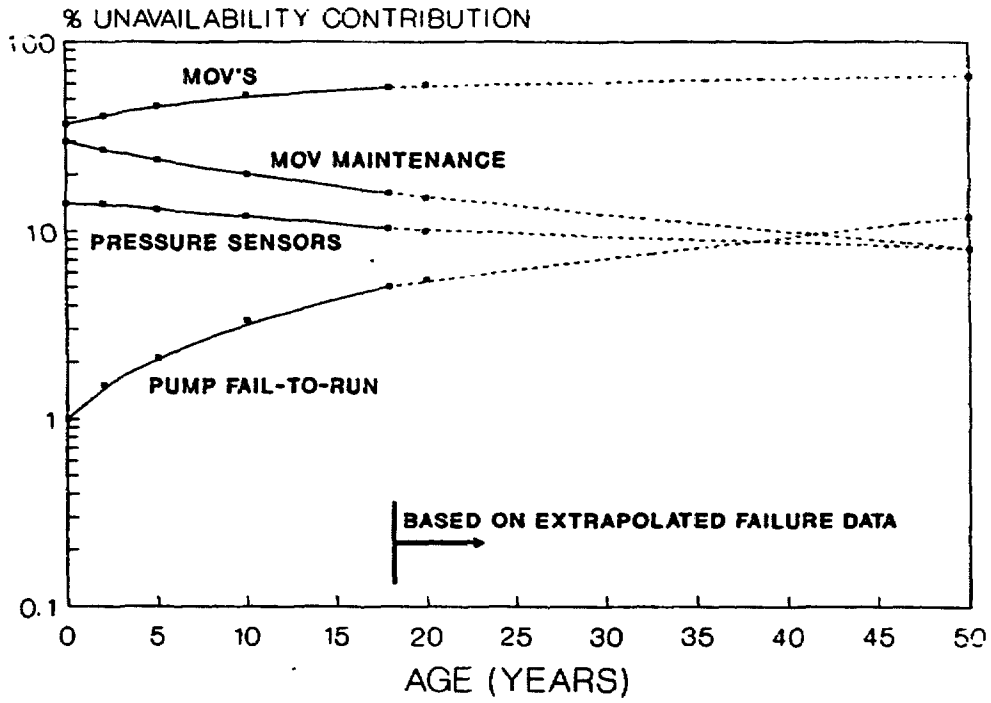


Figure 8.7 SDC Component Importance Ranking-
Maximum Pump Aging Rate

8.3 Summary

Sensitivity studies have been performed on each of the component aging models that were developed in Section 5. The results show that the nominal results presented in Section 7.6 are only moderately sensitive to variations in the aging parameters. Analysis of the LPCI mode unavailability shows that if the upper bound failure rates are used for the MOV's and pressure sensors, the largest LPCI unavailability increase would be a factor of approximately 2.5 in 50 years. This is only slightly higher than the factor of 2 increase observed using the nominal failure rates. For the SDC mode, upper bound failure rates produced a maximum unavailability increase of approximately a factor of 5 in 50 years. Again, this is only slightly higher than the factor of 4 increase obtained using nominal failure rates.

9. RESULTS

9.1 RHR Design Reviews

The RHR design reviews performed for this study have shown that several different configurations exist among the BWR plants. A common element among the designs is the use of redundancy in both the number of loops included in the design and the number of components available in the system. The use of redundancy to improve system availability was demonstrated to be important in the data analysis since failures were found in past operating experience which could have disabled the system if a redundant component or flow path were not available.

One area where redundancy is typically not employed is in the design of the shutdown cooling flow path. When operating in the SDC mode, suction for the pumps is obtained from one of the reactor water recirculation lines. Current designs typically use one common line to supply all loops of the RHR system. In the event of an isolation valve failure to open in this line, all RHR shutdown cooling capability can be lost. This has occurred on a number of occasions, as reported in the data bases. Under most conditions this is not considered to be a major problem since other means of providing shutdown cooling are normally available and, typically, sufficient time is available to correct the problem before plant safety is jeopardized. However, this design aspect does increase the unavailability of the SDC mode of RHR. Since MOVs are predominantly used as the isolation valves in the suction line, and the data have indicated a potential for increasing MOV failures with age, this problem could become more significant with age. In addition to MOVs, piping and other components in non-redundant lines should also be monitored for future aging degradation.

One design concern noted during the review of RHR system configurations is the use of a common minimum flow line for two RHR pumps. In this design, both pumps could be required to operate simultaneously with minimum flow established for both through the common line. As noted in NRC Bulletin 88-04, this could lead to dead headed operation and possible failure of one pump. This could occur if pump performance has degraded to the point where its developed head relative to the other pump is insufficient to maintain minimum flow. Since aging degradation has been found to occur in RHR pumps this is a valid concern.

9.2 Review of RHR Stresses

The review of operational and environmental stresses on the RHR system has shown that a number of different aging degradation mechanisms are present. However, the degree to which they contribute to the effects of aging on system performance are related to the operating status of the system. It was found that the RHR system is predominantly in the standby mode aligned for LPCI operation. While in standby, the aging mechanisms which could be active include corrosion, calibration drift, embrittlement and chemical reactions. Aging mechanisms related to operational stresses, such as wear and erosion, would not be active during standby and, therefore, should not be the dominant failure mechanisms for the RHR system. From the data analysis review, however, it was seen that wear was a significant RHR failure mechanism (Figure 5.14). Since this

mechanism is only active during system operation, the aging degradation due to wear which takes place while the system is in the SDC mode or while it is being tested contributes to a significant number of RHR failures. It is also possible that the effects of these operational stresses are enhanced by synergistic effects with various standby stresses.

9.3 Review of Current Practices

From a survey of nine BWR units it was found that a significant amount of testing and maintenance is currently performed on RHR systems. Test requirements are typically identified in the unit's technical specifications and include quarterly pump operational tests and periodic valve stroke time and seat leakage tests. It was found that several plants record additional information during testing which may be valuable for trending purposes; for example, pump bearing temperature and vibration. Since the data have shown that aging degradation does occur in RHR pumps this practice is expected to be very helpful in providing an indication of impending failure.

The survey also indicated that pumps and valves receive the most attention in terms of testing and maintenance. Only a few plants responded that monitoring and maintenance actions for heat exchangers were extensive. In addition, the survey showed that very few plants performed any type of data trending for any of the RHR components. Since trending is an important tool for detecting aging degradation, the implementation of a trending program should be considered by all plants.

The survey responses also indicated that operational readiness of the system could best be assured from three tests: 1) valve stroke tests, 2) control logic response tests, and 3) in-service inspection pump tests. This is supported by the data analysis findings which showed valves and instrumentation/controls to be the two predominant types of components experiencing failure in the RHR system. It is further supported by the survey response which identified these components as the ones requiring the most corrective maintenance.

9.4 Evaluation of Operating Data

An analysis of operating data has shown that a large percentage of RHR failures are aging related (69%). This indicates that degradation due to age is present in RHR systems and is not completely detected or controlled before it leads to failure. Most failures are detected by tests and inspections, however, 27% of the failures are not detected until they result in some operational abnormality. This includes events such as a valve failing to open on demand or radiation levels exceeding specified limits. It should be noted that no operational abnormalities were found which resulted in failure of the LPCI mode to perform its design function when called upon.

It was found from the data analysis that over half of the RHR failures resulted in degraded operation of the system. Failures in this classification typically were minor and included events such as valve packing leaks. The system could still perform its function with these failures present, however, the failed components would eventually have to be repaired or they would worsen.

The data also indicated that approximately one-fourth of the RHR failures resulted in a loss of redundancy. These events typically included pump seal leaks or MOV failures which required a component or loop to be taken out of service. These failures are significant since a loss of redundancy has a direct adverse effect on system availability. Other significant RHR failures are those which result in reactor scrams or engineered safety feature actuations. These events are varied and can include mechanical failures, such as valve leakage, or electrical failures, such as instrumentation/control malfunction. A third class of RHR failures which are significant are those that result in radiological releases. These events typically involve heat exchanger leaks. These findings show that aging degradation in RHR systems can have an adverse impact on system reliability, as well as on plant safety. It is, therefore, important to focus attention on better monitoring and mitigation of aging effects in RHR systems.

As part of the analysis, the causes of RHR failures were examined. It was found that normal service was the predominant cause, which is consistent with the large aging fractions seen. When examined as a function of age, it was seen that the fraction of failures caused by normal service remained relatively constant with time. This indicates that the aging effects present in the RHR system are either being mitigated by current maintenance and monitoring practices, or the increase with age is very slow and cannot be detected from this data source. In either case, the time-dependent aging effects on RHR systems appear to be moderate.

To characterize the effects of aging in the RHR system, the failure modes and mechanisms were identified. The failure modes are quite diverse; the predominant modes are leakage, loss of function and wrong signal. Leakage typically involves valves or pumps, while loss of function and wrong signal typically involves instrumentation and controls. The predominant failure mechanisms were found to be wear and calibration drift. This information can be useful in the assessment of inspection and monitoring practices since it identifies areas which should receive increased attention.

At the component level, the unnormalized data showed valves to be the most frequently failed component, followed by instrumentation/controls and supports. These findings are consistent with those from the survey results discussed previously. In particular, MOVs were found to be the type of valve most frequently failed, while switches were found to be the most frequent instrumentation/control failure. This can be attributed to the large populations for each of these components.

Component testing was also investigated as a contributor to aging degradation. It was found that the most frequently failed components also had the highest functional test frequency. This includes valves and instrumentation. This supports the findings discussed previously which identified wear due to operation as an important failure mechanism. Since RHR is predominantly maintained in standby, testing could contribute to a significant amount of the operating wear experienced by RHR components.

To investigate the existence of aging trends, the failure data were normalized to account for component population, and time-dependent failure rates for several RHR components were calculated. Using statistical techniques, it was found that plants may not all have the same failure rate at a specific age. Some plants were found to have failure rates which were very different from industry averages. This is significant since it confirms the fact that generic failure rates have limitations which must be considered when they are used. Although they are good representations of industry averages, they may not be representative of conditions at a specific plant. Actual plant failure rates may be higher or lower than the generic values.

For comparison purposes, time-dependent failure rates for several RHR components were calculated using average values. Relatively good agreement (order of magnitude) was found between initial (early age) failure rates and those used in industry. It was also found that mechanical components, such as valves and pumps, show a low to medium increase in failure rate with age. Failure rates for these components showed a moderate increase (8% to 17% per year) as the components age. Electrical components showed little or no increase in failure rate with age (0 to 3% per year). It should be noted that these are generalized results which could vary from plant-to-plant depending on maintenance and monitoring practices. It should also be noted that the failure rates are based on NPRDS data and are probably lower than actual due to under reporting to the data base.

9.5 Evaluation of Plant Specific Data

Plant specific data were obtained and reviewed for this study as a means of validating and supplementing the data base information. Results were consistent with data base findings and showed a large aging fraction for RHR failures, with valves and instrumentation/controls the most frequently failed components. Normal service was again found to be the predominant cause of failure.

Plant specific failure rates for several components were calculated from the data and compared to the generic curves generated from data base results. Relatively good agreement was found, however, the plant data did indicate higher failure rates in some cases. This supports previous findings that plants may not be accurately represented by generic failure rates for all applications.

9.6 Probabilistic Analysis

To examine the effect of aging on RHR system unavailability, the failure rate curves generated from the data base information were used in computerized PRA models. Results showed that for the LPCI operating mode, miscalibration of instrumentation due to human error was the most significant contributor to unavailability. However, it was found that during later years, MOV failure can become just as important and could possibly overtake miscalibration as the predominant contributor. This is due to the increasing MOV failure rate with age. With the two components contributing the most to system unavailability being aged, the LPCI unavailability was found to increase by a factor of only 1.7 over approximately 50 years (1.1×10^{-4} to 1.9×10^{-4}).

For the shutdown cooling mode, MOVs in critical locations were found to be the most important contributor to unavailability. Particularly important are MOVs in common suction supply lines where a failure could disable all RHR loops. With increasing failure rates due to aging, MOVs become even more important during later years. Unavailability for the SDC mode was found to increase by a factor of 3.5 over approximately a 50 year period (8.1×10^{-3} to 2.8×10^{-2}) when aging of four of the most important components was taken into account.

10. CONCLUSIONS AND RECOMMENDATIONS

10.1 Phase I RHR Aging Study

The results presented in this study represent the first phase of the research required to understand and manage the effects of aging in RHR systems. These findings serve to characterize the aging phenomenon and provide a technical basis upon which future work can be performed. This work has also resulted in the following specific conclusions:

Aging Effects

- Aging has a moderate impact on RHR component failure rates (0 to 17% increase per year) and system unavailability (factor of 2 to 4 increase in 50 years). The fact that aging effects appear to be mitigated to some degree can be attributed to two factors; 1) RHR is a safety system and has relatively stringent testing and monitoring requirements which identify aging degradation before performance is adversely affected, and 2) the RHR system is typically maintained in standby, which minimizes exposure to wear related degradation.

- Preliminary comparisons of unavailability for standby and continuously operating systems has shown that standby systems are potentially less severely affected by aging. Using this work as a basis, the differences in operation and management of these two types of systems will be further evaluated with the ultimate goal of developing methods that are effective in mitigating aging effects.

- Examination of plant specific failure data has shown that many plants have failure trends for certain components which differ from industry averages. Although aging was found to have a moderate impact on the RHR system based on average values, the impact on plants which differ from these average values could be significant. This will be addressed in future work.

Data Analysis

- Generic failure rates have limitations which must be considered for certain applications. Although they are good representations for industry averages, they may not accurately represent conditions at specific plants. It may, therefore, be possible to reduce the uncertainty in plant specific risk estimates by updating calculations using actual plant data.

- Mechanical components in the RHR system show a low to moderate increase in failure rate with age. This can be attributed to their minimal use in this standby system. Electrical components show little or no increase in failure rate with age.

Design Considerations

- Current shutdown cooling mode designs include one common suction line to supply all RHR loops. Failure of an isolation MOV in this line can cause temporary loss of SDC capability. Since MOV failure rates have been seen to increase with age, MOVs in this location should receive increased attention in later years.

- Several RHR designs use a common minimum flow line for two pumps. Since pump performance can degrade with age, simultaneous operation of both pumps could result in dead headed operation and, therefore, damage to one of them. Pump relative performance should, therefore, be closely monitored and trended to identify performance degradation due to aging and allow timely corrective actions to be taken to preclude this from occurring.

10.2 Future Work for Phase II RHR Study

The results from this study provide the framework for future phase II work to be performed. Although the time-dependent aging effects appear to be mitigated to some extent for this standby system, additional work is necessary to complete the aging assessment. Since RHR is predominantly a standby system, exposure to operating stresses is limited which could contribute to the mitigation of aging effects. However, as plants continue to age and operating time increases, the RHR system could experience rapid increases in failure rates, as was found in previous work on a continuously operating system. This should be addressed in future work. In addition, the relatively stringent tests and inspections performed for the RHR system may contribute to the moderate aging effects. Future work should, therefore, be performed to determine if the practices which detect and mitigate aging degradation in the RHR system can be identified and adapted for use in other systems.

Future work to be performed under the NPAR program will include a Phase II aging assessment of RHR systems. The Phase II study will be an applications oriented, multiyear program which will be based on the body of data collected and the findings of the Phase I work. Using the aging characteristics identified in this study, the following specific tasks will be performed:

- An in-depth review will be performed of current plant maintenance, monitoring and inspection practices. The review will be based on specific plant information and a utility survey of several plants. It will focus on the identification of strengths and weaknesses in current practices relative to managing the effects of aging.

- Recommendations will be made regarding specific maintenance and monitoring activities which have been found to mitigate aging effects. Functional indicators will also be identified which are effective in detecting performance deviations associated with aging. Statistical techniques will be developed to aggregate these indicators for a more global and timely assessment of aging.

- The recommendations made to improve current practices will be implemented in a plant environment as a pilot study. The aging trends before and after implementation will be monitored and compared to evaluate the effectiveness of the changes made using functional indicators and relevant statistical tests.

- Computer models (e.g., FRANTIC) will be used to evaluate the time-dependent effects of aging on plant risk. Using the results of the pilot studies, the benefit due to improved maintenance and monitoring practices will be evaluated and compared in terms of reduction in plant risk.

The findings of the Phase II RHR aging study will be documented in the form of a NUREG at the completion of the program. During performance of the work, Research Information Letters (RILs) will be issued as important results are obtained. Generic safety issues related to RHR will also be addressed as applicable information becomes available.

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APPENDIX A

PLANT SPECIFIC RHR DESIGN INFORMATION

CONTENTS

TABLES

A.1 BACKGROUND

Residual Heat Removal system designs vary between plants. Since the design of the system is important to understanding the function and performance of various system operating modes, a detailed designed review of G.E. BWR plants was performed. The plant designs reviewed are summarized in Table A.1. This review was useful since it provided a more complete understanding of RHR system characteristics and the variations that can be expected between plants. The reviews were performed using information from the plant's Final Safety Analysis Report (FSAR). The detailed design information obtained from this work is presented in Table A.2.

Table A.1: G.E. BWR Plants Evaluated

1.	Nine Mile 1 & 2	14.	Duane Arnold
2.	Oyster Creek	15.	Fitzpatrick
3.	Millstone 1	16.	Brunswick 1 & 2
4.	Dresden 2 & 3	17.	Shoreham
5.	Peach Bottom 2 & 3	18.	Grand Gulf 1
6.	Monticello	19.	LaSalle 1 & 2
7.	Quad Cities 1 & 2	20.	Clinton
8.	Pilgrim	21.	Fermi 2
9.	Vermont Yankee	22.	Susquehanna 1 & 2
10.	Browns Ferry 1,2,3	23.	Perry
11.	Limerick 1	24.	River Bend
12.	Hatch 1 & 2	25.	WNP 2
13.	Cooper	26.	Hope Creek

Table A-2: BWR Plant Specific Information

	LIMERICK 1	HATCH 1&2	COOPER	DUANE ARNOLD	FITZPATRICK	BRUNSWICK 1&2	SHOREHAM
LPCI							
# Pumps	4	4	4	4	4	44	4
Capacity	10,000GPM @ 20 psid each	77000GPM @ 395 ft TDH each	7700GPM @ 20 psid each	4800GPM @ 20 psid each	7710GPM @ 20 psid each	7700GPM @ 20 psid each	7700GPM @ 20 psid each
Suction	Supp. Pool	Supp. Pool	Supp. Pool	Supp. Pool	Supp. Pool	Supp. Pool	Supp. Pool
Power	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)
Support	EWS-cools pump seals, pump cooler & area coolers	Plant SW - pump seal & area coolers	RBCLCW - area & pump lube oil coolers	ESW-cools area and pump motor coolers	ESW-Pump area & pump coolers	Nuclear SW - room coolers	RBCLCW - pump seal coolers CRAC/RBSVS-area coolers
Shutdown Cooling							
# Pumps	4	4	4	4	4	4	4
# Heat Exchangers	2	2	2	2	2	2	2
Capacity	10,000GPM @ 20 psid each	7700GPM @ 20 psid each	7700GPM @ 20 psid each	4800GPM @ 20 psid each	7710GPM @ 20 psid each	7700GPM @ 20 psid each	7700 GPM @ 20 psid
Power	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)
Support	RHRSW-Cools heat exchangers ESW-Cools pump seals, pump coolers & area coolers	RHRSW-Cools heat exchangers Plant SW-Cools pump seal & area coolers	RHRSW-Cools heat exchangers RBCLCW-Cools area & pump lube oil coolers	RHRSW-Cools heat exchangers ESW-Cools pump seals, motor & room coolers	RHRSW-Cools heat exchangers ESW-Cools pump & area coolers	RHRSW-Cools heat exchangers Nuclear SW-room coolers	SW-Cools heat exchangers RBCLCW-Cools pump seals CRAC/RBSVS-cools room coolers
Diesel Generators	4	5 (2/plant, one shared)	2	2	4	4	3

Table A-2: BWR Plant Specific Information

	LIMERICK 1	HATCH 1&2.	COOPER	DUANE ARNOLD	FITZPATRICK	BRUNSWICK 1&2	SHOREHAM
Support System							
	1) RHRSW - 2 loops. Each loop serves 1 RHR heat exchanger	1) RHRSW-4 50% capacity pumps rated 4000GPM @ 955' TDH each. AC (D.G.). Cools	1) SW-4 pumps. 8000GPM @ 125' TDH each. Cools RBCLCW & D.G. heat exchangers	1) RHRSW-4 50% pumps rated 2400 GPM @ 674' TDH Cools RHR heat exchangers	1) RHRSW-4 50% pumps rated 4000 GPM @ 267' TDH Cools RHR heat exchangers	1) RHRSW-4 50% pumps rated 4000 GPM @ 570' TDG each. Cools RHR heat exchangers	1) S -4 33% pumps rated 8600 GPM @ 64 psig. Cools RHR RBCLCW, RBSVS/ CRAC heat exchangers & D.G.'s.
		2) Plant SW-4 33 1/3% capacity pumps rated 8500 GPM @ 275' TDH each. Cools ECCS pumps, ECCS pump areas & 4 of the 5 D.G.'s.	2) RHRSW Booster Pumps-4 pumps rated 4000GPM @ 800' TDG. 2 loops with 2 pumps/loop. Each loop has 100% capacity Cools RHR heat exchangers.	2) ESW-2 100% pumps rated 1200 GPM @ 170' TDH. Cools ECCS pumps and area coolers and D.G.'s.	2) EST-2 100% pumps rated 3700 GPM @ 168' TDH. Cools ECCS pumps and area coolers and D.G.'s.	2) Nuclear SW-2 100% pumps rated 8000GPM @ 115' TDG. Cools ECCS pumps & area coolers and D.G.'s.	2) RBCLCW-3 50% pumps & 2 100% heat exchangers 1600GPM/pump. Cools ECCS pump seals.

Table A-2: BWR Plant Specific Information

	PEACH BOTTOM 2&3	PILGRIM	MONTICELLO	QUAD CITIES 1&2	VERMONT YANKEE	BROWNS FERRY 1,2,3	Grand Gulf 1
LPCI							
# Pumps	4 (part of RHR)	4 (part of RHR)	4 (part of RHR)	4 (part of RHR)	4 (part of RHR)	4 (part of RHR)	3
Capacity	10,000GPM @ 20 psid each	4800GPM @ 20 psid each	4000GPM @ 20 psid each	4830GPM @ 20 psid each	7200GPM @ 20 psid each	10,000GPM @ 20 psid each	7450GPM @ 20 psid each
Suction	Supp. pool	Supp. pool	Supp. pool	Supp. pool	Supp. pool	Supp. Pool	Supp. Pool
Power	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)a	AC (D.G.)
Support	*	RHRSW cools heat exchanger; RBCLCW cools pumps & area coolers.	RHRSW cools heat exchanger; RBCLCW cools pump seals; ESW cools pump motors.	RHRSW cools heat exchanger..	RHRSW cools heat exchanger; RBCLCW cools pump cooler.	RHRSW cools heat room cools heat heat exchangers.	Standby SW-cools pump coolers & heat exchanger.
Shutdown Cooling							
# Pumps	4	4	4	4	4	4	3
# Heat Exchangers	4	2	2	2	2	4	2 (only 2 pumps can pass thru heat exchanger)
Capacity	10,000GPM @ 20 psid each	4800GPM @ 20 psid each	4000GPM @ 20 psid each	4830GPM @ 20 psid each	7200GPM @ 20 psid each	10,000GPM @ 20 psid each	7450GPM @ 20 psid each
Power	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)
Support	High press. SW cools heat exchangers.	RBCLCW cools heat exchangers, pump cooler, area cooler.	RHRSW cools heat exchangers, RBCLCW cools pump seals; ESW cools pump motors.	RHRSW cools heat exchangers.	RHRSW cools heat exchangers, RBCLCW cools pump coolers.	RHRSW cools heat exchanger.	Standby SW - cools heat exchanger & pump coolers.

Table A-2: BWR Plant Specific Information (Continued)

	PEACH BOTTOM 2&3	PILGRIM	MONTICELLO	QUAD CITIES 1&2	VERMONT YANKEE	BROWNS FERRY 1,2,3	Grand Gulf 1
LPCI							
# Pumps	4 (part of RHR)	4 (part of RHR)	4 (part of RHR)	4 (part of RHR)	4 (part of RHR)	4 (part of RHR)	3
Capacity	10,000GPM @ 20 psid each	4800GPM @ 20 psid each	4000GPM @ 20 psid each	4830GPM @ 20 psid each	7200GPM @ 20 psid each	10,000GPM @ 20 psid each	7450GPM @ 20 psid each
Suction	Supp. pool	Supp. pool	Supp. pool	Supp. pool	Supp. pool	Supp. Pool	Supp. Pool
Power	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)a	AC (D.G.)
Support	*	RHRSW cools heat exchanger; RBCLCW cools pumps & area coolers.	RHRSW cools heat exchanger; RBCLCW cools pump seals; ESW cools pump motors.	RHRSW cools heat exchanger..	RHRSW cools heat exchanger; RBCLCW cools pump cooler.	RHRSW cools heat room cools heat heat exchangers.	Standby SW-cools pump coolers & heat exchanger.
Shutdown Cooling							
# Pumps	4	4	4	4	4	4	3
# Heat Exchangers	4	2	2	2	2	4	2 (only 2 pumps can pass thru heat exchanger)
Capacity	10,000GPM @ 20 psid each	4800GPM @ 20 psid each	4000GPM @ 20 psid each	4830GPM @ 20 psid each	7200GPM @ 20 psid each	10,000GPM @ 20 psid each	7450GPM @ 20 psid each
Power	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)
Support	High press. SW cools heat exchangers.	RBCLCW cools heat exchangers, pump cooler, area cooler.	RHRSW cools heat exchangers, RBCLCW cools pump seals; ESW cools pump motors.	RHRSW cools heat exchangers.	RHRSW cools heat exchangers, RBCLCW cools pump coolers.	RHRSW cools heat exchanger.	Standby SW - cools heat exchanger & pump coolers.

Table A-2: BWR Plant Specific Information (Continued)

	NINE MILE PT 1	NINE MILE PT 2	LASALLE 1&2	OYSTER CREEK	MILLSTONE 1	DRESDEN 2&3
LPCI						
# Pumps	None	3	3	None	4 (33%)	4 (33%)
Capacity	N/A	5050GPM @ 20 psid each	7450GPM @ 20 psid each	N/A	7500GPM @ 165 psid for 3 of 4 pumps.	8000GPM @ 200 psid for 3 of 4 pumps.
Suction	N/A	Supp. Pool	Supp. Pool	N/A	Supp. Pool	Supp. Pool
Power		AC (D.G.)	AC (D.G.)		Gas Turbine or EDG.	EDG-ESW
Support		RBCLCW-Cools pumps & room coolers, SW-cools heat exchanger.	RHRSW-Cools heat exchangers.		ESW cools LPCI heat exchanger.	Cools LPCI heat exchanger.
Shutdown Cooling	(Not safety related)			(Not safety related)		
# Pumps	3	3	3	3	2	3
# Heat Exchangers	3	2(only 2 pumps can pass water thru heat exch.)	2(only 2 pumps can pass water thru heat exch.)	3	2	3
Capacity		5050GPM @ 20 psid each	7450GPM @ 20 psid each		2900GPM/pump	6750GPM/pump
Power		AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)	AC (D.G.)
Support	RBCCW-3 pumps 3 heat exchangers, 4500GPM/pump	RBCLCW-Cools pumps & room coolers, SW-cools heat exchangers.	RHRSW-Cools heat exchangers.	RBCLCW-1 main & 1 booster pump 3400GPM	RBCLCW-Cools heat exchangers	RBCLCW-Cools pumps and heat exchangers.

Table A-2: BWR Plant Specific Information (Continued)

	NINE MILE PT 1	NINE MILE PT 2	LASALLE 1&2	OYSTER CREEK	MILLSTONE 1	DRESDEN 2&3
Diesel Generators		3	5 (2/plant, 1 shared)			
Support Systems		1) SW-Four pumps rated 12,500 GPM @ 295' TDH each. Cools RHR & RBCLCW heat exchangers & D.G.'s.	1)RHR SW-4 pumps rated 7400GPM each Cools RHR heat exchangers.			

APPENDIX B

COMPONENT POPULATION ESTIMATES

CONTENTS

TABLES

B.1 DISCUSSION

The data analysis performed for this study included the calculation of time-dependent failure rates for several RHR components. As input to these calculations, estimates were made for component population, operating hours and operating demands. This appendix describes the rationale used in making these estimates and presents the results.

Population estimates for mechanical components were made based on the review of all plant designs from Final Safety Analysis Reports. Each plant was categorized into one of six RHR designs, as shown in Table 2.1. Population estimates were then made for each design. Results are presented in Table B.1.

Since not all plants reported to NPRDS for their entire life, the ages during which a plant's component populations were reportable was accounted for.This was done by estimating the age at which the plant first started reporting. The zero age for each plant was assumed to be the date of initial criticality. The date at which the plant first started reporting was taken to be the earliest data start date reported in the NPRDS records for the plant. Using these two dates, the age of the plant when it first started reporting was calculated. Since the data for this study were obtained at the end of 1986, the plant age contributions included the plant age when reporting began and all subsequent ages through 1986. The plant initial criticality dates and the date at which NPRDS reporting began are included in Table B.2.

Table B.1 Population Estiamtes for RHR Mechanical Components

Plant	Component Population		
	MOV's	Pumps	HX's
Duane Arnold	22	4	2
Browns Ferry 1	22	4	4
Browns Ferry 2	22	4	4
Browns Ferry 3	22	4	4
Brunswick 1	22	4	4
Brunswick 2	22	4	2
Cooper	22	4	2
Dresden 2	29	6*	4
Dresden 3	29	6*	4
Fermi 2	22	4	2
Fitzpatrick	22	4	2
Hatch 1	22	4	2
Hatch 2	22	4	2
Hope Creek	22	4	2
La Salle 1	15	3	2
La Salle 2	15	3	2
Limerick	22	4	2
Millstone 1	29	6*	4
Monticello	22	4	2
Nine Mile Point 1	15	3	2
Oyster Creek	15	3	3
Grand Gulf	15	3	2
WNP2	15	3	2
Peach Bottom 2	22	4	4
Peach Bottom 3	22	4	4
Pilgrim	22	4	2
Quad Cities 1	22	4	2
Quad Cities 2	22	4	2
Susquehanna 1	22	4	2
Susquehanna 2	22	4	2
Vermont Yankee	28	4	2

*Four pumps for LPCI and two separate pumps for SDC.

Table B.2 Age Contributions for BWR Plants

Plant	Initial Criticality	NPRDS Start Date	Age Contributions
Duane Arnold	3/74	12/83	10-13
Browns Ferry 1	8/73	8/74	1-14
Browns Ferry 2	7/74	1/84	10-13
Browns Ferry 3	9/76	3/77	1-11
Brunswick 1	10/76	4/77	1-11
Brunswick 2	3/75	4/76	1-12
Cooper	2/74	1/84	10-13
Dresden 2	1/70	7/74	4-17
Dresden 3	1/71	7/74	3-16
Fermi 2	6/85	3/84	1-2
Fitzpatrick	11/74	10/77	3-13
Hatch 1	9/74	1/80	6-13
Hatch 2	7/78	9/79	1-9
Hope Creek	6/86	6/86	1
LaSalle 1	6/82	1/84	2-5
LaSalle 2	3/84	4/85	1-3
Limerick	12/84	12/84	1-3
Millstone 1	10/70	10/74	4-17
Monticello	12/70	7/74	4-17
Nine Mile Point 1	9/69	7/74	5-18
Oyster Creek	5/69	7/73	4-18
Grand Gulf	8/82	6/82	1-5
WNP2	1/84	10/84	1-3
Peach Bottom 2	9/73	7/74	1-14
Peach Bottom 3	8/74	12/74	1-13
Pilgrim 1	6/72	7/74	2-15
Quad Cities 1	10/71	7/73	2-16
Quad Cities 2	4/72	4/76	4-15
Susquehanna 1	9/82	6/83	1-5
Susquehanna 2	5/84	2/85	1-3
Vermont Yankee	3/72	7/74	3-15

Population estimates for instrumentation were made based on functional requirements for the various RHR modes. The requirements were based on descriptions obtained from an FSAR for a typical RHR design. It was assumed that the populations of pressure and level switches were the same as for their respective sensor. Results are presented in Table B.3.

Table B.3 Population Estimates for RHR Instrumentation

FUNCTION	Pressure Sensor/Switch		Level Sensor/Switch	
	LPCI	SDC	LPCI	SDC
Reactor Vessel Water Level	0	0	4	4
Primary Containment Pressure	4	0	0	0
Reactor Vessel Pressure	4	4	0	0
RHR Pump Discharge Pressure	4	4	0	0
RHR Injection Pressure	2	0	0	0
RHR Flow Orifice	2	2	0	0
Miscellaneous	2	2	0	0
TOTAL	30		8	

Estimates for component operating hours and demands were based on available plant test frequencies and typical operating characteristics for an average plant. The results are as follows:

Pump Operating Hours Per Year:

5 shutdowns/year with each requiring 6 days of SDC using 2 pumps	=	1440 pump-hours
5 days/year of suppression pooling using 2 pump	=	240 pump-hours
5 days/year of testing using 2 pump	=	240 pump-hours
TOTAL	=	1920 pump-hours
For typical system with 4 pumps $\frac{1920}{4}$	=	480 hours/pump

MOV Demands Per Year:

12 tests/year requiring 2 demands/year	=	24 demands
5 operations/year requiring 2 demands each	=	10 demands
TOTAL		34 demands/MOV

Heat Exchanger Operating Hours Per Year:

5 shutdowns/year each requiring 5 days of SDC using 1 HX	=	600 hours/HX

Pressure Sensor/Switch Demands Per Year:

12 tests/year each requiring 1 demand	=	12 demands
8 operations/year each requiring 1 demand	=	8 demands
TOTAL		20 demands

Level Sensor/Switch Demands Per Year:

12 tests/year each requiring 1 demand	=	12 demands
8 operations/year each requiring 1 demand	=	8 demands
TOTAL		20 demands

APPENDIX C

STATISTICAL COMPARISON TEST FOR FAILURE DATA

CONTENTS

C.1 INTRODUCTION

In the calculation of time-dependent failure rates using failures reported to a national database, the first question which must be addressed is whether there is one common failure rate across all plants. If there is only one rate, the data from all plants may be combined to form a sufficiently large sample and a generic failure rate curve can be generated. The following sections describe the statistical test used to compare failure data between plants.

C.2 DESCRIPTION OF STATISTICAL TEST

The null hypothesis to be tested is that all plants have a common failure rate. In order to test this hypothesis, the following procedure can be used

Suppose plant i ($i=1,\ldots,I$) has n_i pumps, and the sum of the pump failures in that plant over a period of length t is x_i. We know $(n_1,\ldots,n_I)$ and $(x_1,\ldots x_I)$. We assume that each pump in plant i fails according to a Poisson process with rate μ_i per unit time, and all pumps in plant i are independent, so that the pdf of the sum of the failures in plant i is:

$$Pr[X_i=x_i] = \exp(-n_i\mu_i t)(n_i\mu_i t)x_i/x_i!$$

Supposing the various plants to be also independent, it is easy to show that the Generalized Likelihood Ratio test statistic for:

H_0: $\mu_1 = \ldots = \mu_I$

versus

H_1: at least one $\mu_i \neq \mu_j$

is:

$$T = 2 \Sigma x_i \ln (\bar{x}_i/\bar{x})$$

where

$$\bar{x}_i = x_i/n_i$$

and

$$\bar{x} = (\Sigma x_i)/(\Sigma n_i)$$

If the null hypothesis is true, T should have approximately a chi-squared distribution with (I-1) degress of freedom.

To test the accuracy of the chi-squared approximately to the distribution of T, a simulation of 40 plants was performed of which 10 had 1 pump, 10 had 2 pumps, 10 had 3 pumps, and 10 had 4 pumps. The true annual rate was 2. (This is roughly similar to the conditions under which this test will be used.) The simulation, comprising 1000 iterations, shows that the test exhibits some tendency to liberality in the upper tail, rejecting about twice as often as it should. The nominal and simulated upper-tail rejection rates are as follows:

Nominal	Simulated
.05	.118
.04	.100
.03	.075
.02	.050
.01	.024

Based on these results, the test was used at a nominal rate of $\alpha/2$ in order to obtain a true rejection rate of α.

(Note that this test takes place at one moment in time. Suppose two plants have no significant difference in rates at time 1, but do have differences at time 2. It would sound inconsistent to assert that the true rates were the same at time 1 and different at time 2. Instead, it was assumed that the rates at time 1 really are different, but that the test does not have enough power to discover this.)

If the null hypothesis is accepted, the maximum likelihood estimate for the common rate is:

$$\text{est}(\mu) = \bar{x}/t$$

However, if the alternative hypothesis is favored, the rate of each plant must be separately estimated as:

$$\text{est}(\mu_i) = \bar{x_i}/t$$

APPENDIX D

DETAILED DESCRIPTION OF THE PRAAGE-1988 CODE

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FIGURES

TABLES

D.1 INTRODUCTION

This appendix provides a short review of aging theory and how PRA is used to calculate the effects of aging on systems; in particular the RHR system in the SDC and LPCI modes. Next it describes the design, implementation, and operation of the interactive IBM-PC code for performing reliability and importance calculations for these systems. Finally, it describes the necessary steps to use PRAAGE to calculate reliability and importances for any system for which cutsets are available.

D.2 COMPONENT AGING THEORY

Aging is a loosely used term which may refer to nearly any failure process. In this report it refers to the end-of-life region of the wearout curve (mortality curve) in which the probability of failure is no longer characterized by a constant failure rate. The report Higgins, 1988[12] discussed in some detail the mathematical modeling of the wearout process which may be approximated as a linear increase in the failure rate with time when a component has seen sufficient use to be in the wearout region. The theory of linear aging was presented in Vesely, 1987[13] as being the result of Poisson-distributed assaults on a component until it finally fails. This leads to a failure rate that linearly increases with time starting when the component is new. Higgins, 1988, using nuclear power plant experience data, showed that certain classes of components such as pumps and valves do not show such a simple dependence but show a basic dependence characterized by two consecutive linear

dependencies. The first segment is a constant failure rate i.e. the failure rate is independent of time; the second segment is a continuation of the first but with a discontinuity in slope in which the failure rate linearly increases. Both the rate of increase and the location in time at which the break occurs are characteristic of the component. Because this subject was discussed in Higgins, 1988, it will be discussed no further.

D.3 SYSTEM AGING

The aging of systems, in most cases, is not just the summation of the aging of the components. If a system is redundant, then systems age at a rate that is the train aging rate raised to the power of the redundancy. For example, a system composed of three redundant trains each of which ages at a rate of 10%/year will age 30%/year. (This system effect is discussed further in Higgins, 1988). Because systems, not individual components, are needed to protect the public safety, aging analysis must be performed in a system context.

D.4 PROBABILISTIC RISK ASSESSMENT (PRA) APPLIED TO AGING

The first major modeling of two nuclear power plants, their systems and components to determine their risk and the reliability of the safety systems was WASH-1400. This was followed by further PRA methods development, PRA applications to regulatory issues and PRAs for many power plants. (Fullwood and Hall, 1988, provides a review of PRA development in this period.) A major development in improving the quality of PRAs and extention to additional plants was the NUREG-1150 study.

The Reactor Risk Reference Document, NUREG-1150 provides the results of major risk analyses to five different US plants (Surry, Zion, Sequoyah, Peach Bottom and Grand Gulf) using state-of-the-art methods. This work provides a data base and insights to be used for a number of regulatory applications: 1) Implementation of the NRC Severe Accident Policy Statement, 2) implementation of NRC Safety Goal Policy, 3) consideration of the NRC Backfit Rule, 4) evaluation and possible revision of regulations or regulatory requirements for emergency preparedness, plant siting, and equipment qualification, and 5) establishment of risk-oriented priorities for allocating agency resources.

The work presented here is a further application of the NUREG-1150 work by applying these system models to the investigation of aging in the residual heat removal system (RHR) at Peach Bottom -the oldest of the plants analyzed in NUREG-1150. Because of the quality of the PRA work in the NUREG-1150 models, one of the ground rules for the work presented in this document was to accept the models without modification. This aging systems analysis begins by accepting the NUREG-1150 models in the form of cutsets (Fullwood and Hall, 1988). The development of the PRA models is described in NUREG/CR-4450 and the cutsets were obtained on magnetic tape from the Sandia National Laboratory, the primary contractor for the NUREG-1150 studies. The probability data base is also obtained on magnetic tape and is used as the default data when nuclear power plant experience data showing aging effects was not obtained in the previously described data gathering activities. The models and data are incorporated into the interactive IBM-PC code PRAAGE for the study of system aging.

D.5 GENERAL APPLICABILITY OF PRAAGE88 TO OTHER SYSTEMS AND PLANTS

Higgins, 1988, used an early version of PRAAGE (PRAAGE87) for modeling the CCW system at Indian Point based on the Indian Point Probabilistic Safety Study (IPPSS). IPPSS was prepared by Pickart Lowe and Garrett Inc. (PL&G) in a style unique to this company. The cutsets were published in the report but not available in a computer-compatible form. One of the problems faced with implementing PRAs in PCs is exceeding the available memory. PRAAGE87 addressed this problem by discovering symmetries in the cutsets that made their representation in a unique, compact matrix format possible. While this worked well for the particular case of modeling the CCW at Indian Point, it is not generally applicable.

An objective in the design of PRAAGE88 was to design a code that could, with minor modification, provide a interactive living model based on cutset input like that available from the NUREG-1150 study. Such a code would avoid major rewrites for analyzing all plants for which cutsets can be obtained.

In order to accomplish this objective, memory limitations of most IBM-PCs at 640 kbytes must be considered. Since an aging model is essentially a "snap-shot" of the system reliability at six time periods (chosen to new, 2, 5, 10, 20, and 50 years), in a sense it is necessary to have 6, not one, PRA models in computer memory. These memory limitations were circumvented primarily by using a series of codes used to reduce the models to the essentials. For example, the data base that was obtained was for the whole RHR system whereas the aging analysis performed is for two modes of the RHR; namely SDC and LPCI, and memory savings can be achieved by removing data that is not

used. Furthermore, the NUREG-1150 format uses up to 16 characters for identification of each component. PRAAGE88 achieves considerable memory savings by reducing the component identification to an indexed variable, which also facilitates the computations.

D.6 DESIGN CRITERIA

PRAAGE88 was designed on the bases of the experience with PRAAGE87 with further enhancements to accomplish the current aging task efforts. The principal criteria were:

- Perform an accurate analysis of the affects of aging on the Peach Bottom RHR system,

- Accept aging data in the bilinear form found to be necessary as reported in Higgins, 1988,

- Include any test and maintenance models that are developed in the data modeling,

- Be easily converted to analyzing the aging of other systems for which cutset and data block information are available. Easy convertability is taken to mean that it can be done in a few hours or less.

- Minimize the manual inputting of data,

- Be operable on all grades of IBM and compatible personal computers using the MDOS operating system with a disk drive and graphics adapter.

- User friendliness by instructing the operator and providing default values which the operator may choose to modify.

- Perform the calculations rapidly enough that the operation may be considered interactive. The current longest computation time is 30 seconds in which the unreliability, normalization and all the necessary importance information is calculated.
- Perform generic groupings of components. This is needed because component specific data are not available and such a large number of components is difficult to manipulate and interpret. Presently PRAAGE is dimensioned for 20 generic components. Generic components may be grouped by any ANDing and/or ORing of the four-element component name identifiers. The search mask for generic component construction is constructed in this fashion to assist the operator and avoid the possibility of typographical errors which would result in no component selection. PRAAGE assumes that the operator will select the generic components of present concern but that these selection may not include all components. Those components omitted are grouped as a "residual" generic component and treated the same as those specifically identified, i.e. subjected to the aging and T&M models as well as probability modification as a group.

- Allow individual component probability modifications.
- Record the parameters used in an aging analysis and the results,
- Print results,
- Provide graphical displays and printed output.

All of these criteria were met with the possible exception of test and maintenance modeling (T&M). This is a very complex problem because the aging data obtained from nuclear power plant experience reflects the effects of T&M that is performed on each component in each plant. To introduce an explicit T&M model into PRAAGE would require that the effects of T&M be removed from the data and a generic T&M model be developed and applied in the system model. To date such has not been done.

Other limitations in current PRAAGE (which may be circumvented if need be) are:

- PRAAGE-1988 operates in the small probability approximation. This means that it will not calculate accurate results if probabilities are set to "1", as is commonly done to simulate a component outage. (This feature was provided in PRAAGE87 but was not used in PRAAGE88 for reasons of calculational simplification. Its inclusion would increase the code complexity and running times.)
- The data input to the cutsets are probabilities - not failure rates. If a component is modeled as failing during a mission time, the failure rate must be multiplied separately by the mission time. (PRAAGE87 identified and accepted both types of data and performed

the necessary multiplication when needed. It will probably be necessary to modify the data representation if the T&M module is implemented.)

- The current maximum sizes are 701 cutsets, 134 components, 20 generic components, and six time steps. These are not the ultimate limits, which have not been explored.

D.7 PRAAGE88 - AN INTERACTIVE CODE FOR SYSTEM AGING ASSESSMENT

D.7.1 Overview of the Code

Figure D.1 shows the computational flow that takes place in PRAAGE. The basic input information is obtained by down-loading a data block and the cutset results from a SETS code analyses of the fault trees representing the RHR mode being studied. The data block contains probability data (not failure rates) for 384 components. The 4 configurations of the RHR system are: LPCI, SDC, RHR and CSS. This work studied LPCI, which has 494 cutsets using 127 components and SDC which has 701 cutsets involving 134 components. Since memory requirements are a paramount concern in personal computer programming, the extraneous data is removed by TRIM.

The new data block containing only data for the components in the cutset block being processed are stored on floppy disk to provide the input to PPOSETS (Post Processor of SETS). PPOSETS converts the component names and the component probabilities into indexed variables, p[i,1], and nam(i), respectively, for array processing. Beginning with the name of the first

selected component, PPOSETS looks at each component name in each cutset. If a match is detected, the cutset is modified by replacing the 16 character name with p[i,j]. This process continues until the whole equation block has been converted into a form using indexed variables. PPOSETS goes a step further and decomposes the original cutset block which is one long equation into many separate equations - one for each cutset. This transformation is performed automatically to produce programming in the Pascal language. These many new equations are stored on floppy disk for reading into LPCIEQ which does the processing of the equations. In this sense PPOSETS is a program that actually writes some of the Pascal programming language used in the LPCIEQ computer program.

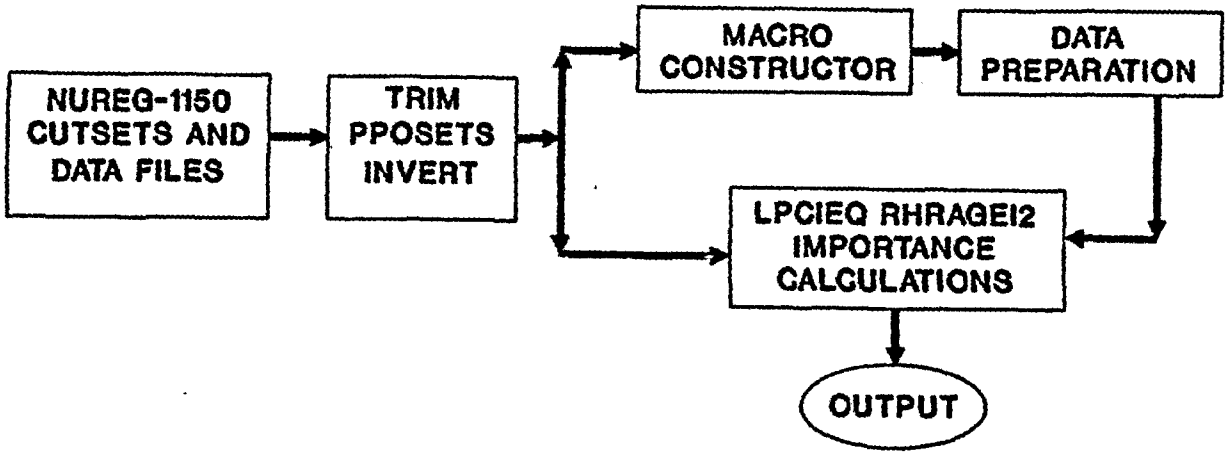

Figure D.1 Computational Flow in PRAAGE

PPOSETS is followed by INVERT which takes the cutset file with the components identified by number and determines the cutsets in which a component appears. The purpose of this is to provide a directory to LPCIEQ for the grouping of cutset values to form the importances.

The cutset transfer to LPCIEQ is done in a very unusual fashion. The equations cannot be read in as string variables because these are programming instructions for LPCIEQ. The manner of entry is to read them into the LPCIEQ program as a block transfer in the editor mode. Further discussion of LPCIEQ will be deferred until completion of the discussion of the macro construction and data preparation.

The macro constructor code (MCROCON1) groups similar components for common treatment. A component name in the NUREG-1150 format is made up of 4 elements or subnames. These four subnames for the component respectively represent the system it is in, the dominant failure cause, the dominant failure mode and a unique identification for the component. MCROCON1 requests a name for the generic component and then lets the operator construct the generic component by ANDing and ORing the contents of each selected column. When the operator indicates completion, all components not selected for one of the generic groups are placed in the "residual" group. This is a fairly lengthy selection process, not to be frequently repeated, so the generic component groupings thereby constructed are saved to disk where they can be reused without having to repeat the generic grouping process. MCROCON1 also offers a simpler assembly process by selecting on first subname which is the system identification. There is good physical reason for the use of system groupings but it is also faster then individually tailoring the groupings and was very convenient for code development.

This menu also provides for a printout of generic component groupings, but this is actually the compressed file used for generic component storage and transfer and is not easy to understand. So far, there has been no reason to make it more user-friendly, but it is retained for its usefulness in code diagnostics. The first number indicates the number of components, next the number of generic components, then by the number of components in each group, by the component identifying numbers in each group and finally by the names of the generic components. If the first column construction method of generic component construction is selected, the name of the generic component is the same as the search mask.

This generic component code is passed to the data preparation code (DATAPREP) which allows the modification of the probability of failure at startup time ($t = 0$, $j = 1$) for each of the components by directly changing the values. The operator may modify the failure probabilities of all of the components in a group by the multiplication of their values by a common multiplier. (A common multiplier produces a proportional change even if the absolute value of each component probability is different). The parameters for each aging model are specified by the analyst in an interactive process. When aging model preparation is selected, the analyst is requested to select a generic component for aging model preparation. Then the analyst is requested to input the time at which aging starts followed by the aging slope in percent per year. This process is repeated until all generic components subject to aging have had their model specified. If no aging model is specified, it is assumed there is no aging (the aging slope is set to zero).

After the aging models have been specified, they are not automatically applied to the time zero probabilities, but the analyst is required to order their incorporation. This is done to allow the analyst a last opportunity to modify the data. However, to perform the aging analysis the models must be implemented, which results in the construction of the component failure probabilities for each time step. (The time steps are 0, 2, 5, 10, 20, and 50 and cannot presently be changed by the analyst without recompiling the code). These time dependent failure rates may be stored on floppy disk for use by LPCIEQ.

LPCIEQ may retrieve the time-dependent probability data as well as the generic component descriptor data from floppy disk or LPCIEQ may access the data from memory. If the latter is the case, it is necessary precede the running of PRAAGE by selecting and running MCROCONI and DATAPREP.

LPCIEQ is rather slow starting (requiring about 30 seconds) because the code is calculating all 494 or 701 cutset equations involving 127 or 134 components for six aging times and executing a complex assembly process to construct the Birnbaum and Inspection importances. Upon completion, the remaining operations are very fast because the code is only grouping importances for the individual components into the generic component groupings and performing the necessary computations for the importance measure selected. When the calculation is complete, the results are automatically displayed. The results can be displayed, printed, graphed or saved to disk. The graphs may also be reproduced on a dot matrix printer.

D.7.2 Operating Instructions for PRAAGE

Table D.1 lists the present contents of the PRAAGE distribution disk. PRAAGE is written in Turbo Pascal 4.0 (TP4 Borland International). This code is considerably different from Turbo Pascal 3.0 which is the language used for PRAAGE87, as reported in Higgins, 1988. A major change in the codes was in discontinuing chaining and overlays in favor of "units". This is done by setting up an executive code, PRAAGE2.exe with a "uses" statement that names the "units" that it uses. Units can also use other units. These are compiled codes designated as "TPU" for Turbo Pascal Unit that contain a public section declaring variables and subroutines accessible to programs that have the units name in the uses statement. When the disk is loaded, the user simply types "PRAAGE". This calls PRAAGE.bat - a batch file which calls "BANNER.exe". This displays a full screen sign stating "Brookhaven National Laboratory presents PRAAGE - PRA applied to Aging". This is followed by a synopsis shown as Figure D.2. A key is pressed when through reading each of these. On the second key press, PRAAGE2.EXE, which is the main program, is called. This immediately calls MAINTITL.tpu presenting the default title for the problem and asking if the analyst wishes to change it. It also requests the disk drive to be used and path. With this information, it presents the main menu shown as Figure D.3. These tasks may be performed in any order, but if they are performed out of sequence they use results stored from previous runs. If it is an entirely new problem they must be run in sequence. (Note TRIM and PPOSETS are not involved. They are used to setup LPCIEQ). If task 1 is selected, the menu shown in Figure D.4 will be presented. Only the first 2 tasks are actually used in generic component definition. By far the most versatile is selection 1. If this is selected, PRAAGE asks for a name for the

generic component. This may contain 12 characters. This is followed by the menu shown in Figure D.5 in which a dialogue has taken place between the operator and the code. To understand the meaning refer to Figure D.6 showing typical names for the components in the model. As stated earlier, the first column is the system designator, the second column is the cause, third is the mode and the last column is the unique identifier. In the dialog shown, the analyst indicated to key on the first subnames. Then PRAAGE displayed all of the first subname designators and the operator selected "1" from this list. PRAAGE responded by saying that ACP (AC power) was selected and asking if this is correct.

Table D.1 List of Codes Comprising the PRAAGE Ensemble
RHR/SDC Mode

Name	Size (kilobytes)	Purpose
PRAAGE2.exe	132608	Main program calling units
MAINTITL.tpu	1664	Global data file
MCROCON1.tpu	12736	Performs generic groupings
DATAPREP.tpu	21456	Data edit, aging and T&M
LPCIEQ.tpu	70608	Computes the cutset equations
RHRAGEI2.tpu	21600	Importance assembly and graph
GENCOMP1	660	Generic component identification
INVBLOK.pas	8491	Inverse file
NAM1PREP.PAS	104	First column component name
NAM2PREP.PAS	121	Second column component name
NAM3PREP.PAS	133	Third column component name

Table D.1 (Continued)

NAM4PREP.PAS	715	Fourth column component name
NAMBLOK	2093	Full component names
DATBLOK	3939	Default time zero data
AGEPDAT1	21802	Aging data file
BANNER.exe	47840	Title and synopsis
PRAAGE.bat	43	Calls BANNER and PRAAGE2

SUMMARY

This version of PRAAGE is specific to the Peach Bottom NUREG-1150 PRA for the RHR System. Its inputs are the data blocks and the cutsets for the RHR in one of 4 configurations: LPCI, RHR, CSS, or SDC. These are converted into equations for each of the cutsets and the operator of PRAAGE may construct generic groupings of the 4 groups of identifiers used in the component names. The failure rates of components in a generic group may be presented for data modification. When the data are as desired, the operator may calculate the Inspection or Birnbaum Importances of the generic groups, of those not in the generic groups or of individual components. The percentage unavailability contribution or the unavailability contribution may be calculated. These results are rank ordered, displayed and may be printed or plotted. The effects of aging are calculated with a bi-linear model by recomputing the cutset equations for each time step. The effects of test and maintenance are incorporated using the fine structure averaged equations as done in the SOCRATES code.

Press a key when through reading

Figure D.2 Synopsis of PRAAGE88

MAIN MENU

Select the Tasks to be performed for:
the NUREG-1150 Peach Bottom RHR/SDC Aging Study

---

1. Define the generic component groupings
2. Modify individual or generic component groups, aging and test and maintenance models
3. Compute, display, print and graph age dependent system unavailability and generic component iomportances
4. Quit

Figure D.3 Main Menu for PRAAGE88

Generic Component Menu

---

1. Construc generic component grouping
2. Construct grouping from first column of component id.
3. Record the constructed groupings
4. Print the constructed groupings
5. Leave the generic component construction

Select the number identifying your job

Figure D.4 Menu for Defining Generic Components

Construction of a Generic Component (x1x,x2x,x3x,x4x)

```
What component i.d. position do you want to key on?

1
1 ACP, 2 CSS, 3 DCP, 4 DGACTA, 5 DGACTB, 6 DGACTC, 7 DGACTD, 8 ECW, 9 EHV
10 ESF, 11 ESW, 12 HSW, 13 IAS, 14 LCI, 15 RBC, 16 RHR, 17 CDC, 18 LOSP,
Select part of the Generic Component Mask
1
You selected "ACP", is that correct? (Y/N)
Do you wish to "and" this with another identifier? (Y/N)
```

Figure D.5 Menu for Forming a Generic Component by ANDing and ORing

```
ACP-RHN-LP-ESWG
ACP-TAC-LP-EDG1
ACP-TAC-LP-EDG2
ACP-TAC-LP-EDG3
ACP-TAC-LP-EDG4
CSS-MOV-MA-MV26A
DCP-BAT-LP-A2
DCP-BAT-LP-B2
DCP-BAT-LP-C2
DCP-BAT-LP-D3
DCP-INV-LP-24C
DCP-INV-LP-24D
DCP-PHN-LP-BATR
DCP-REC-LP-2
DCP-REC-LP-4
DGACTA
DGACTB
DGACTC
```

Figure D.6 Four Subname Component Naming used in NUREG-1150

When the operator replied "Y" for yes, PRAAGE asked the operator if he wished to AND this with another designator. When the operator said "N" (no), PRAAGE displayed the information shown in Figure D.7. After the display, PRAAGE asked the analyst if he wanted to OR the designator with something. If he said yes, then he could perform further ANDing operations to construct a composite mask using these logical operations. In this case, the operator said no and PRAAGE asked if another generic component is to be constructed. If he had said yes the whole process would have been repeated starting with a name for the generic component. Since no was designated, the construction process was ended. Note that only a few components were included in the generic components defined. To avoid losing information, PRAAGE assigns the remaining components to a generic component called "Residual". Thus, the minimum number of generic components is two.

If the operator had selected 2 in the generic menu, the screen would blink and state that the first column construction (i.e. system groupings) is complete.

If the laborious process of building the generic components by AND and OR groupings has been performed, then the operator will probably choose to save these definitions on disk. This is done by selecting item 3 in the generic menu (Figure D.4). If item 4 is selected (print the generic groupings), a rather cryptic print out results. The first number is the number of components, while the second number is the number of generic components. The next number-of-generic component lines provide the decoding of the following string listing the numbers of the individual numbers of the components in the groups. This is followed by the names of the generic components that have been assigned.

Construction of a Generic Component (x1x ,x 2x ,x3x ,x4x)

```
What component i.d. position do you want to key on?

1
1 ACP, 2 CSS, 3 DCP, 4 DGACTA, 5 DGACTB, 6 DGACTC, 7 DGACTD, 8 ECW, 9 EHV
10 ESF, 11 ESW, 12 HSW, 13 IAS, 14 LCI, 15 RBC, 16 RHR, 17 CDC, 18 LOSP,
Select part of the Generic Component Mask
1
You selected "ACP", is that correct? (Y/N)
Do you wish to "and" this with another identifier? (Y/N)

                    Generic Name:  acp Consists of:

Components selected are: 1 ACP-PHN-LP-ESWG
Components selected are: 2 ACP-TAC-LP-EDG1
Components selected are: 3 ACP-TAC-LP-EDG2
Components selected are: 4 ACP-TAC-LP-EDG3
Components selected are: 5 ACP-TAC-LP-EDG4
Components selected are: 135 ACP-BAC-LP-416A
Components selected are: 136 ACP-BAC-LP-416B
Components selected are: 137 ACP-BAC-LP-416C
Components selected are: 138 ACP-BAC-LP-416D
No. of acp items selected: 9
Do you want to "OR" with other selections as the same generic component?(Y/N)
Do you want to construct another generic component?(Y/N)
```

Figure D.7 Display of the Components Selected by the First Subname, First Designator Mask

Individual and Generic Component Modification Menu

```
1:    Modify components in generic component groupings
2:    Modify individual component probabilities in PRA order
3:    Modify and prepare the aging models
4:    Modify and prepare the test and maintenance models
5:    Implement aging into the probabilities
6:    Implement test and maintenance into the probabilities
7:    Record the time dependent probability data base
8:    Display the time dependent probability data base
9:    Print the time dependent probability data base
10:   Leave the component modification

          Select the number identifying your job
```

Figure D.8 Data Modification Menu

The Generic Component Names Are:

1 ACP	2 CSS	3 DCP
4 DGA	5 ECW	6 EHV
7 ESF	8 ESW	9 HSW
10 IAS	11 LCI	12 RBC
13 RHR	14 SDC	15 LOS

Select number of individual generic component for change
Or type "C" to cycle or type "Q" to quit

Figure D.9 List of Generic Names for Selection

Leaving the menu is executed by selecting "4" from the generic menu which returns to the main menu (Figure D.3). Following the sequence, the operator selects "2" to modify the component probability data. The purpose of this menu (shown in Figure D.8) is not just to edit data and inject the aging or T&M, but it also creates the remaining probabilities for the time steps. If this is not done PRAAGE will fail. If task 1 is selected, the editing is convenient by dealing with the components according to the generic component definitions. This results in the menu shown in Figure D.9 being displayed. This lists the names of the generic components what were previously constructed. The operator can change select generic components for change in whatever order he chooses or, if most of them will be changed, he can select "C" for cycle and it will cycle through the names thereby obviating the need for designating individual names. When a name is selected, the menu in Figure D.10 is displayed showing not only the component name but also the current probability value. If the operator chooses to change these as a group, he enters a multiplier (positive but may be greater or less than one) and PRAAGE responds with a new menu (Figure D.11) displaying the effects of the operator's modification. If the change is wrong, it can be corrected by multiplying by the reciprocal of the previous change.

Generic Component No. 1 Named ACP is Composed of:

1 ACP-PHN-LP-ESWG	1.0E-0001;	2 ACP-TAC-LP-EDG1	2.2E-0002;
3 ACP-TAC-LP-EDG2	2.2E-0002;	4 ACP-TAC-LP-EDG3	2.2E-0002;
5 ACP-TAC-LP-EDG4	2.2E-0002;	126 ACP-BAC-LP-416B	1.1E-0005;
127 ACP-BAC-LP-416C	1.1E-0005;		

Select # and enter new probability in "E" or 0.xx format or "G" for generic multiplier, "Q" to quit, or "N" for next cycle.

Figure D.10 Components in the Generic Component Definition

If it is necessary to change a probability value within a generic grouping, the operator may select "2' from the data modification menu and a listing of components by number, name and probability value is presented as shown in Figure D.12. From this, the operator selects the number of the component for modification. If this is done, PRAAGE repeats the old value and requests a new value in real format, as shown in the menu. If a typo such as a letter is typed, a notice is displayed to retype the number. If integer format is used, no warning is displayed and no change is made. When the values of the un-aged probabilities are as desired, PRAAGE returns to the main data modification menu and the operator designates task 3 to inject the aging models.

Generic Component No. 1 Named ACP is Composed of:

1 ACP-PHN-LP-ESWG	1.0E-0002;
2 ACP-TAC-LP-EDG1	2.2E-0002;
3 ACP-TAC-LP-EDG2	2.2E-0002;
4 ACP-TAC-LP-EDG3	2.2E-0002;
5 ACP-TAC-LP-EDG4	2.2E-0002;
135 ACP-BAC-LP-416A	1.1E-0005;
136 ACP-BAC-LP-416B	1.1E-0005;
137 ACP-BAC-LP-416C	1.1E-0005;
138 ACP-BAC-LP-416D	1.1E-0005;

Select # and enter new probability in "E" or 0.xxx format
or "G" for generic multiplier, "Q" to quit, or "N" for next cycle

Figure D.11 Effects of the Analyst's Change

No	Component	Failure Rate
1	ACP-PHN-LP-ESWG	1.0E-0001
2	ACP-TAC-LP-EDG1	2.2E-0002
3	ACP-TAC-LP-EDG2	2.2E-0002
4	ACP-TAC-LP-EDG3	2.2E-0002
5	ACP-TAC-LP-EDG4	2.2E-0002
6	CSS-MOV-MA-MV26A	8.0E-0004
7	DCP-BAT-LP-A2	1.3E-0003
8	DCP-BAT-LP-B2	1.3E-0003
9	DCP-BAT-LP-C2	1.3E-0003
10	DCP-BAT-LP-C3	1.3E-0003
11	DCP-BAT-LP-D2	1.3E-0003
12	DCP-BAT-LP-D3	1.3E-0003
13	DCP-INV-LP-24C	1.3E-0002
14	DCP-INV-LP-24D	1.3E-0002
15	DCP-PHN-LP-BATR	2.7E-0001
16	DCP-REC-LP-2	5.4E-0004
17	DCP-REC-LP-4	5.4E-0004
18	DGACTA	1.6E-0003
19	DGACTB	1.6E-0003
20	DGACTC	1.6E-0003

Select # and enter new probability in "E" or 0.xx
format No # then next list.

Figure D.12 Individual Components for Probability Modification

This results in the menu shown in Figure D.13 being displayed (this Figure is the composite of considerable dialogue) and the operator is asked to designate a generic component for age modeling. In this case the operator chose item 1 and PRAAGE answered back that ACP was selected and asked for confirmation. PRAAGE then asks for the time that aging begins. The operator responds in real format and PRAAGE repeats the entry so the operator can check it. PRAAGE then asks for the slope of the aging ramp in fractional (not percent) change per year in real format. The operator responds and PRAAGE repeats the response and asks the operator if another aging model is to be constructed for some other generic component. If the answer is no PRAAGE returns to the main data modification menu.

```
The Generic Component Names Are:

1 ACP  2 CSS  3 DCP   4 DGA   5 ECW  6 EHV   7 ESP
8 ESW  9 HSW  10 IAS   11 LCI   12 RBC   13 RHR  14 SDC
15  LOS

Select a Generic Component for Age Modeling or Type Q to Quit
1
You selected No. 1 named ACP
When does the aging ramp begin? (years from startup, x.x)
5.0
What is the slope of the ramp? (fraction/year, x.x)
0.1
You specified start 5.0E-0000 and slope 1.0E-0001
Do you want to prepare another model?
```

Figure D.13 Aging Dialogue

Before leaving the data modification menu, it is essential that aging be implemented into the failure probability data to cause construction of all but the time zero probabilities which come from the data base as modified by the analyst. This is done by selecting tasks 5 and/or 6 in the main data modification menu. If task 7 is selected, the time dependent probability data will be saved to disk under a name of the operator's selection, or a default name may be used.

If the operator wishes to see the data that will be used in PRAAGE, he selects task 8 and a printout results, a sample of which is shown in Figure D.14.

After the printout and return to the main data modification menu, the operator selects "9", returns to the main menu and selects "3" to go to PRAAGE for the importance calculations. After some preliminary questions the main importance menu is presented (Figure D.15). Seven importance measures are displayed for selection. (Percent unavailability contribution per component as done in PRAAGE-1988 is not implemented due to lack of interest). In this case, the operator selected "2" for the Inspection importance. Nearly immediately (since the individual importances were precalculated) the importances are displayed, as shown in Figure D.16. If the operator decides to print out the results, task 8 is selected from the main importance menu. If plotting is desired, task 9 is selected; except that this code is not written at the present time. No provision for saving the results to disk has been made but this could be easily done if desired.

The Age Dependent Probabilities are:

Prob.	No./Initially	2nd Year	5th Year	10th Year	20th Year	50th Year
1	1.0E-0002	1.0E-0002	1.0E-0002	1.5E-0002	2.5E-0002	5.5E-0002
2	2.2E-0002	2.2E-0002	2.2E-0002	3.3E-0002	5.5E-0002	1.2E-0001
3	2.2E-0002	2.2E-0002	2.2E-0002	3.3E-0002	5.5E-0002	1.2E-0001
4	2.2E-0002	2.2E-0002	2.2E-0002	3.3E-0002	5.5E-0002	1.2E-0001
5	2.2E-0002	2.2E-0002	2.2E-0002	3.3E-0002	5.5E-0002	1.2E-0001
6	8.0E-0004	8.0E-0004	8.0E-0004	8.0E-0004	8.0E-0004	8.0E-0004
7	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003
8	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003
9	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003
10	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003
11	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003
12	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003	1.3E-0003
13	1.3E-0002	1.3E-0002	1.3E-0002	1.3E-0002	1.3E-0002	1.3E-0002
14	1.3E-0002	1.3E-0002	1.3E-0002	1.3E-0002	1.3E-0002	1.3E-0002
15	2.7E-0001	2.7E-0001	2.7E-0001	2.7E-0001	2.7E-0001	2.7E-0001

Figure D.14 Sample of the Age Dependent PRAAGE Probability Data

Select Importance Measures for:
the NUREG-1150 Peach Bottom LPCI Aging Study

1	Birnbaum Importance
2	Inspection Importance
3	Percent Unavailability Contribution
4	Unavailability Budget Contribution
5	Vesely-Fussell Importance
6	Risk Achievement Worth Increment
7	Risk Reduction Worth Increment
8	Print Selection
9	Plot Part of Selection
10	Print Component Unavailability Fract.
11	Quit Menu

Figure D.15 Importance Selection Menu

Inspection Importance for
the NUREG-1150 Peach Bottom LPCI Aging Study

The generic component name is: ACP
9.6E-0005 9.6E-0005 9.6E-0005 1.7E-0004 3.5E-0004 1.1E-0003

The generic component name is: DCP
4.8E-0005 4.8E-0005 4.8E-0005 5.1E-0005 5.8E-0005 8.8E-0005

The generic component name is: EHV
2 7E-0005 2.7E-0005 2.7E-0005 4.5E-0005 8.2E-0005 2.0E-0004

The generic component name is: DGA
5 5E-0006 5.5E-0006 5.5E-0006 5.7E-0006 6.3E-0006 8.2E-0006

The generic component name is: ECW
3 9E-0007 3.9E-0007 3.9E-0007 5.0E-0007 8.9E-0007 2.1E-0006

The generic component name is: IAS
9 5E-0009 6.5E-0009 6.5E-0009 1.1E-0008 2.1E-0008 5.0E-0008

The generic component name is: RBC
1 2E-0009 1.2E-0009 1.2E-0009 2.1E-0009 3.9E-0009 9.2E-0009

The system unavailability is:
4 1E-0002 4.1E-0002 4.1E-0002 4.1E-0002 4.1E-0002 4.2E-0002

Figure D.16 Inspection Importances for Some of the Generic Components

D.8 OVERVIEW OF THE PRAAGE88 CODES AND FILES

Referring to Figure D.17, items above the horizontal line are programs, items below the line are files. Items to the left of the dog-leg line are programs and files used in preparing PRAAGE for an aging calculation and are not used once this function has been fulfilled. Items to the right constitute the PRAAGE ensemble of user-friendly, interactive codes that perform the PRAAGE design objective.

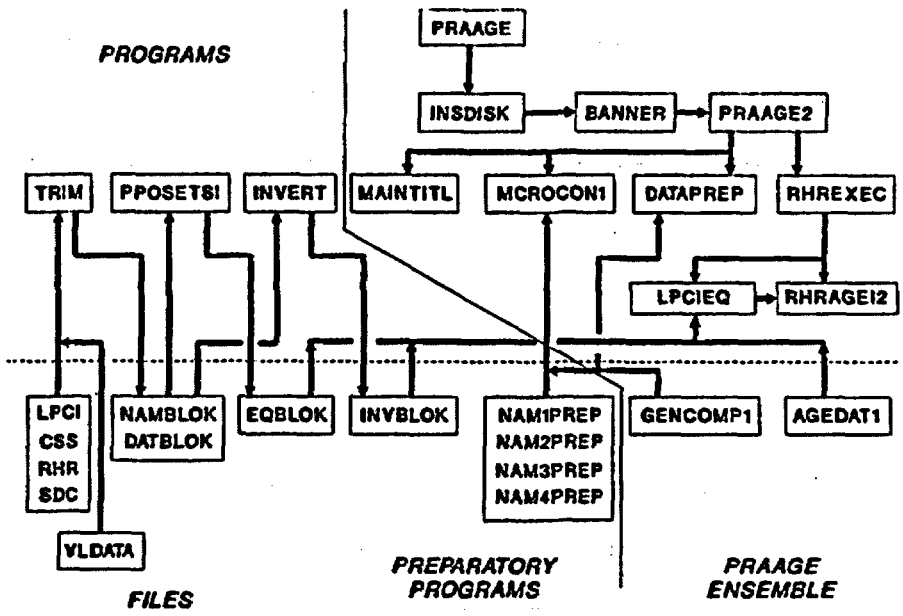

Figure D.17 Taxonomy of PRAAGE88

Figure D.17 presents the taxonomy of codes and files used in preparing and using PRAAGE. To prepare a new systems analysis, begin with the data block identified in the figure as the floppy bedisk file VLDATA (all files are on floppy disks) and select a cutset file from the 4 shown here: LPCI, CSS, RHR, SDC. These are input to trim which writes two files: NAMBLOK and

DATBLOK, containing uniquely, the names of the components and their failure probabilities. These files are input to PPOSETSI to convert the SETS cutset equations into the Pascal programming language used in PRAAGE. NAMBLOK is also input to INVERT to generate the file: INVBLOK which identifies the all of the cutsets containing a given component. The taxonomy of the PRAAGE suite will be discussed with regard to Figure D.17. When PRAAGE is typed in response to a drive prompt, a bat file calls INSDISK. The reason for this is that all of the PRAAGE suite will not fit on one double density disk so the BANNER program was located on a second disk. INSDISK calls BANNER which displays the sign and synopsis and calls PRAAGE2 on the first disk. PRAAGE2 calls MAINTITL to get information regarding the default disk drive and the default problem title and returns to PRAAGE2 for the display of the main menu. From this the analyst selects MCROCON1 to construct the generic component. (MCRO is an abbreviation for "macro" which was used as a name for generic components at one time in the code development.) Or DATAPREP to edit the probabilities or construct aging models or RHREXEC, an executive program that calls LPCIEQ to do the mathematical calculations of the basic importances as well as the groupings of importances for the generic components. RHRAGEI2 calculates secondary importance measures as well as graphically displays results.

D.8.1 TRIM

As just stated, TRIM accepts as input, VLDATA, the SETS data block containing the failure probabilities for all four modes of RHR i.e. for 4 equations blocks. Since only two modes are being studied and these are only

studied one-at-a-time, the extraneous data must be removed to assure the ability to calculate the largest problems that may be encountered.

As each line is read, as a 120 character string, the component name beginning at location 20 and extending for the next 16 characters is copied and assigned to the indexed variable nam[], using a different index for each unique name. An examination of the component names exemplified in Figure D.6, shows they are not of the same size. Since nam[] is to be used for masking, the extra blanks will cause the masking to fail so it is necessary to remove the blanks by the repeat-until loop that searches for and removes blanks. At the same time that the file of component names is being constructed, each probability associated with a component name is recorded and associated with the indexed probability variable p[] in which the same index is used for name identification as is used for the probability of component failure. This is copied as a character string from locations 4 to 12 and converted to a real variable in a "VAL" statement. A Boolean indicator is set for each of the indices, the purpose of which will become apparent. This is the DATBLOK file. After completion of reading the VLDATA file and the operations just described, TRIM begins reading in the SETS cutset file named LPCI or SDC depending on which mode of RHR is being analyzed. This is done by the "while do" repetitive operation which continues until an end of file statement is encountered. A "for-do" statement sets up a loop that causes each of the component names (nam[]) to be compared with each equation line as it is read in.

In the Pascal language, a search is particularly simple because of the string-substring search command - "POS" which not only identifies a substring within a string but tells the beginning location. In this case only the presence is needed and this is indicated by POS > 0. This sets the Boolean indicator to true upon the first detection. Thereafter duplicate names are not copied and the names for which the Boolean exited, it does so with 2 files: NA

D.8.2 PPOSETSI

To recapitulate, PPOSETSI reformats the SETS equation block, which is one big equation, into separate equations for each cutset and formats the results in Pascal coding language.

PPOSETSI (post processor of SETS - indexed) begins by reading in NAMBLOK. (The reason that TRIM and PPOSETS were not combined was to save memory by trimming before much processing takes place.) This is followed by ASSIGN statements for writing the results but the key process begins with the "while do" to the end of file statement. As each line of the equation block is read in, the line is search for each of the names in the NAM[] file when a detection is made, the "IF-THEN-ELSE" statement converts the index of nam[] that produced the match to a string variable (STR statement) and the string beginning with "cut" is produced by concantation. This is an interesting statement because it has the form of a standard "add" statement except it is working with strings. It may be noticed that this concantation ends with a "*" which would result in Pascal's failure to compile the program and it is

necessary to delete it. So after the equation is constructed, the procedure TRMWRIT is called. (The term "procedure" is Pascal-ese for something like a subroutine in FORTRAN). It measures the length of the equation, cuts the "*" off the end and adds a ";" as required for the Pascal language. It should be pointed-out that "cut" had been defined to cause the number identifying the cutset to be written followed by a ":" to produce the format needed for a "case of" statement. The form of the result is:

```
2: eqval:=p[3,j]* p[4,j]* p[125,j] ;
```

where j is the index of the time step for aging. The compactness of this example of a cutset equation shows that much memory is saved by not using the long NUREG-1150 component identifiers. After each call to TRMWRIT, PPOSETSI writes the cutset equation in this compact form. Originally the next processing step was the RHRAGE program but that was for test purposes to see if the code would fit the memory limitations. Once this was assured, the next step was to construct of generic variables because 127 components are too many for the details of the aging data base.

D.8.3 INVERT

INVERT begins by reading in the file IDBLOK from PPOSETS. This file is a list of the component identifiers in the indexed form that appear in each cutset. At the end of the file is the number of cutsets and the number of the components. Since the components are now identified by an index, it is not necessary to read in a file of component names, so the component identification may be made by cycling in an index. INVERT begins by reading in the cut-

set and component numbers to set up the loops. It simply reads in a cutset and tests to see if the component identifier number is present. If it is, it writes the number to a block that was preceded by a component identification number at the top and left justified. The component numbers are right shifted by 2 places to make the decomposition easier in LPCIEQ.

D.8.4 MCROCON1

Prior to the running of MCROCON1, the files NAM1PREP, NAM2PREP, NAM3PREP, and NAM4PREP were prepared manually, using the Norton editor. These consist of the subnames appearing in the first through fourth columns for each component name in NUREG- 1150 naming convention (see Figure D.6). When MCROCON1 begins, it calls these files by a compound CASE-OF statement, assigns a number to the index i and calling procedure READIN(h). This assigns the names NA, NB, NC, and ND to the names in the order of the columns. A similar process is used to readin the NUREG-1150 names according to the nam[] identification. The program enters the repeat-until loop calling the procedure MENU which displays the main menu for MCROCON1.

This menu is displayed and a request is made of the operator for a task identifying number. This number is used in a CASE-OF statement directing the operation to the proper procedure.

If the operator enters a "1" then procedure GENCOMCONST is selected (Pascal allows as long names as desired - except for DOS files when eight character convention must be followed). This procedure sets up the ORing loop for the generic component construction. It calls procedure IMPCONSTR that asks for the name of the generic component which calls procedure MACPRT.

MACPRT request the operator to identify the column in the NUREG-1150 component name from which the generic component will be constructed. When this is done, a CASE-OF statement assigns these subnames to a common variable MSKS. The procedure repeats the selection and asks if ANDing with another identifier is required. If the identification is current and ANDing is not required, the NUREG-1150 names that were selected by this macro are presented and the procedure exits to go from where it was called - IMPCONSTR and from there to GENCOMCONST. If the operator indicates that no more generic components are to be constructed, the procedure GETRESIDUALS is called. When NUREG-1150 names were selected in the generic component selection (procedure MACPRT), every time a name is removed, the label "gone" is put in its place. Procedure GETRESIDUALS scans these names and skips the indices that contain the "gone" label. When this is done, the return is to the main menu through the other calling procedures.

If the operator selects "2" to construct generic components from first column identifiers (systems), the call is to procedure FIRSTCOL. Basically this sets up a loop for cyclically calling each of the first column names and then calling the procedures used in constructing the generic names just described.

If the operator selects "3", the generic name construction is saved to disk in a packed format. The file consists of NONAM (the number of NUREG-1150 names), MACNO (the number of generic names), then gencount[] that provides the beginning of the component identifiers in the packed files, genid[] - the list of the component indices in the packed file and finally the macimprt[] the names assigned to the generic components. This file may be printed although

it is hard to read. Finally selection of "5" (actually any number outside of the range of the CASE-OF statement breaks the initial repeat-until loop and an exit takes place.

While MCROCON1 was written as a stand alone program, it is converted to a unit so the return is to the main program PRAAGE2.exe that called it. From this program, the operator selects the next unit - DATAPREP.

D.8.5 DATAPREP

DATAPREP begins with one of these CASE-OF imbedded calls for file loading to procedure READIN(h) to load the probability data, p[] and the NUREG-1150 component names, nam[] and finally the compressed generic name file just described. It also sets default values for aging if the operator does not provide them - namely, a start at time 50 years and a slope of zero. It enters a repeat-until loop to display the main menu by a call to procedure MENU.

If the operator selects task 1, the call is to procedure GENCOMPONENT for modifying the probabilities in generic component groups. GENCOMPONENT displays the generic component names and asks the operator which one is to be modified or if he wants to cycle through all of them. The selection of a number calls procedure GENLMDA.

GENLMDA causes a call to procedure DISPLAYGENCOMP which displays the NUREG-1150 component names that are contained in the generic component that was selected along with the time zero probabilities (not failure rates). It

instructs the operator to select the number of the component that is to be modified. If this is done, it returns to procedure GENLMDA for accepting the new value. If the operator selected "G" for global, then the call is to procedure GENMULT. This procedure asks the operator to input a real number to be used as a multiplier of the probability data. This is accepted and all of the probabilities defined by the generic component group are multiplied by this factor and the results displayed by a call to procedure DISPLAYGENCOMP.

If the operator selects "C" for cycle in GENCOMPONENT, the call is to procedure CYCLE that causes a cycling through all of the generic groups.

In the main DATAPREP menu, if the operator selects task 2 to modify the data in the PRA order, the call is to procedure LMDA which is a modification of one of the procedures from PRAAGE- 1987. It calls procedure LMDAMNU to display the NUREG-1150 names in the order that the components were numbered in TRIM. It asks the operator to input the number of the component to be changed and then the call is to LMDACNG to display the old number and accept the new entry in real format. Some typing errors are trapped by function NUMCK. This process is continued until the operator comes to the last of the data and then the return is to the main menu.

If the operator selects task 3 - prepare the aging model, the call is to procedure AGING. This procedure presents a list of the generic component names and requests the operator to identify one for the aging model construction. When this is done, the operator is requested to input the time aging

starts and the slope of the aging curve. These are stored as indexed variables for association with their generic component. When all aging models have been entered, the return is to the main menu from which the operator may select task 5 to implement the aging probabilities.

This results in a call to procedure IMPAGING where the "j" index is used that was constructed in PPOSETS. The time steps were defined at the beginning of unit DATAPREP as being times 0, 2, 5, 10, 20, and 50 years. A decoder is used to unpack the generic component packed array and each probability is multiplied by a factor constructed from the linear aging model. The return is to the main menu.

If the operator selects "7" the call is to procedure SAVCOMPVAL where the probabilities as well as the generic component packed file are saved to disk. The selection of task "8" results in a display of the time dependent probabilities and similarly task "9" results in a printout for the time dependent probabilities - 60 to a page. Selection of "10" exits the unit and the return is to PRAAGE2.exe.

From here the operator may select task "3" to calculate importances. This results in a call to unit RHREXEC which, in turn calls LPCIEQ.

D.8.6 LPCIEQ

LPCIEQ begins with reading in DATBLOK containing the probability and generic component data and reading in NAMBLOK containing the NUREG-1150 names. Its first call is to procedure READPDATA which asks the operator if he wants

to choose a file for the generic component definitions and the aging probabilities or accept the default. Whether a new file is selected, the default file or RHRAGE uses the memory access through unit coupling, the procedure ends by a call to SYSTEM-REL which calls procedures SELA and SELB each using the 2 large CASE-OF statements to calculate the probability of each cutset and add them together as a measure of the system reliability. A CASE-OF statement in TPAS4 is limited to about 300 statements so this procedure consists of 2 "for-do" loops breaking at 249 and ending at 494, the number of cutset. These equations are calculated for 6 time steps or 2964 solutions. the return is to the main program for reading in the INVBLOK i.e. the block of data that indicates the cutset probability groupings for form the importance measures. A register is setup and the inverse block is scanned beginning with component 1 using the procedure SELA for the first block of cutset equations and SELB for the second set. This is followed by a loop to add all of the importances together to provide the importance normalization. Next is a call to procedure INS-IMPT followed by a repeat-until loop calling the main menu. The program ends with a loop that adds all of the inspection importances to provide the normalization for the calculation of the absolute and percentage contributions of each of the generic components to the system reliability - two of the secondary importance measures. The return is to RHREXEC.

D.8.7 RHREXEC

RHREXEC contains the procedure IMPTCALC called from the main PRAAGE menu. This procedure calls procedure IMCALC in LPCIEQ to execute the operations described in D.8.6, it calls INS-IMPT also in LPCIEQ to cause the formation of both the Inspection importances and Birnbaum importances for the

generic components by adding the component importance groups. From these, the other 5 importances are calculated. Then it goes into a REPEAT-UNTIL loop that calls procedure MENU causing the display the menu of importance selections and data operations that are available (Figure D.15). Any selection from this menu, calls procedures in the RHRAGEI2 unit.

D.8.8 RHRAGEI2

From the importances menu in RHREXEC, if the operator selects task 1 - Birnbaum Importances, procedure BIRNBAUM in RHRAGEI2 is called. The title indicating the selection is passed to become a generic title. Since the BIRNBAUM importances are explicitely calculated in LPCIEQ as one of the two primary importances, procedure BIRNBAUM transfers the Birnbaum Importances to generic variables and procedure DSPLY is called to display the results (this saves writing a separate display for each importance measure). The data are ranked in descending order using procedure SORT before being displayed. These display techniques are the same as those used for the other importances and this aspect of the description will not be repeated.

If the operator selects task "2" - Inspection importance, a process similar to the Birnbaum calculation and display is performed.

If task "3" - Percent Unavailability Contribution is selected, the call is to procedure PERCENT-UNAVAIL which divides all of the Inspection Importances by the importance normalization factor and multiplies by 100.

A selection of "4" - Unavailability Budget Contribution results in a call to procedure UNAVAIL-BUDGET to divide the Inspection Importances by the normalization and multiply by the system reliability.

Selection of "5" - Vesely-Fussell results in a call to procedure VES-FUS where Inspection Importance is divided by the reliability.

If "6" - Risk Achievement Worth is selected the call is to procedure RAWI to subtract Inspection Importance from Birnbaum Importance.

Selection of Risk Reduction Worth Increment ("7") calls procedure IIMP because this measure is the same as Inspection Importance.

If the operator selects task "8" - Print Selection, the last importance measure that was selected is shown on the monitor.

Selecting task "9" calls the PLOT procedure located in RHRAGEI2. This begins by calling procedure INTRO that displays text explaining that more than about six time-dependent importances can be displayed at one time on the CRT or else there will be too much clutter. It then calls procedure DSPLYGR which displays a menu that looks very much like the importance menu previously displayed from RHREXEC except identifying numbers are presented. From this the analyst selects numbers identifying up to six generic components for plotting. These are stored as the indexed variables selimp[]. PLOT then calls procedure MINMAX to find the range of numbers for calculating a logarithmic ordinate (the abscissa is also logarithmic actually lin-log since it starts from time = zero). Logarithms of all of the numbers to be plotted are also

calculated and procedure DISPLAY is called to do the plotting. Besides plotting the points using different symbols and connecting the points with straight lines, (PRAAGE87 used spline fitting and PRAAGE88 is also set up for spline fitting but not implemented due to lack of request.) Vertical scales are calculated according the the range of the ordinate for each decade and for 2 and 5 times the ordinate. A grid of dots which may be turned off is also drawn according to these divisions as well as similar divisions on the abscissa. Display also associates plotting symbols with the generic component names and presents the problem title. If "p" is selected when the graph is displayed a routine is called to cause the production of a hardcopy of the screen on an Epson dot-matrix printer. This routine is not completely satisfactory because the vertical height is controlled by the screen resolution. Therefore, it is only about 3 inches high if a CGA monitor is used although the width may fill the screen. Pressing any key causes a return to the Importance Menu.

If task 10 is selected from this menu, procedure COMPFRAC is called that prints the normalized Inspection Importance of each component - not each generic component. These are normalized by division by the sum of all Inspection Importances.

The selection of task 11 leaves this menu to go to the main menu and selecting task 4 from this menu results in leaving PRAAGE.

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-5268
BNL-NUREG-52177

2. TITLE AND SUBTITLE

Aging Study of Boiling Water Reactor Residual Heat Removal System

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	1989

4. FIN OR GRANT NUMBER

A-3270

5. AUTHOR(S)

R. Lofaro, M. Subudhi, W. Gunter,
W. Shier, R. Fullwood, J. H. Taylor

6. TYPE OF REPORT

Final

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Brookhaven National Laboratory
Upton, NY 11973

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

A study of the aging effects on Residual Heat Removal (RHR) systems in Boiling Water Reactors (BWRs) was performed as part of the Nuclear Plant Aging Research (NPAR) program. The objectives of the NPAR program are to provide a technical basis for the identification and evaluation of degradation caused by age in nuclear power plant applications. The information from this and other NPAR studies will be used to assess the impact of aging on plant safety and to develop effective mitigating actions.

The effects of aging in the RHR system were characterized using the Aging and Life Extension Assessment Program (ALEAP) Systems Level Plan developed by BNL. Failure data from various national data bases were reviewed and analyzed to identify predominant failure modes, causes and mechanisms. Time-dependent failure frequencies for major components were calculated to identify aging trends. Plant specific information was also reviewed to supplement data base results.

A computer program (PRAAGE-1988) was developed and implemented to model a typical RHR design and perform time-dependent Probabilistic Risk Assessment (PRA) calculations. Time-dependent failure probabilities were input to the PRAAGE program to evaluate the effects of aging on component importance and system unavailability.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

RHR	PRAAGE Computer Code
BWR	PRA
NPAR	time-dependent
ALEAP	Nuclear Plant Aging
	Aging

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)
Unclassified

(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)
*U.S. GOVERNMENT PRINTING OFFICE:1989-241-590:00112