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OPERATING EXPERIENCE AND AGING ASSESSMENT OF COMPONENT COOLING WATER SYSTEMS IN PRESSURIZED WATER REACTORS

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ABSTRACT

An aging assessment of Component Cooling Water (CCW) systems in Pressurized Water Reactors (PWRs) 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 generated will be used to assess the impact of aging on plant safety and to develop effective mitigating actions.

Aging in the CCW system was 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 in CCW systems. Time-dependent failure rates for major components were calculated to identify aging trends. Plant specific data were obtained and evaluated to supplement data base results.

A computer program (PRAAGE) was developed and implemented to model a typical CCW system design and perform Probabilistic Risk Assessment (PRA) calculations. Time-dependent failure rates were input to the program to evaluate the effects of aging on component importance and system unavailability. Changes in component importance and system unavailability with age were observed and are discussed.

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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
DBA	Design Basis Accident
DHR	Decay Heat Removal
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
FI	Functional Indicator
FMEA	Failure Modes and Effect Analysis
FSAR	Final Safety Analysis Report
HPSI	High Pressure Safety Injection
HX	Heat Exchanger
INPO	Institute of Nuclear Power Operations
IPPSS	Indian Point Probabilistic Safety Study
IPRDS	In Plant Reliability Data System
LER	Licensee Event Report
LPSI	Low Pressure Safety Injection
MOV	Motor Operated Valve
NII	Normalized Inspection Importance
NPAR	Nuclear Plant Aging Research
NPP	Nuclear Power Plant
NPRDS	Nuclear Plant Reliability Data System
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RCM	Reliability Centered Maintenance
RCP	Reactor Coolant Pump
RHR	Residual Heat Removal
SW	Service Water
TDH	Total Developed Head

SUMMARY

This report presents an aging assessment of the Component Cooling Water (CCW) system. The CCW system was selected as one of the first systems for analysis under the Nuclear Plant Aging Research (NPAR) program since it is important to plant safety and is vulnerable to aging degradation.

To perform the complex task of analyzing an entire system, the Aging and Life Extension Assessment Program (ALEAP) System Level Plan was developed by Brookhaven National Laboratory. The work presented herein was performed using two parallel work paths, as described in the ALEAP plan. One path used deterministic techniques to assess the impact of aging on CCW system performance, while the second path used probabilistic methods. Results from both paths then were used to characterize aging in the CCW system.

The major conclusions from this work are highlighted in the following paragraphs. Some of the conclusions have application beyond the CCW system.

- This study has identified aging trends in component failure rates, component relative importances and system unavailability that could have adverse impacts on plant safety in later years. Passive components, such as piping and heat exchangers were found to have a potentially significant increase in failures during later years.
- The systems level approach (ALEAP) which uses probabilistic as well as deterministic techniques is an effective method of performing systems level aging analyses.
- Based on the preliminary findings of this study, current PRAs could be underpredicting long-term plant risk.
- Existing national databases are useful for performing aging analyses if appropriate review techniques are employed.
- The more redundant a system is, the faster its relative aging rate, because aging is a common cause effect.
- The majority of CCW failures are not detected until an operational abnormality occurs or until a test is performed.

The following paragraphs provide information from the work which supports these conclusions and summarize other important results.

As part of the deterministic work, CCW system failure data from various national data bases were reviewed and analyzed. The data showed that over 70% of the failures reported were related to aging. The dominant cause of failure was "normal service" while the major mechanism was "wear." These findings show that aging is a significant factor in failures of CCW systems.

Fifty percent of the failures resulted in degraded performance of the CCW system, while 27% caused a loss of redundancy. Complete loss of CCW system function occurred only once and was not related to aging. This shows that CCW failures typically are detected before they become serious enough to cause a complete loss of system function, but not always before system performance has been affected.

To supplement and validate the information obtained from the data bases, maintenance records from the Indian Point-2 (IP-2) nuclear power plant were reviewed. As with the industry wide data, IP-2 had a large percentage of age related failures (80% to 100%), with pumps and valves providing the predominant number. The aging characteristics leading to failure were also found to be similar.

Component failure rates were calculated from the data base information and from the actual plant data. Results showed good agreement with failure rates used in PRA studies. Also, there was a trend toward increasing failure increases with age. It should be noted that current PRA techniques assume constant failure rates and, therefore, predict constant system unavailabilities throughout plant life.

The probabilistic work included the development of a PRA type model based on the IP-2 CCW system, and a PC based computer program called PRAAGE to perform PRA calculations as a function of age. Time-dependent failure rates developed from the data bases were input to the program, and system unavailability and component importance were calculated for various ages.

When the time dependent effects of aging from the data analysis were included in the PRA calculations, two significant results emerged: 1) CCW system unavailability increased with age, and 2) the relative importance of components changed with time. Using the time dependent failure rates calculated from the data, pumps became more important than valves after the first 20 years of plant life because pump failure rate increased more rapidly with age than valve failure rate. Therefore, improvements in maintenance and/or monitoring methods may be required to prevent system unavailability from reaching an unacceptable level during the later years of plant life. More attention may need to be focussed on pumps as they age. However, heat exchangers and piping appear to have the potential to become very important to system unavailability during later years of plant life. This fact should be considered in assessing monitoring and maintenance practices and in evaluating plant life extension.

With the findings presented in this report, the first step in understanding and managing aging in CCW systems is complete. The aging phenomenon has been characterized, and a sound technical basis for future work has been established.

1 INTRODUCTION

1.1 Background

As nuclear power plants age, it becomes increasingly important for the nuclear community to understand and be able to manage aging phenomena. As a first step in addressing the issue of aging, the NRC Office of Nuclear Regulatory Research, Division of Engineering has initiated a comprehensive long-range research program for assessing the 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 generation and safety by understanding and managing aging degradation. Initial work under the NPAR program focused on the evaluation of aging effects for selected plant components[1-8]. As a follow-on to the component level evaluations, plant systems will be studied. The NPAR program is described in detail in NUREG-1144[9].

A system in a Nuclear Power Plant (NPP) performs specific functions and comprises various mechanical and electrical components. The components are located in various buildings and are interconnected by pipes and cables. The components together with the piping and cables, are supported from the various building structures. System functions are diverse and can either aid in generating power, or assure safety of the plant.

The aging assessment of a system is complicated by the fact that, (1) age-related deterioration of components and subcomponents occurs at varying rates, and (2) dynamic interaction of components is inherent in any system design. Besides normal wear, other aging factors which may affect system performance are transients, environmental stresses and human errors.

As nuclear power plants age, the likelihood of common cause failures due to age-related degradation increases. Therefore, steps must be taken to assure that the level of safety on which a plant was designed has not fallen to an unacceptable level.

An assessment at the system level has several advantages over component level studies. The effect of individual components within a system on its overall performance can be assessed. Design redundancies and interfaces with other systems and components can be included to make more objective decisions on their importance. Hence, the priorities in testing, maintaining, and operating the system can be developed or altered as the plant becomes older.

Several studies have identified the component cooling water (CCW) system as important.[10-12] NRC Generic Issue 65 relates to the high probability of core-melt due to CCW system failures. Studies performed for the NPAR program listed CCW as a system that is important and subject to aging degradation. A literature search revealed that very little work has been completed on the CCW system. The recent study by the Electric Power Research Institute (EPRI)[13] concludes that the reliability centered maintenance (RCM) approach can be effectively applied to the CCW system.

This report describes an aging assessment of the CCW system at PWRs, primarily based on design, plant operating experience, and risk assessment. The study also considers accidents such as seismic events, fires, and loss of coolant, and discusses their effects on the performance of the system as the plant ages. Particular emphasis was given to the predominant causes of component degradation and the effects of failure of these components on overall reliability and availability.

1.2 Objectives

In accordance with the NRC-NPAR Program Plan[9], the primary goals of the CCW system study are:

1. 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.

2. To identify methods of inspection, surveillance and monitoring, or of evaluating the residual life of structures, components, and systems, which will assure timely detection of significant aging effects before loss of safety function.

3. To evaluate the effectiveness of storage, maintenance, repair, and replacement practices in mitigating the effects of aging and diminishing the rate and extent of degradation caused by aging and service wear.

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 CCW System

The Component Cooling Water System in pressurized water reactors is a common system used to remove heat from various plant components and transfer it to an open loop cooling system such as Service Water. The CCW system is a non-radioactive, closed-loop cooling water system, which serves as a barrier between radioactive components and the open loop cooling systems. The basic CCW system generally consists of several pumps, heat exchangers, a surge tank, and piping supplying the loads in a variety of header arrangements. Figure 1-1 functionally shows the CCW system.

The CCW system function is sometimes served by 2 or 3 separate systems.will not be counted as CCW system failures. The various interfaces and boundaries are:

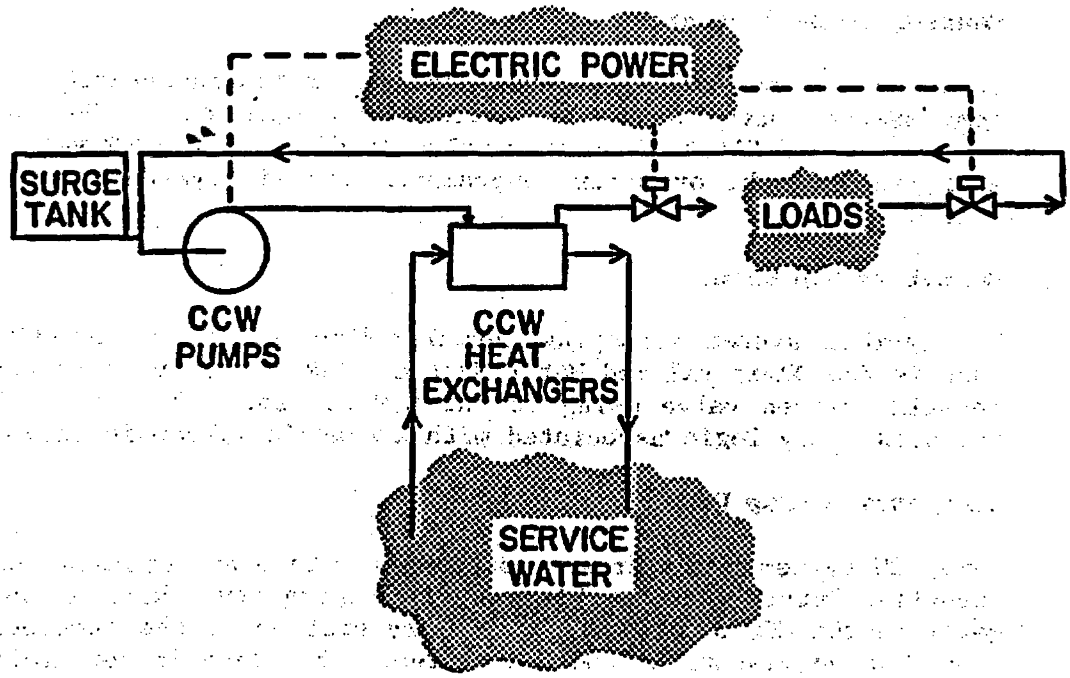

Figure 1-1 Functional Diagram of CCW System

1. **Service Water System (SW)**

This system is the open loop cooling system which cools the CCW heat exchangers. The boundary will be at the point where SW enters the CCW 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." However, "Loss of Service Water" is an important way in which the CCW system itself can fail.

2. **AC Electric Power**

AC electric power at various voltages is needed to supply the CCW pumps, valves, and instrumentation. The boundaries are as follows:

- Power to Pumps and Valves - The boundary will be at the circuit breaker, and will include the breaker and the breaker logic.

- Power to Instrumentation and other CCW Items - The boundary will be at the first circuit breaker or fuse from the equipment, and will include the breaker or fuse.

3. **DC Electric Power**

DC electric power is needed for circuit breaker controls, instrumentation, and control logic. The boundary will be at the first circuit breaker or fuse from the equipment supplied.

4. **Control or Service Air**

Some CCW systems use air-operated valves or instrumentation. Also, some systems have surge tanks pressurized with air. The boundary will be at the first air system valve leading to the CCW valve operator, surge tank, or other component. The air system valve is included.

5. **Normal Makeup Water**

The typical system supplying normal makeup water to the CCW surge tank is the Demineralized Water System. The boundary is at the last normally closed valve going to the CCW system. This valve is not included. Any logic associated with automatic makeup is included.

6. **Emergency Makeup Water**

Some CCW systems have cross-connections with other systems, such as Essential Service Water, which allow an emergency makeup of non-pure water to the CCW system. The boundary will be at the last normally closed valve leading to the CCW system. The valve is not included.

7. **Cross-Connection Between Units**

At dual unit sites, there are usually cross-connections between CCW systems. These vary from completely shared systems to a swing pump or heat exchanger which could be used by either unit, to cross connections only at certain selected loads. Shared systems are treated together. Separate systems with cross-connections will have the boundary at the first normally closed valve.

8. **Systems Drains**

Valves exist which drain the CCW systems to various waste water systems. This boundary is at the first normally closed valve, which is included in the system definition.

9. **Structures and Buildings**

The CCW system usually runs through a large portion of the plant, including the Auxiliary Building and Primary Containment. The system is normally safety related and hence, typically is mounted or supported so that it will withstand seismic shocks. The structures and buildings to which CCW 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.

10. Loads

The loads cooled or supplied by the CCW system vary from plant to plant. Typical loads are:

Typical Safety Related Loads:

Westinghouse:	Residual Heat Removal (RHR) Heat Exchangers (HX), RHR pump seals, Safety Injection (SI) Pumps, Containment Spray Pumps, Containment Coolers.
Combustion Engineering:	Shutdown HXs, Low Pressure SI Pumps, High Pressure SI Pumps, Containment Spray Pumps, Chillers, Containment Air Coolers.
Babcock & Wilcox:	Decay Heat Removal (DHR) HXs, DHR Pumps, High Pressure Injection (HPI) Pumps, Reactor Building Fan Coolers.

Typical Non-Safety Related Loads:

Reactor Coolant Pump motor, Letdown HX, Excess Letdown HX, Seal Water HX, Spent Fuel Pool HX, Charging or Makeup Pumps, Control Rod Drive or Control Element Drive Mechanism Cooling, Miscellaneous Loads.

The CCW piping to and through the loads is included, but the loads or equipment serviced by CCW are not included within the system boundary. However, it is important to keep track of loads, since the effect on the overall plant of loss of CCW or loss of portions of CCW depends on which loads are lost (i.e., not cooled).

1.4 Analysis Methodology

Recognizing that the characterization of aging in a system in a nuclear power plant is a complex task, a system level program plan was developed, entitled "Aging and Life Extension Assessment Program (ALEAP)."[14] 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. Phase I 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 (Task A). Also included in Phase I is a preliminary review of current test, maintenance, and inspection practices (Task B). 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 (Task A) work.

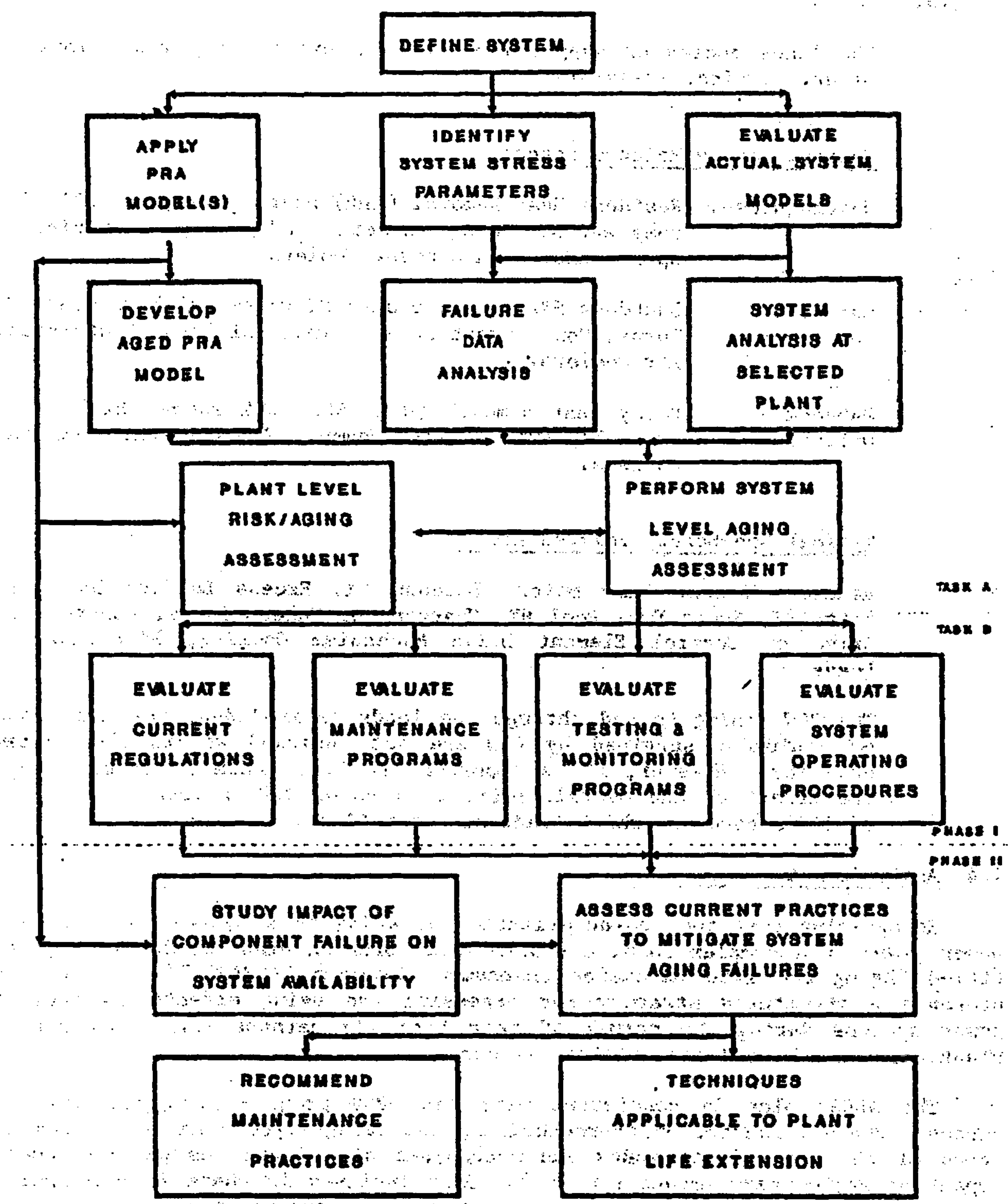

Figure 1-2 Detailed Task Structure of ALEAP System Level Plan

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 CCW system designs in use. The scope of the design review encompassed all operating PWR plants in the United States. A selected number of BWR plants were included to determine their similarity to the PWR systems.

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

- Nuclear Plant Reliability Data System (NPRDS),
- Licensee Event Reports (LER),
- In-plant Reliability Data System (IPRDS),
- Plant Specific Failure Data Bases.

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 importance of particular components and system unavailability as a function of age.

Plant specific data for the system was obtained to supplement the parameters mentioned above for component failure from the individual failure histories. These data were then compared with those from other data bases as a check on the database results. Information also was used from various NPAR component level studies.[5-8]

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 apply aging analyses or to perform a failure mode and effects analysis (FMEA) for all components and subcomponents. Therefore, those predominant 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) were developed to reflect the essential features of the CCW 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.

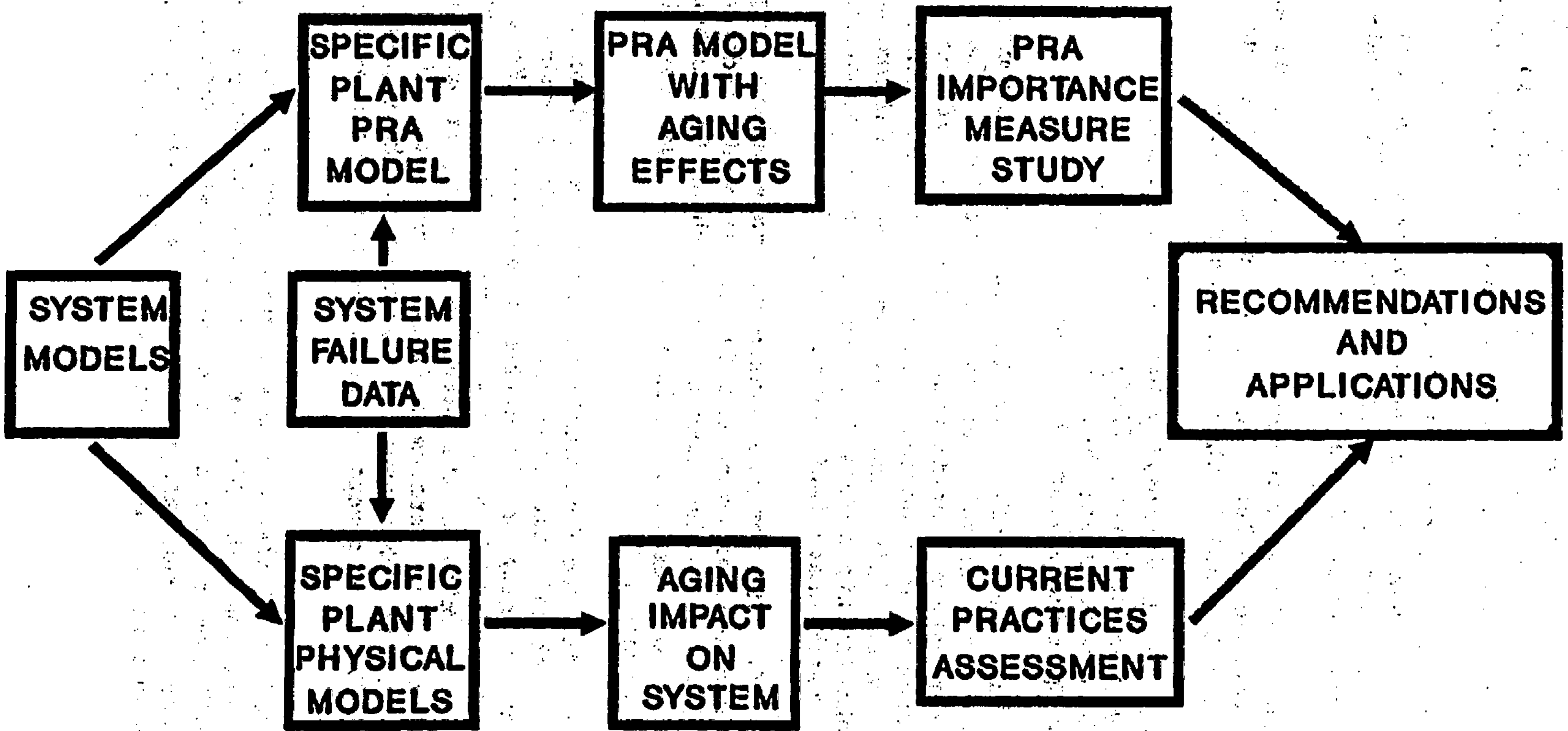

Figure 1-3 Overall Strategy for CCW System Study

Section 2 of this report describes the design review of the CCW system for all PWRs in the United States. The operational stresses and their correlation with accidents are discussed in Section 3. Section 4 provides the results of all data bases and identifies the predominant CCW system failures from the operating experience at nuclear plants. The detailed review of the CCW system at the Indian Point Nuclear Station 2 (IP2) is summarized in Section 5. Section 6 discusses the PRA model of the CCW system at IP2 and applies the statistical data taken from the CCW system operating experience to rank the importance of components within the system. Section 7 discusses the sensitivity studies, while the conclusions of this work are summarized in Section 8. Several appendices give detailed information on the specific areas discussed.

2. CCW DESIGN REVIEW

2.1 Overview

This section of the report describes the design review of all CCW systems in U.S. nuclear power plants. Functionally, the CCW system is quite straightforward, however, the details of the design can be complex and vary considerably between plants. This review was performed to determine the extent to which results of this study could be generalized. Information also was needed: to fully understand the system's design and operation; to provide insights for analysis of the failure data; to aid in later determination of appropriate system functional indicators; to determine the effects of variations in system design on its reliability and availability; to assure the applicability of the work to all the PWRs in the United States; and, to provide the population data necessary for normalization of the failure data. The CCW design information was obtained primarily from the Final Safety Analysis Report (FSAR) for each plant.

The design of each plant's system was cataloged and then summary analyses were performed to determine the overall status of CCW system design in United States PWRs. Appendix A presents the type of review performed and the results of the summary analyses, along with charts and tables.

The basic functional design and arrangement of CCW systems was found to be the same at all plants, although individual design details varied considerably. The few unique arrangements are discussed in Appendix A.

2.2 Typical Design

Figure 2-1 shows a typical CCW system design for one unit which has three CCW pumps arranged in parallel. The pumps are motor-driven centrifugal pumps powered from Class 1E buses. Downstream of the pumps are pressure gauges for monitoring pump operation and two parallel heat exchangers (HXs) cooled by the Plant Service Water System. Downstream of the HXs are temperature detectors and flow meters.

Normal operation requires two pumps and one heat exchanger. The valves in the cross-connect line downstream of the HXs normally would be left open, so that one operating HX could supply flow to all three loops of loads. If there is an accident and an Engineered Safety Feature (ESF) actuation occurs, the headers would be split by automatic closure of the motor-operated valves (MOVs). In this situation, the non-safety related loads are tripped and their CCW flow is secured. The Train I and the Train II safety-related headers are isolated from each other and one pump and one HX are aligned to supply each header.

There is one surge tank which is internally baffled to provide a separate water volume for two CCW pump trains. Thus, should a leak occur in the surge tank, only one train of CCW would be lost. The surge tank provides net positive suction head (NPSH) for the CCW pumps and a surge volume for the overall CCW system. It is normally vented to atmosphere through a valved line. If high radiation is detected in the system (e.g., due to a leak at a radioactive load) an alarm is sounded and the vent valve will automatically close. In this mode the CCW System provides two trains of reliable cooling to the safety related reactor plant loads.

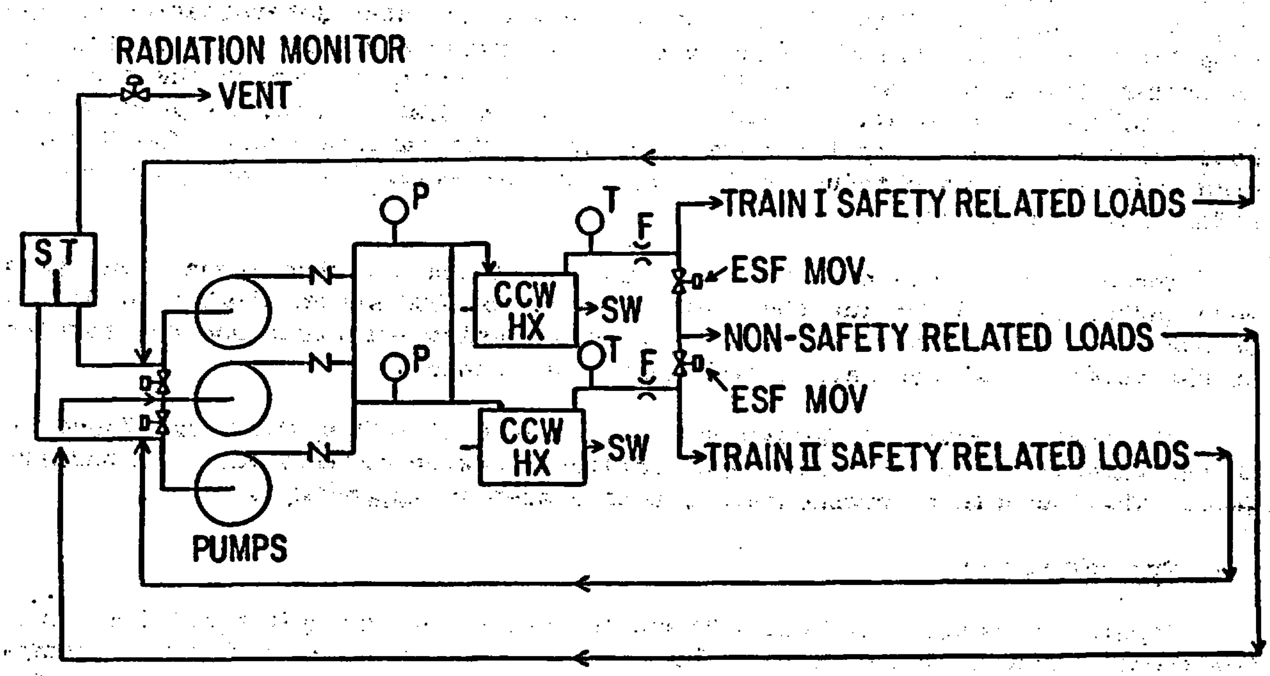

Figure 2-1 Typical CCW System

2.3 Design Variations

Using the typical design shown in Figure 2-1, a few of the major design variations will be discussed here. These are cataloged and discussed in more detail in Appendix A.

The overall layout or arrangement of the system varies: some plants use two or three systems rather than one to fulfill CCW functions; some dual unit sites use a shared system, while other such sites have separate systems but with selective cross connect features; some plants have emergency back-up water supplies to CCW; and there are a number of different pipe header arrangements. The number of pumps, heat exchangers, and surge tanks also varies. The actual loads supplied by CCW vary somewhat as well. Table 2-1 summarizes the design variations encountered and lists some typical operating parameters for CCW systems. Charts of these variations are included in Appendix A, as Figures A-8 through A-11. Additionally, the appendix discusses the advantages and disadvantages of the different designs.

Table 2-1 CCW Design Variations and Operating Parameters

Number of Pumps:	2 to 8
Pump Flow:	25 gpm to 17,500 gpm per pump
Pump Head:	54' to 275' (Total Developed Head)
Number of HXs:	1 1/2* to 8
Number of Surge Tanks:	1/2* to 4
Operating Pressure:	<100 psig
Operating Temperature:	75°F to 150°F
Working Fluid:	Purified water with corrosion inhibitors
Electrical Power:	4160 volt or 480 volt AC for pumps 480 volt AC for MOVs 125 volt DC and 15 volt AC for instrumentation and control

* 1/2 indicates 1 component shared between 2 units.

2.4 Conclusions

As a result of the design review, a number of insights have been obtained, described in Appendix A, Section A.3, which relate to shared systems, multiple systems, header arrangements, and components. The information gleaned was useful in the failure analyses that were performed. It is concluded that Indian Point-2 is acceptable as a representative plant to use as a baseline in the PRA-type system analysis and for the analysis of plant specific experience. The results of this study are applicable to other individual plants, however, specific differences in design in key areas must be considered. For example, when establishing programs for their individual CCW system, a plant with more than one surge tank would place less importance on them than Indian Point-2, which has only one. Also, different pipe header arrangements that increase redundancy of the cooling water supply, would decrease the importance of certain critical supply valves. These issues will be covered in more detail in later sections of this report.

3. OPERATIONAL STRESSES AND CORRELATION WITH ACCIDENT SCENARIOS

Aging degradation occurs when a material is subjected or exposed to a stress condition for a period of time. 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 CCW system. These parameters include system and component level stresses such as those induced by testing, human factors, environmental parameters and their synergistic effects. They also included external loads imposed on the system by earthquakes, floods, and fires. The correlation with accident conditions when the function of the CCW system becomes vital for plant safe shutdown also is discussed.

3.1 System and Component Level Stresses

During normal operation, the component cooling water system accommodates the heat loads from various plant auxiliary components. During an accident, other heat loads are added to the system while some are removed or reduced. The CCW system can accommodate any single failure of an active component and still operate in a manner to avoid undue risk to the public health and plant safety. The CCW system also can detect and isolate radioactivity entering from reactor coolant systems and its auxiliaries.

Since the CCW is a fluid system, the operating condition of the fluid and the external loads govern the level of stress. The normal operating temperature of the fluid ranges between 90°F and 130°F while the operating pressure is less than 100 psig. The pressure boundary components are typically designed for a pressure of 150 psig and a temperature of 200°F. The effects of these relatively low operating conditions on pressure boundary components are minimal. However, normal wear and degradation can cause leaks at the welds in flange connections, and in seals.

The flow rate of fluid in the system depends on the plant design and number of loads being cooled at any one time, which generally varies. Therefore, CCW pump flow capacity varies between plants from 25 gpm to 17,500 gpm. The fluid flow can contribute to erosion of the insides of pipes and pipe fittings. Although the CCW fluid is chemically treated this does not completely eliminate corrosion. The presence of oxygen in the water can cause corrosion of metal surfaces.

The main components of the CCW system and the piping header arrangements usually are located in the auxiliary building. These components are exposed to uncontrolled room conditions with a very low level of radiation. The equipment, therefore, may be exposed to extremes of temperature, humidity and dust. Electrical and mechanical components may also be affected by adverse weather conditions or by humidity and salty air in coastal areas. The loads of the CCW system are located throughout the plant and may be exposed to quite different environments; this includes valves inside the reactor building in high radiation areas.

Table 3-1 summarizes the operational and environmental stresses causing the various CCW components to age. The operational stresses include fluid temperature and pressure, flow, fluid contaminants, flow and machine induced vibrations, and electrical transients caused by breaker trips or degraded voltages. The environmental parameters include temperature, humidity, radiation, dust, and other adverse atmospheric conditions. The components also may be affected by heat dissipation from electrical devices such as transformers, rectifiers, and steam/oil leaks.

3.2 Stresses Induced by Testing

Components are periodically tested to monitor and maintain their condition during the life of the plant. Plant technical specifications require that certain safety-related equipment be tested regularly for operational readiness. If the tests are performed too frequently, unnecessary stresses could be induced in the component. Also, certain tests such as high-potential testing of electric motors or emergency diesel generator tests requiring fast starts, could impart a larger stress than the component is expected to experience normally.

Mechanical tests include vibration, temperature, valve stroking, leak tests and measurements, functional tests, and crack detection tests. Electrical tests are insulation resistance or dielectric strength tests, contact resistance tests, and certain high potential tests. These tests affect mostly contact points, nonmetallic components such as insulating systems, and other electronic devices sensitive to high temperature and humidity. Chemical tests of lubricants to detect wear in the component, might not affect the component.

Certain pieces of safety equipment, such as CCW pumps, remain on standby and are required to become operational anytime the safety of the plant is challenged. The technical specifications may require periodic start/stop testing of this equipment to assure their operational readiness. This requirement could involve cold starts of the equipment, which introduces a higher stress than is usually experienced during normal operation. Similarly, valve tests may subject the valve motor or other moving parts to abnormal stresses. In general, the amount of testing required for CCW systems is less than most standby, safety-related systems so that stresses caused by testing are not expected to be a serious concern.

Table 3-1 Aging Effects on CCW System Components

		Components Affected		
Stress Conditions	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	

3.3 Human Factors/Maintenance

To maintain and operate a system, human activities are involved. These activities can be in design or manufacturing, shipping, installing, operating, maintaining and testing. From past experience, it is evident that human errors contribute to the failure of CCW system components. These failures are attributed to incorrect installations, improper operation leading to overloading of components, and errors in testing and maintenance.

Inadvertent actions, such as stepping on pipe supports or snubbers, have caused damage and failure. Other incidental human errors include inadvertent spraying of water into electrical components, use of the wrong replacement parts, improper tightening of bolts or screws, improper lubrication of moving components, and leaving protective components open to hostile environments. Most human errors are likely to occur during activities such as testing, monitoring, inspection, maintenance or repair of components.

Typical maintenance errors include incorrect calibration of set points, wrong wiring of test equipment and improper adjustment of the test equipment itself. These may result in erroneous test data indicating the wrong state of the equipment condition. Since most equipment is a complex arrangement of many subcomponents, plant maintenance personnel usually follow a particular test and maintenance procedure and perform the activities in a specified order. If properly performed, the equipment will be kept in operating condition. A large number of equipment failures may be due to people using wrong procedures or incorrectly following the set procedure in performing the maintenance. Improper maintenance is indicated where there are repeated failures, for example, leaks in pump seals shortly after initial failure and repair.

An incorrect operating procedure can adversely stress the equipment's subcomponents and may accelerate the aging process. For example, frequent starting of certain electrical equipment before cooling them could age the insulation and cause premature electrical shorts or grounds. Certain equipment such as pumps, valves, and switchgears require a definite sequence of operations (i.e. starts or stops) to run them. These sequences are documented in operating manuals or procedures for reference, and should be properly followed.

3.4 External Effects (Earthquakes, Floods, Fires)

The design of a nuclear facility accounts for the effects of earthquakes, floods (both internal and external), and fires in its design basis accident (DBA) loads. Since the CCW system is vital for plant normal operation and for safe shutdown in an accident, the entire system is designed to withstand these loads specific to the plant site. The seismic design levels for operating and for safe shutdown are determined by the location and the geological survey of the site. External flood is considered in the structural design if there is a potential of failure at nearby dams on a river or reservoir. Internal flood and fire typically are included in the design for locating the equipment inside the plant.

None of these loads contribute directly to the aging process of components or subcomponents. Rather, after being aged under the stresses discussed earlier, the components become more vulnerable to accident loads which could affect the capability of the plant to shutdown safely. Thus, the design margins for the CCW components must account for aging to prevent a common cause failure in the system during an accident.

All components including piping, pumps, motors, heat exchangers, tanks, and electrical cabinets are supported with restraints to withstand earthquakes. The components are also qualified to seismic levels specific to the site. Since seismic loads are transmitted to the equipment via the plant buildings housing the system, seismic damage could affect the entire system.

The three types of supports that are typically used are rigid restraints, spring hangers, and snubbers. The rigid restraints are made out of carbon steel and are vulnerable to corrosion, stress relaxation of bolts, distortion and degradation from fatigue. The spring hangers are not a concern for the seismic load since they support the dead weight of the component. However, changes in the characteristics of the spring caused by corrosion or other mechanical deterioration could alter the overall dynamic characteristics (i.e. natural frequency) of the system and result in increased vulnerability to lower seismic loads than allowed for in the design basis for the component.

Snubbers are the seismic restraints and they must remain operational for the system's life. Since the CCW system is a low temperature system, there should not be many of this type of restraint (systems with small amounts of thermal growth can use rigid restraints in place of snubbers). Hydraulic snubbers are vulnerable to reservoir oil leaks or oil contamination. Mechanical snubbers, are fragile and are easily damaged and distorted by human error. Snubbers which freeze during normal operation and do not allow normal thermal expansion of the system's piping can impose abnormally high stresses on the system.

Relays and circuit breakers are vulnerable to trip during earthquakes. This might lead to a change in the state of operating or standby equipment. Electrical components are primarily vulnerable to high humidity due to steam leaks or water leaks, causing short circuits.

Floods can be either external or internal. Generally the plant is built to withstand any external flooding that is likely in the area. Internal floods are caused by pipe breaks or storage tank failures within the CCW system or other nearby systems. For example, in one nuclear station all three CCW pumps were failed by a pipe break which shorted the pump motors and disabled the entire CCW system.

Fire is always considered to be hazardous to electrical components inside a nuclear plant. Typically, fire barriers are built to protect vital components from such damage. Electrical cables and other nonmetallic components are vulnerable to fire and could cause complete or partial failure of the CCW system. Fire sometimes is caused by burning of cables due to short circuits or electrical heat. Degradation of the cable's insulation increases this vulnerability.

A loss of offsite power should also be considered as an external event which can stress the CCW system, primarily due to the effects of a loss of instrument air. Generally, air compressors for the instruments are not powered from the emergency electrical bus, therefore when the diesel generators provide power to the plant following a loss of offsite power, instrument air will not be available. The air-operated temperature-control valves typically employed in the CCW system for modulating cooling water flow to the various loads, are designed to either fail open or fail as-is. In either case, depending upon the plant configuration, a high flow demand may result with the potential to stress the CCW pump or other loads supplied by CCW.

3.5 Summary

To summarize, the CCW system is exposed to a variety of stresses that can contribute to degradation with age. The potential effects which can result are shown in Table 3-1. This information provides insight into the failure mechanisms and modes which can be expected in the CCW system. The aging effects identified from this review were used as a baseline for comparison of the results from the data analysis discussed in Section 4.

4. ANALYSIS OF FAILURE DATA

As part of the aging assessment of CCW systems, failure data and operating experience from various national data bases were analyzed. This section briefly discusses the data sources used and presents the results of the analyses.

4.1 Data Bases

4.1.1 Descriptions and Limitations

The data bases used include the Nuclear Plant Reliability Data System (NPRDS), the In-Plant Reliability Data System (IPRDS), and Licensee Event Reports (LERs). All of the information obtained from these sources was reviewed and analyzed to obtain insights into the effects of aging on CCW system performance.

The national data bases have several virtues that make them suitable as sources of failure information. They contain a large amount of data representing a broad cross-section of nuclear power plants. The data is accessible, 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. The later approach was taken for this study. The data bases and their limitations are discussed in more detail in Appendix E.

4.1.2 Methods of Analysis

The information obtained from the NPRDS data base was the most extensive, consequently, the majority of the effort spent on data analysis focused on this data. A total of 1179 failure records related to CCW systems were obtained from the NPRDS. 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 was established of "aging related" based on the NPAR definition presented in NUREG-1144 [9] which 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 stressors (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).

To illustrate the data review process, Figures 4-1 and 4-2 present sample NPRDS failure records. In Figure 4-1, failure of a CCW isolation valve to pass a leak rate test is reported. Since the failure was due to wear on the valve disc and seat, and the valve was over six months old, the two aging criteria were met and this filure was classified as "aging-related." The failure mode is "leakage," the failure mechanism is "wear" and the failure cause is "normal service." Since the failure degraded the capability of the CCW system to provide isolation from containment, which is one of its functions, the effect of the falure was classified as "degraded operation."

NPRDS COMPONENT FAILURE CANNED REPORT
Plant Type-

COMPONENT FAILURE REPORT

ENTRY DATE: 06/26/86

1. Utility/Plant/Unit............
2. NPRDS Component Code...........VALVE
3. Utility Component ID...........
4. Discovery Date.................860227
5. Discovery Number...............1

6. Report Date....................860604
7. LER Report Number..............
8. System Affected by Failure.....WBB-COMPONENT COOLING WATER
9. Date Failure Occurred..........850827
10. Time Failure Occurred..........08:00
11. Date Failure Ended.............860430
12. Time Failure Ended.............22:00

13. Status Code....................F-SUBSYS/CHNL IN TEST
14. Severity Level Code............K-DEGRADED
15. Failure Symptom Code...........F-CONTAINED LEAKAGE
16. Failure Detection Code.........C-SURVEILLANCE TESTING
17. Cause Category Code............H-WEAROUT
18. Cause Description Codes........AD-NORMAL/ABNORMAL WEAR
19. System Effect Code.............D-LOSS OF SUBSYSTEM/CHANNEL
20. Plant Effect Codes.............G-RESULTED IN NO SIGNIFICANT EFFECT
21. Corrective Action Code.........AG-REPAIR COMPONENT/PART
22. Documentation Codes............Z-NONE OF THE ABOVE

23. Failure Description Narrative...
IS A REACTOR BUILDING PENETRATION BLOCK VALVE IN THE COMPONENT COOLING SYSTEM . IT FAILED A LEAK RATE TEST , INDICATING SEAT LEAKAGE .

24. Cause of Failure Narrative.....
THE CAUSE OF FAILURE WAS NORMAL WEAR ON THE DISC AND SEATS . THIS CAUSED AN UNEVEN SEATING SURFACE AND ALLOWED THE VALVE TO LEAK PAST THE SEAT .

25. Corrective Action Narrative.....
THE DISC AND SEATS WERE CLEANED AND LAPPED . A BLUE CHECK INDICATED 100% CONTACT AFTER LAPPING . .

COMPONENT ENGINEERING DATA

ENTRY DATE: UNKNOWN

Application Code........
Location................COMPONENT COOLING WATER
(NPRDS System Code= WBB)

Utility System Code.....

Data Start Date.........730716
In-Service Date.........730716
Out-of-Service Date.....

Safety Class............4
Critical Operation Mode.STANDBY CONDITION
Drawing / Doc Number....
Manufacturing Std.......

Internal Environment....AMB TEMPERATURE OVER 180F
SECONDARY COOLANT / TREATED WATER
External Environment....AMB TEMPERATURE -10F TO +120F
INSULATED

Manufacturer............POWELL , WILLIAM CO
Manufacturer Model No...1503WE
Manufacturer Serial No..

Supplier................
Supplier Id No..........

Engineering Codes

A. Type..................GATE
B. Operator..............PNEUMATIC/DIAPHRAGM/CYLINDER
C. Function/Application...SHUTOFF ISOLATION STOP
D. Body Material.........CARBON STEEL
E. Body Material Type.....FABRICATED
F. Nominal Inlet Size.....4 TO 11.99 IN.
G. Inlet Size.............8.0 IN
H. Maximum Pressure.......150 PSIG
J. Maximum Temperature....225 DEGF

% Time Operating When Reactor is Critical...100 %
% Time Operating When Reactor is Shutdown...100 %

Testing Performed	Frequency/Period	Hrs Out of Service
Check Testing	1 / ANNUAL	0
Functional Testing	1 / ANNUAL	0
Calibration Testing	0 / NOT DONE	0

Figure 4-1 Sample NPRDS Record for Aging Related Failure

NPRDS COMPONENT FAILURE CANNED REPORT
Plant Type-

COMPONENT FAILURE REPORT

ENTRY DATE: UNKNOWN

1. Utility/Plant/Unit.............
2. NPRDS Component Code...........PUMP
3. Utility Component ID...........
4. Discovery Date.................781113
5. Discovery Number...............1

6. Report Date....................800320
7. LER Report Number..............
8. System Affected by Failure.....WBB-COMPONENT COOLING WATER
9. Date Failure Occurred..........781113
10. Time Failure Occurred..........12:00
11. Date Failure Ended.............781113
12. Time Failure Ended.............15:00

13. Status Code....................E-SUBSYS/CHNL IN SVC;OP/STANDBY
14. Severity Level Code............K-DEGRADED
15. Failure Symptom Code...........A-PHYSICAL FAULT
16. Failure Detection Code.........E-SPECIAL INSPECTION
17. Cause Category Code............J-OTHER DEVICES
18. Cause Description Codes........BB-MECHANICAL DAMAGE/BINDING
19. System Effect Code.............C-LOSS OF REDUNDANCY
20. Plant Effect Codes.............G-RESULTED IN NO SIGNIFICANT EFFECT
21. Corrective Action Code.........AG-REPAIR COMPONENT/PART
22. Documentation Codes............Z-NONE OF THE ABOVE

23. Failure Description Narrative...
COMPONENT COOLING WATER PUMP INOPERABLE .

24. Cause of Failure Narrative......
THE PUMP CASING THREADS WERE FOUND TO BE STRIPPED .

25. Corrective Action Narrative.....
THE THREADS WERE RETAPPED AND AN INSERT INSTALLED .

COMPONENT ENGINEERING DATA

ENTRY DATE: UNKNOWN

Application Code.........
Location................COMPONENT COOLING WATER
(NPRDS System Code= WBB)

Utility System Code.....

Data Start Date.........771121
In-Service Date.........771121
Out-of-Service Date.....

Safety Class............3
Critical Operation Mode.OPERATING
Drawing / Doc Number....
Manufacturing Std.......

Internal Environment....AMB TEMPERATURE -10F TO +120F
SECONDARY COOLANT / TREATED WATER
External Environment....AMB TEMPERATURE -10F TO +120F
AIR

Manufacturer............GOULDS PUMPS INC
Manufacturer Model No...3415
Manufacturer Serial No..

Supplier................
Supplier Id No..........

Engineering Codes

A. Type....................CENTRIFUGAL
B. Inlet Size..............12-17.99 IN.
C. Body Material...........CARBON STEEL
D. Type of Shaft Seal......PACKING GLAND
E. Flow Capacity...........2500-9999 GPM
G. Total Developed Head....150 FTHO
H. Flow Rating.............7860 GPM
J. Rotational Speed (Cap).1180 RPM

% Time Operating When Reactor is Critical...33 %
% Time Operating When Reactor is Shutdown...33 %

Testing Performed	Frequency/Period	Hrs Out of Service
Check Testing	1 / MONTH	0
Functional Testing	2 / THREE YEARS	1
Calibration Testing	0 / NOT DONE	0

Figure 4-2 Sample NPRDS Record for Non-aging Related Failure

An NPRDS component failure report for a CCW pump is shown in Figure 4-2. Since the failure of a CCW pump is reported. Since the failure was due to the pump casing threads being stripped, and not by any aging mechanism, this failure did not meet the aging criteria and was classified as "non-aging related." The failure cause was classified as "human error," and the failure mechanism was "other," since no aging mechanism was present. Since the pump was taken out of service, the effect of the failure was categorized as a "loss of redundancy."

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 CCW system boundaries defined in Section 1. Once the data base was complete, the computer sorted the data in various ways to obtain the information for this analysis. The database findings were then checked against actual plant data (discussed in Section 5) to verify the results.

The IPRDS data base contains information on pumps and valves from only a few nuclear power plants. These data were not computerized and had to be hand sorted. Also, the data were not in current use and were not actively maintained. Consequently, data on CCW system valve failure from IPRDS were available for only three plants with a reported population of 326 valves; for pumps, failure data were obtained for two plants, with a reported population of 12 pumps. The valve data included 88 failures reported from February 1975 to June 1981; for pumps, there were 92 failures from May 1974 to June 1981.

It is believed that the data obtained from IPRDS are incomplete since some records did not reflect realistic component populations. However, the records were individually reviewed and analyzed to determine if they showed any significant differences to the results obtained from the NPRDS data. Each of the failure records was reviewed to determine if the failure was aging related. The same definition of aging applied to the NPRDS data was used as the criterion, but due to the limited amount of information available, many were categorized as "unknown."

There were 478 reports in the LER data base that were related to the CCW system. As for the IPRDS data, each LER record was individually reviewed and categorized as to whether or not it was aging related. The various failure modes, failure causes, and component types were also identified and analyzed. Many of the LERs did not contain sufficient information to classify the failure characteristics, hence, a number of them were categorized as "unknown."

4.2 Dominant Failure Trends

4.2.1 Aging Fraction

In performing any aging assessment, the primary concern is to determine if there is aging degradation and if it is adversely affecting the performance or reliability of the system. One method of doing this is to examine past operating experience and determine the fraction of failures which are aging-related. If the fraction is large, aging-related degradation is significant, and appropriate measures should be taken to mitigate it.

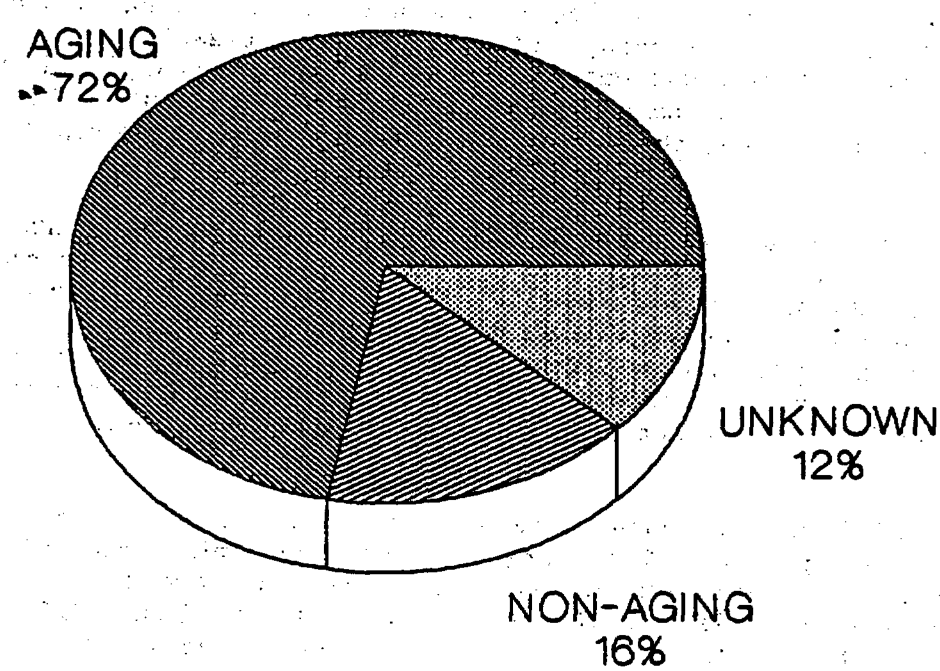

Figure 4-3 Aging fraction - NPRDS data

An analysis of the failure data (Figure 4-3) shows that the CCW system is susceptible to aging-related degradation, and that 72% of the failures reported were aging related. Results obtained from IPRDS data and LER data, which are shown in Figures 4-4 and 4-5, respectively, also show large aging fractions. One reason for the large number of age-related failures is that the CCW system is normally operating at all times, and the components accumulate a large number of operating hours. These results demonstrate that aging is a concern for CCW systems. The ability to monitor and control the aging phenomena, therefore, is important and should lead to increased system reliability.

As shown in Figures 4-4 and 4-5, the aging fraction from the IPRDS and LER data are smaller than the aging fraction from the NPRDS data. For the IPRDS result, this is believed to be due to the limited amount of information which resulted in a large number of failures being classified as unknown.

The LER aging fraction is believed to be lower than NPRDS due to the large number of reports in the LERs dealing with human errors. These include failures to perform various tests or failures to have certain procedures available. Events of this type are not reportable to NPRDS. These events are clearly not aging-related and therefore the LER data give a lower aging fraction.

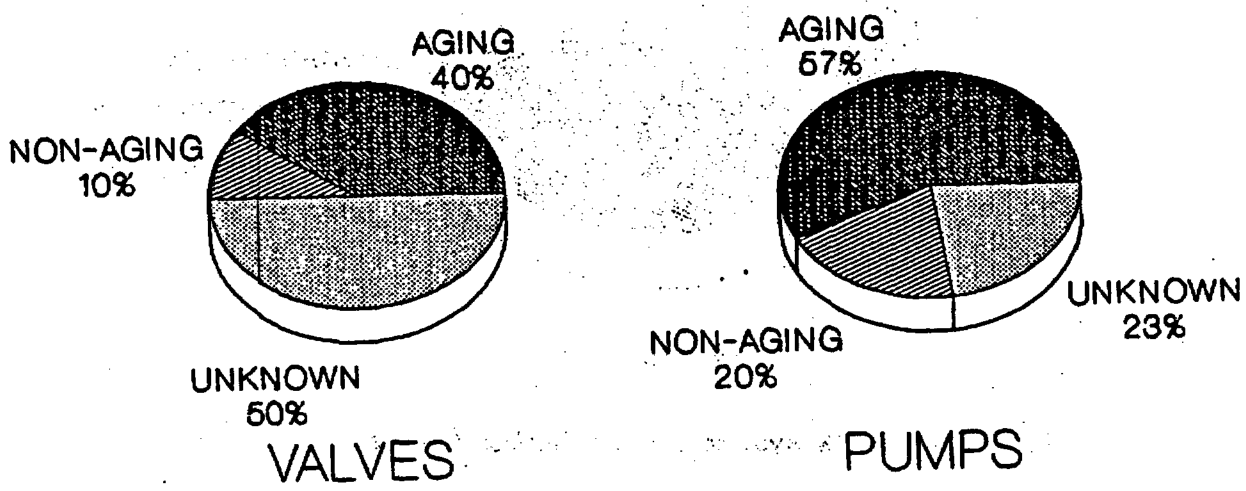

Figure 4-4 Aging fraction - IPRDS data

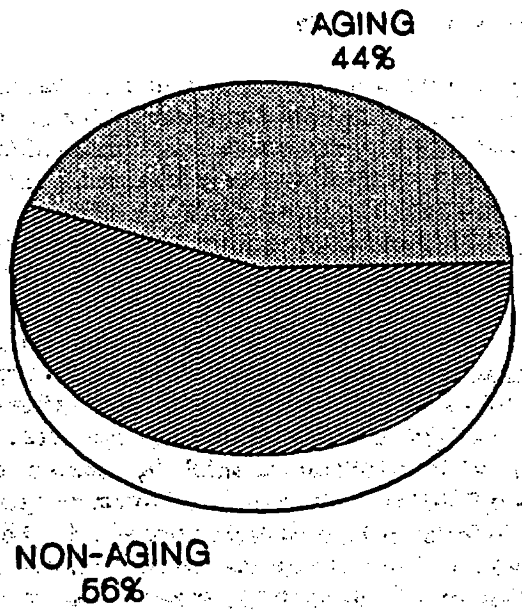

Figure 4-5 Aging fraction - LER data

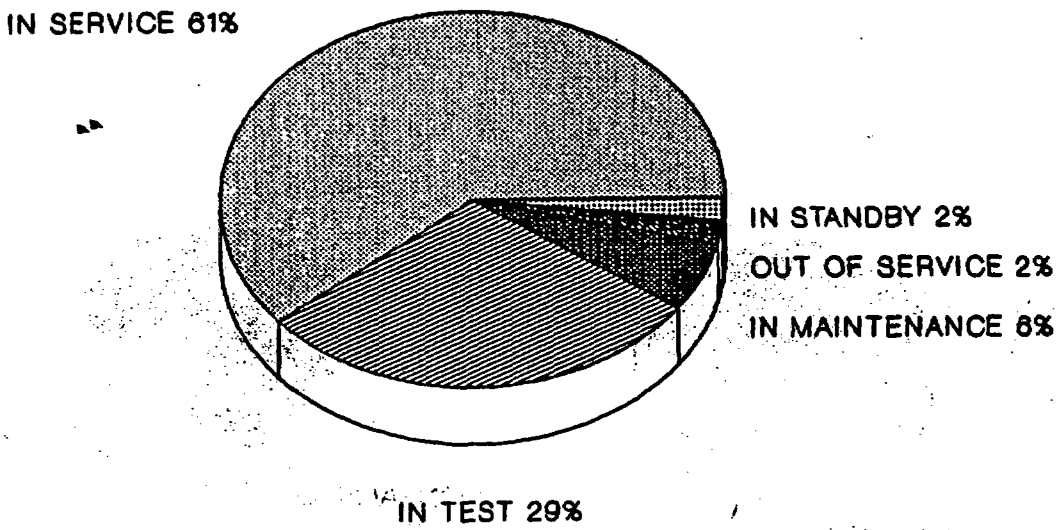

Figure 4-6 CCW system status during failure detection

4.2.2 Failure Detection

The CCW system is a normally operating system, consequently, most failures should occur when the system is operating. This is verified by the data, as shown in Figure 4.6. Since failures most frequently occur when the system is operating, the need for good functional indicators to detect and monitor aging effects becomes more important. The functional indicators should be capable of detecting failures in the incipient stage to ensure no loss of function of the CCW system. These will be developed and addressed in the second phase of the CCW system study.

The methods by which CCW system failures are detected were determined and are presented in Figure 4-7. As shown, operational abnormalities identified 34% of all CCW system failures. Operational abnormalities include events such as a pump failing to start on demand or a valve failing to open. Inspections while the system was operating detected 29% of the failures, while testing detected 30%. This is consistent with the results shown in Figure 4-6 for the status of the CCW system during failure.

Considering testing, alarms, inspections and maintenance to be parts of monitoring and surveillance methods, then 65% of all failures are detected by the methods currently in use. Of the remaining failures, 34% are not detected until system operation is affected. However, the effect on the performance of the system at the time of detection should be examined to assess the overall effectiveness of the monitoring methods. If system performance is degraded to an unacceptable level before failures are detected, improvements in monitoring may be required and should certainly be considered. This conclusion is supported by the small number of plants reviewed which show that monitoring of the CCW system is routine but not extensive. There are areas for improvement without the expenditure of large resources: this subject will be developed in the second phase of the CCW system study.

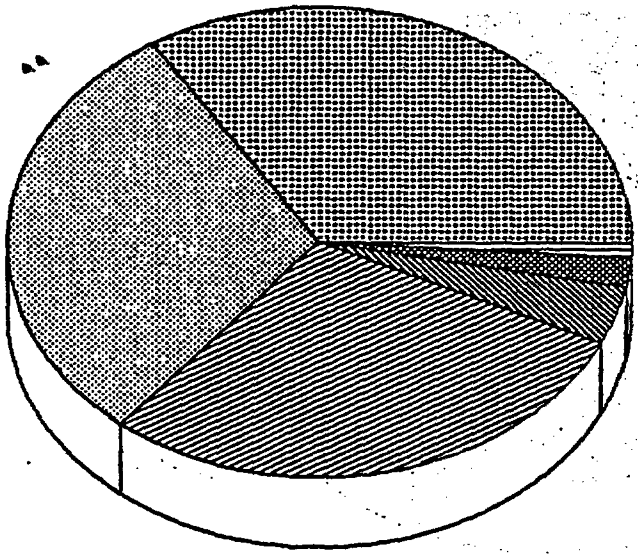

Figure 4-7 CCW system failure detection methods

4.2.3 Effects of Failure

An additional aspect of failure which must be considered is its effect on the performance of the CCW system. The NPRDS results presented in Figure 4-8, show that over half of all failures degrade the operation of the system. Degraded operation implies that the system can still perform its function, but it cannot continue indefinitely without some corrective action being taken. If left uncorrected, the failure would get progressively worse until there was a complete loss of function or an impairment of safety.

The results presented in Figure 4-8 also show that 27% of the failures resulted in a loss of redundancy in the CCW system. This would occur, for example, if a heat exchanger which is normally used for standby service failed and had to be taken out of service for repair. The system could still operate, but if another heat exchanger failed, there would be no backup to replace it, and the system's function could be lost. From a PRA standpoint, a loss of redundancy in the system would result in an increase in unavailability and a decrease in reliability. Therefore, it is important to reduce the number of failures that could cause a loss of redundancy.

Only one failure out of 1179 reported to NPRDS resulted in a complete loss of CCW system function. This failure was due to the inadvertent spraying of water during maintenance into the CCW pump area, which shorted the motors of all three CCW pumps. The event was caused by human error and was not aging related.

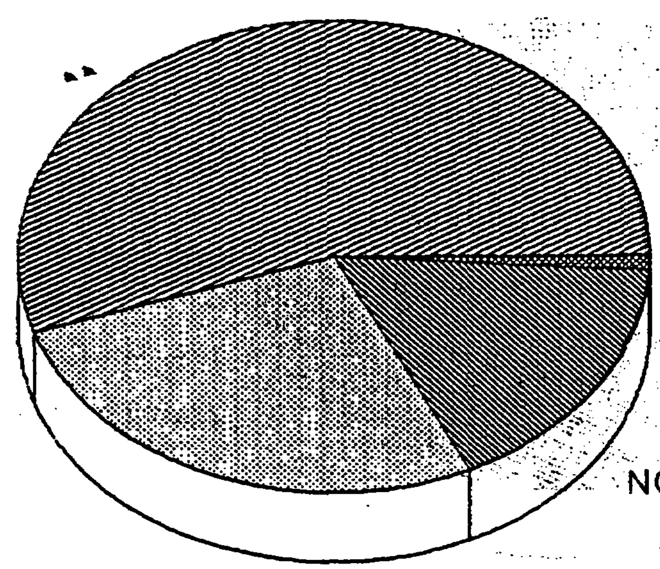

Figure 4-8 Failure effect on CCW system performance

Total loss of CCW system function rarely occurs due to the redundancy designed into the systems. If a complete loss of function should occur at the component level, a backup component is available to replace the failed one and prevent the system's function from being interrupted. This demonstrates that current designs have sufficient redundancy to provide a satisfactory level of reliability; however, it may be possible to make improvements to increase the system's availability.

4.2.4 Causes of Failure

To identify the reason for the various CCW system failures, the data were categorized according to the cause. For this analysis, the cause was defined to be the general condition or event which resulted in component failure. The true "root cause" of failure could not be determined in most cases, due to the lack of information.

In Figure 4-9, the causes of CCW system failures are shown as a function of plant age. Normal service is the predominant cause of failure which is consistent with the large aging fractions seen previously. Normal service includes any function or process which the component is expected to encounter during its life. As plants age, the percentage of failures caused by normal service increases. This indicates that the degradation due to aging increases with time, as would be expected.

The human error failures shown in Figure 4-9 include errors in the design and manufacturing of the systems and components as well as errors in their operation and maintenance. The failure causes in the category classified as "other" include failures of other components and systems outside CCW boundaries, operation in a harsh environment, and operation during accidents requiring service outside normal limits.

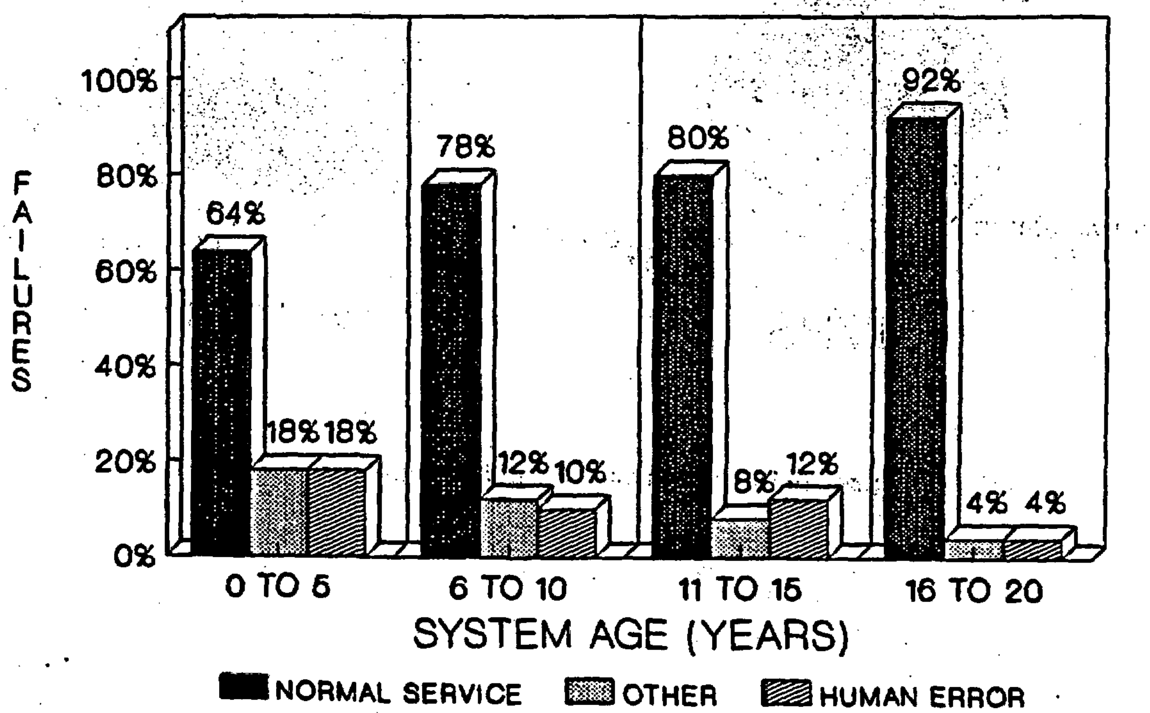

Figure 4-9 CCW system failure causes vs. plant age - NPRDS

Similar results were obtained from analysis of the IPRDS and the LER data (Figures 4-10 and 4-11): both sources showed a high percentage of failures caused by normal service. As discussed previously, the IPRDS data was incomplete and, therefore, a large number of failure causes were classified as unknown. The LER data base included a large number of failures caused by human error as a result of which there was a smaller percentage of failures caused by normal service.

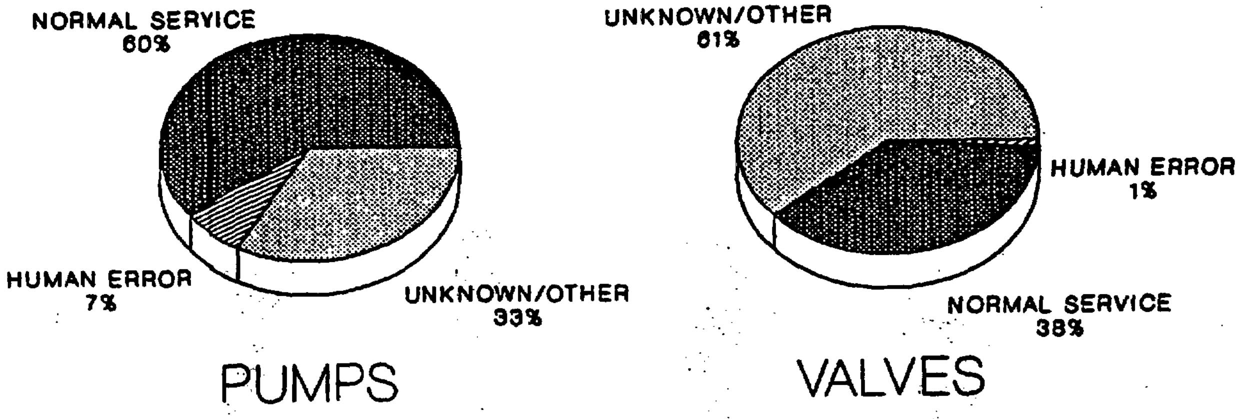

Figure 4-10 CCW system failure causes - IPRDS

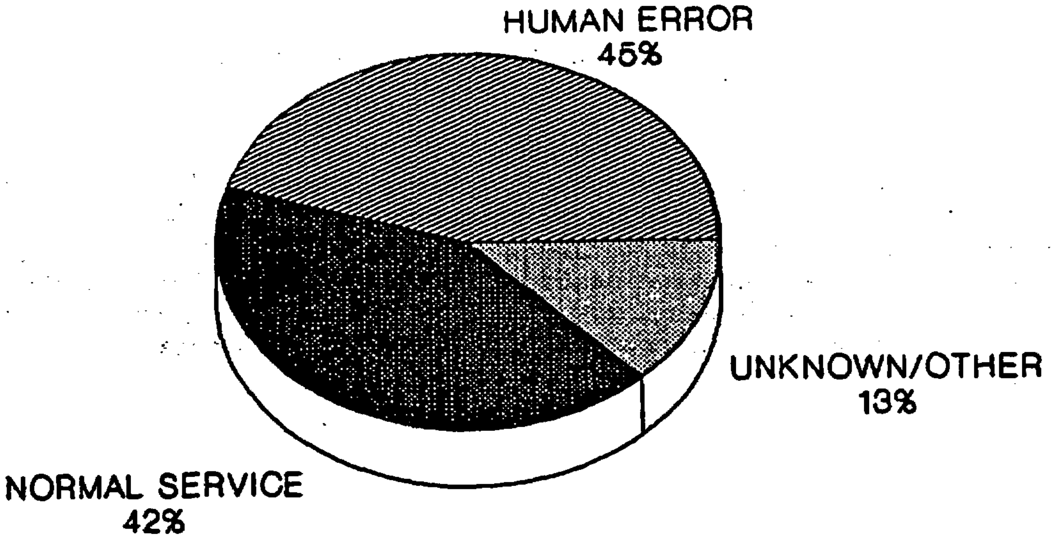

Figure 4-11 CCW system failure causes - LERs

The second largest cause of failures was human error. To identify areas where improvements could be made, the data were analyzed to determine the types of human errors. Excluding manufacturing and design errors, the results, shown in Figure 4-12, indicate that problems related to maintenance account for more than 60% of all human error failures. Any efforts made to mitigate human errors could most productively be focused in the area of maintenance of the CCW system.

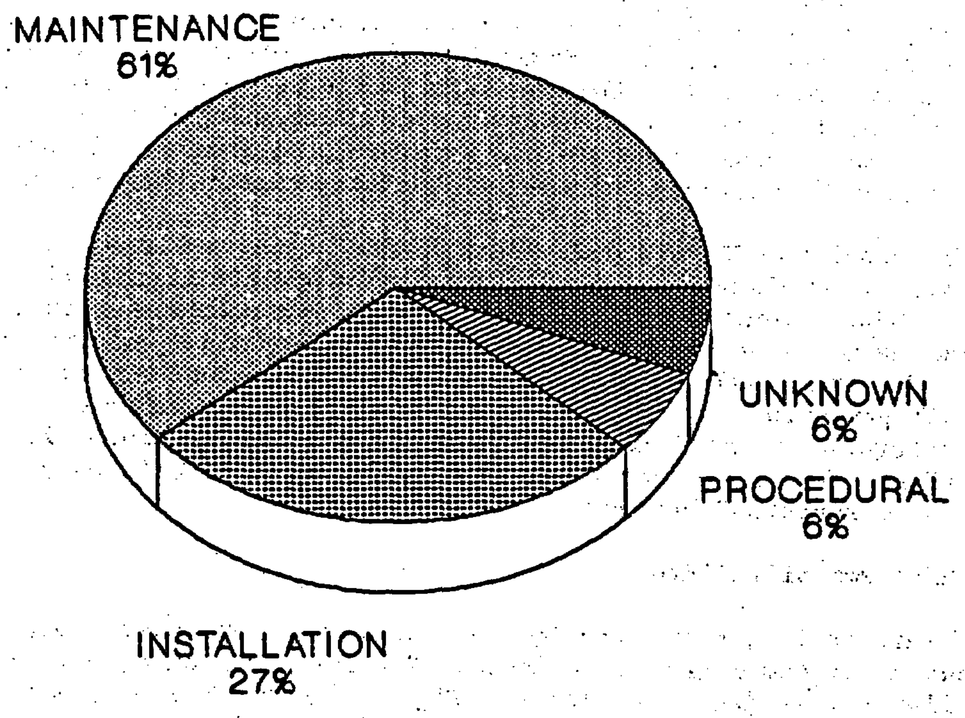

Figure 4-12 Types of human errors

It is noted from Figure 4-9 that failures related to human error tend to decrease with age. In early years, they account for 16% of the failures; in later years, they account for only 4%. This decrease probably can be attributed to the personnel becoming more familiar with operating and maintenance procedures and, therefore, performing them more efficiently (i.e., learning-curve effect). This finding demonstrates the importance and benefit of having experienced personnel available.

4.2.5 Modes of Failure

The effect by which a failure is detected is usually referred to as the failure mode. For a pump, this could be leakage or excessive vibration, among others. Valve failure modes include failure to open/close, or leakage. Failure mode identification is useful in assessing surveillance and monitoring methods since it focuses attention on the proper areas to inspect or test.

The predominant failure modes for the CCW system are quite diverse. As shown in Figure 4-13, leakage (37%) was the most common mode of failure in the NPRDS data. The IPRDS data and LER data also show leakage to be a common failure mode (Figures 4-14 and 4-15). This is a typical failure mode associated with both pump and valve failures and suggests that inspecting and testing for leakage is an important monitoring method.

It should be noted from Figure 4-13 that a number of different failure modes occur in the CCW system. The failure modes classified as "other" include disengaged, engaged, opened, closed, opened circuit, overloaded, ruptured, and tank level changed. Several different monitoring techniques would be required to detect all failures. For example, visual inspections could only be expected to detect a portion of the CCW system failures. A good surveillance and monitoring plan should, therefore, be diverse and include sufficient tests and inspections to cover all the significant failure modes.

It is noted from Figure 4-15 that in the LER data were a large number of events which were classified as failure to meet specifications or loss of function. Those events which failed to meet specifications included items such as not performing a test required by the technical specifications, for example. The large number of these events reportable as LERs account for the high overall percentage of failure to meet specification events. The loss of function category is large due to the lack of detailed information available from the LERs. This resulted in a number of failure modes being placed in the broad "loss of function" category.

4.2.6 Mechanisms of Failure

A failure mechanism is the physical, chemical, or other process by which a component or system degrades or fails. For the CCW system, a number of different failure mechanisms are present (Figure 4-16). The predominant failure mechanism was wear, which accounted for 37% of the failures reported to NPRDS. In this analysis, wear represented an exposure to stresses encountered during operation, as described in Section 3, which resulted in some portion of the component being worn away. This is typically associated with an aging phenomenon. The large number of failures due to wear is consistent with the large aging fractions seen previously.

The other 63% of failures were fairly evenly distributed among the other mechanisms, as shown in Figure 4-16. The failure mechanisms categorized as "fracture" include those where fracture or crack growth lead to failure. The "contamination" category includes failures where a foreign material was introduced into the system/component causing a buildup or blockage. The "calibration" category includes failures where calibration or set point drift of a device occurred, resulting in a violation of specifications. The failure mechanisms categorized as "other" include embrittlement, fatigue and abnormal stresses. Most of these mechanisms may be aging-related.

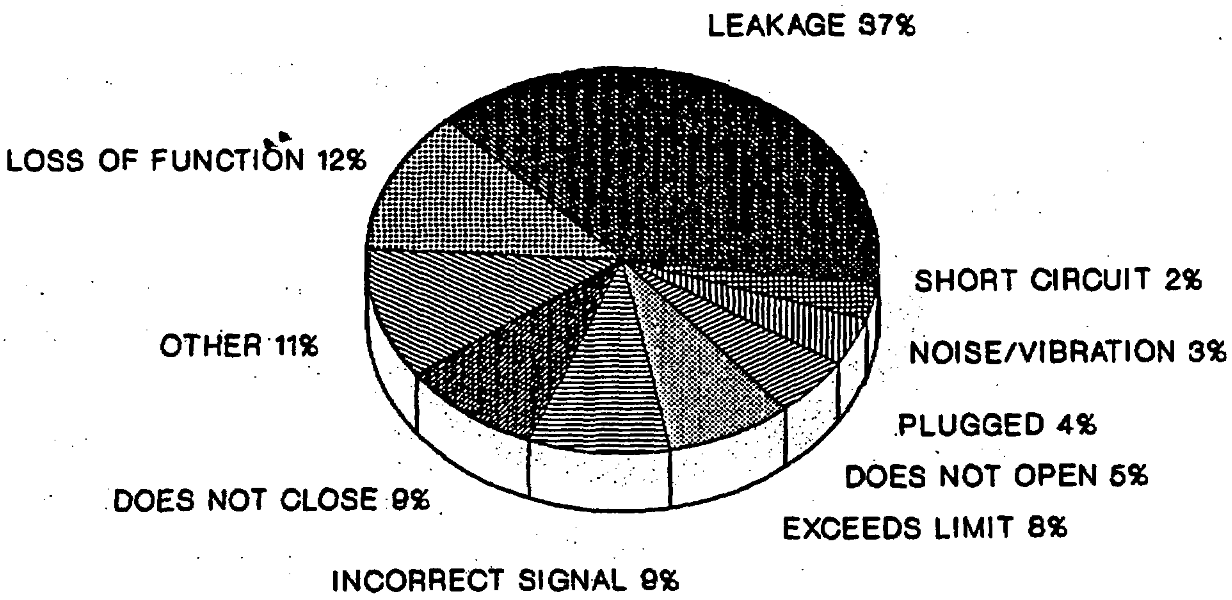

Figure 4-13 CCW system failure modes - NPRDS

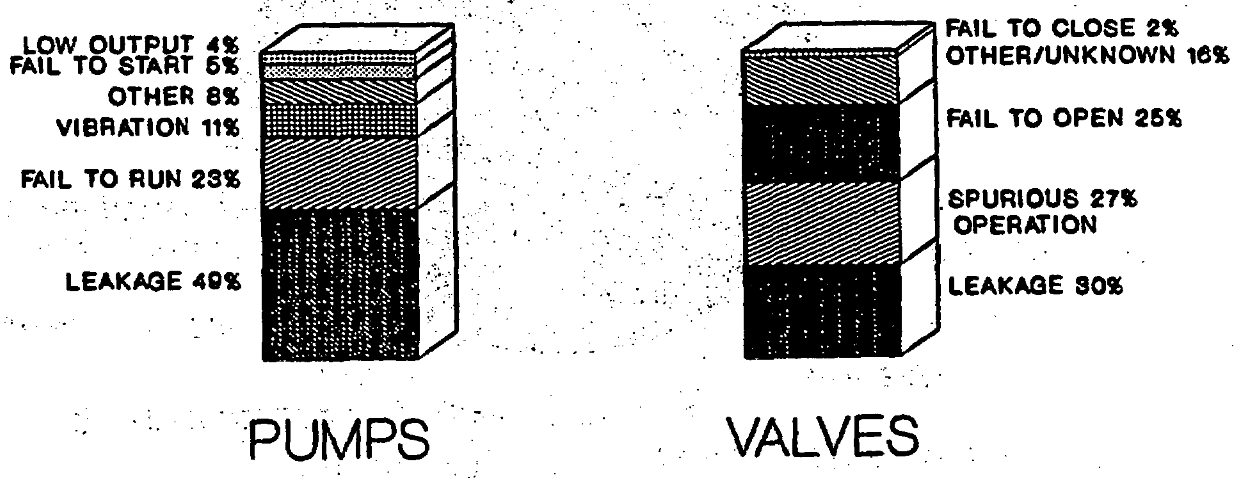

Figure 4-14 CCW system failure modes - IPRDS

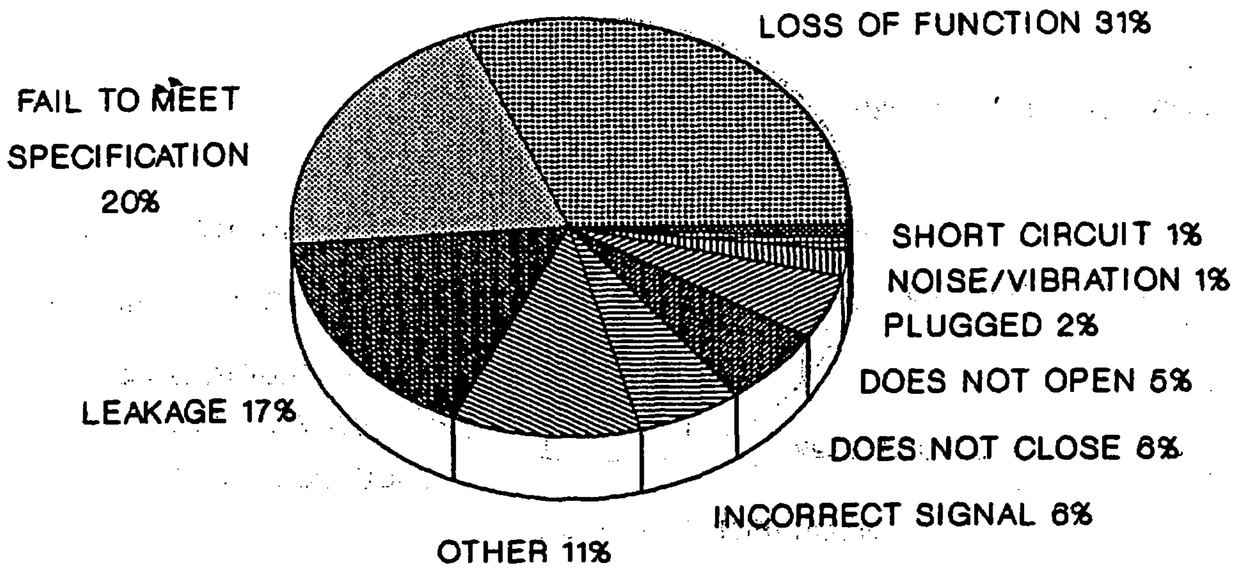

Figure 4-15 CCW system failure modes - LER data

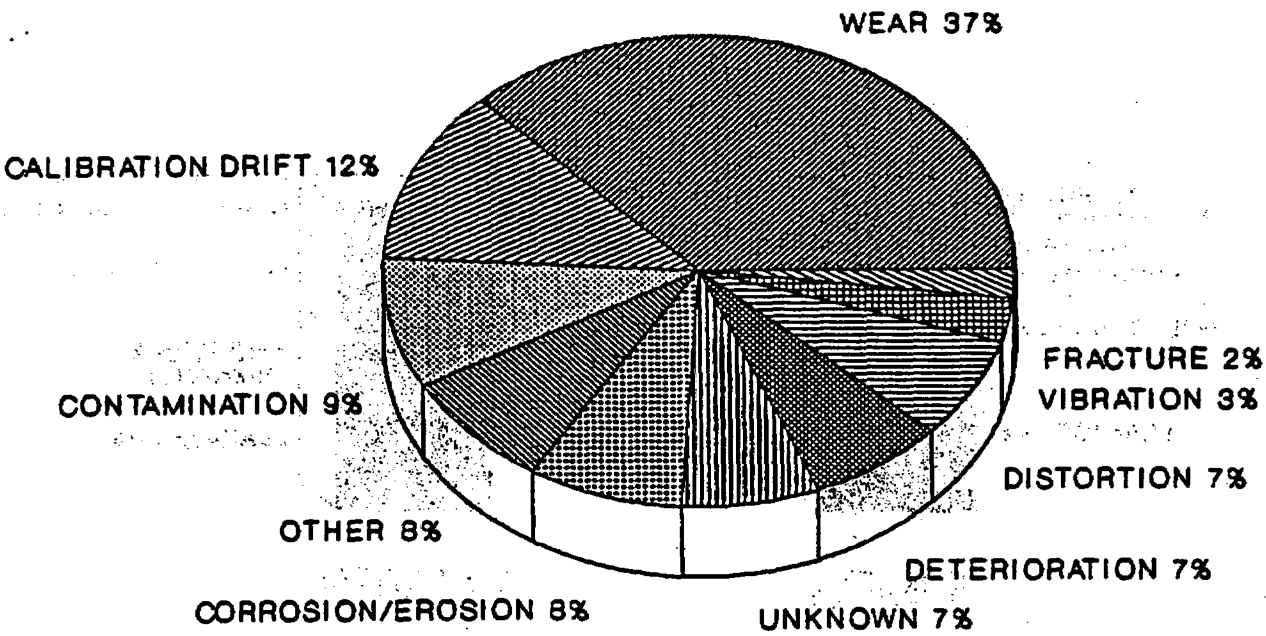

Figure 4-16 CCW system failure mechanisms

4.3 Analysis of Component Failure

4.3.1 Predominant Component Failures

The failure trends discussed in the previous section are characteristic of the CCW system at a system level. To provide more specific information on the aging phenomena taking place in the CCW system, the data were analyzed at a component level.

The number of failures attributed to each of the components in the CCW system is shown in Figure 4-17, together with the aging fraction for each component. The results indicate that valve-related failures are the dominant type. This is expected since the number of valves is much larger than the number of other components (e.g., for IP2 there are 283 valves, 3 pumps, and 2 HX's). The failure data were normalized and results are discussed later in this section. Pumps, instrumentation/controls, and heat exchangers also contribute to a significant portion of the CCW system failures. The aging fraction for each component ranged from 42% to 84%, showing that aging degradation occurs for all CCW system components and accounts for the majority of failures.

Similar results were obtained from an analysis of the LER data. As shown in Figure 4-18, valves were the component most frequently reported as failing. Pumps, heat exchangers, and radiation monitors also made significant contributions. It should be noted that only those LERs related to component failures were included in the data in Figure 4-18. LERs dealing with violations of specifications were not included.

Since valves constitute the highest percentage of CCW system failures, the data were sorted further to identify the types of valves which failed most frequently. The results, in Figure 4-19, indicate that air-operated valves (AOVs) and motor-operated valves (MOVs) experience the most failures, followed by relief valves, manual valves, and check valves. This is understandable since AOVs and MOVs include actuators which make them more complex components than the other valve types. As can be seen from Figure 4-19, aging is a significant factor in failure for each type of valve.

Figure 4-19 shows only the relative frequency of failure for the various types of valves. It does not indicate the importance of each, nor the significance of a failure for any specific type of valve. Manual valves, for example, do not fail as frequently as air-operated valves, but this does not imply that the monitoring of AOVs in general should be stressed in monitoring programs, more than that for manual valves. In some system designs, failure of certain manual valves could have a much more significant impact on system performance than the failure of an AOV. This could be the case in a system which uses manual valves in critical portions of the main flow path where valve failure could result in a loss of flow, as is shown in the Indian Point-2 analysis in Section 6. The PRA analyses discussed in Sections 6 and 7 show that main header and main flowpath valves (motor-operated, air-operated or

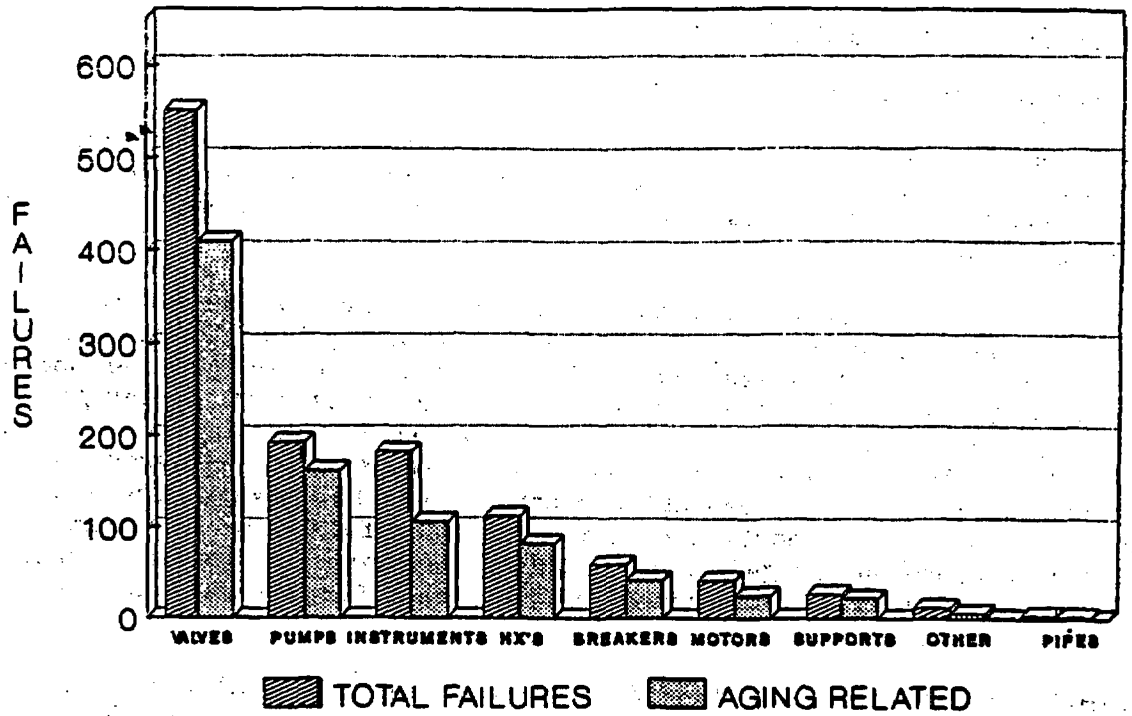

Figure 4-17 Component failures and aging fractions

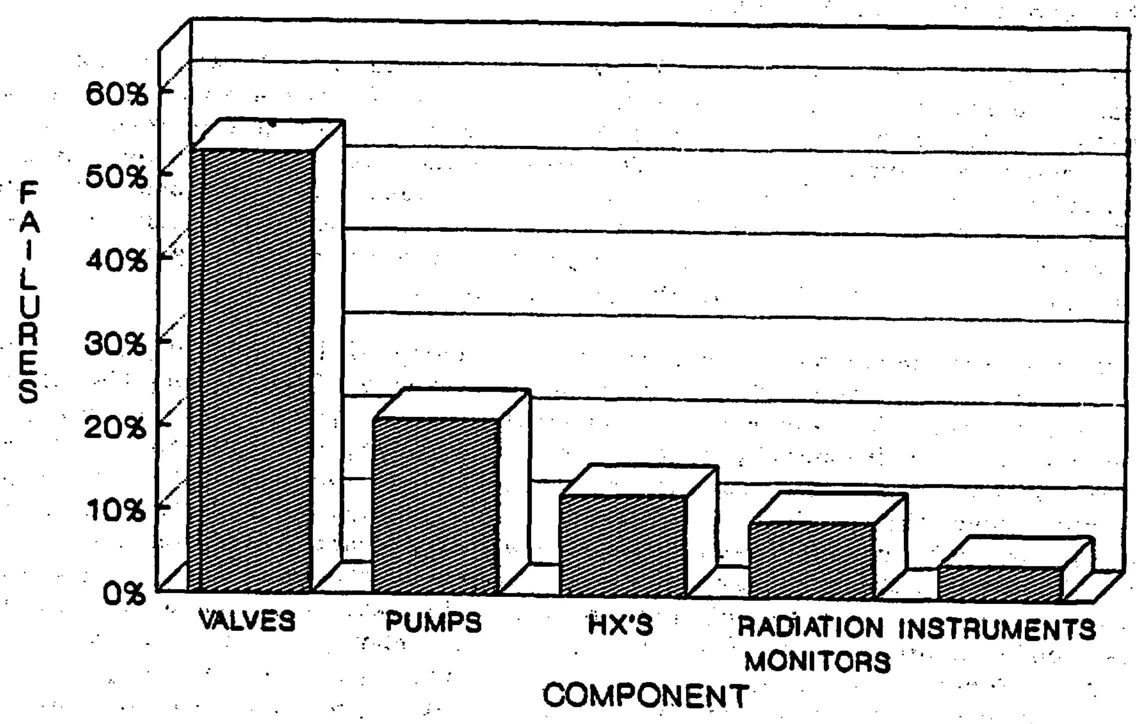

Figure 4-18 Component failures - LERs

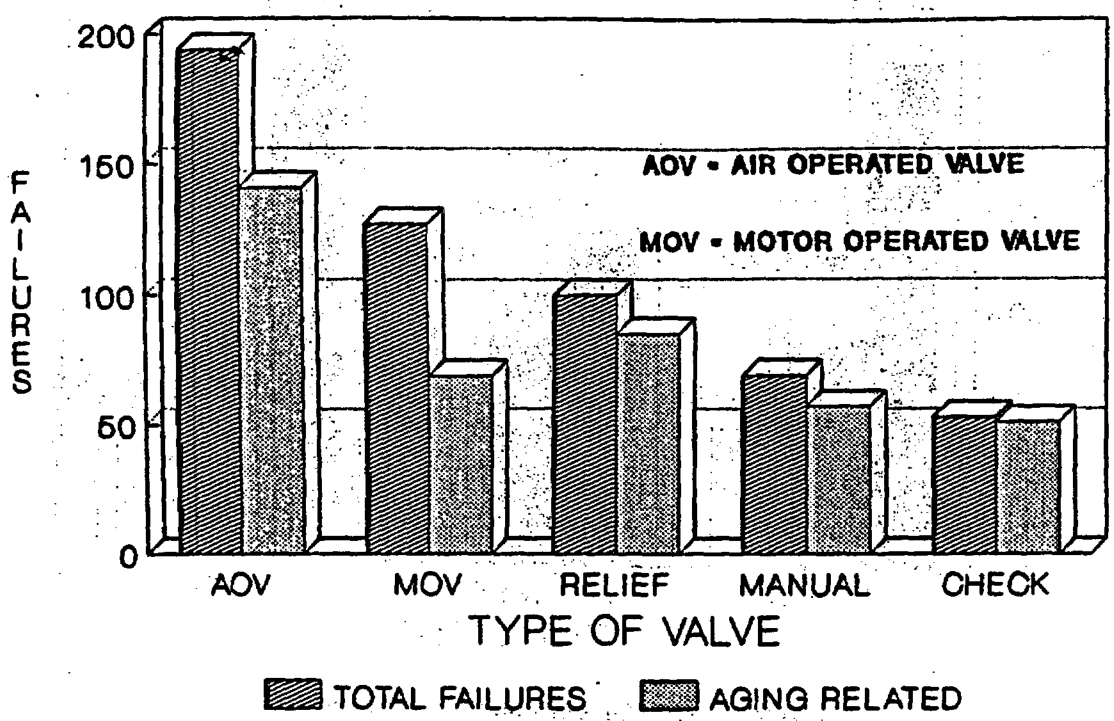

Figure 4-19 Valve failures and aging fraction

manual) are very important and should receive more attention than they currently do. Therefore, the information presented here must be used in conjunction with specific design information to determine the importance of components in the surveillance and monitoring plan for particular plants.

For each component showing a significant number of failures, the data were examined to identify the specific subcomponents which failed. Figure 4-20 shows that pump failures were dominated by failures of the seals and bearings. This is important, since it suggests that any improvements in monitoring or maintenance of pumps should be focused on these parts. Similarly, valve operators and valve seats were shown to be the predominant areas causing valves to fail. Tubes were the predominant subcomponent failing in heat exchangers.

4.3.2 Component Test Frequencies

An additional factor which must be considered in evaluating component failures is the frequency with which the components are tested. Some must be tested at specific intervals by technical specifications, while others are tested at the discretion of plant management. Component test frequencies are commonly based on past experience and manufacturer's recommendations.

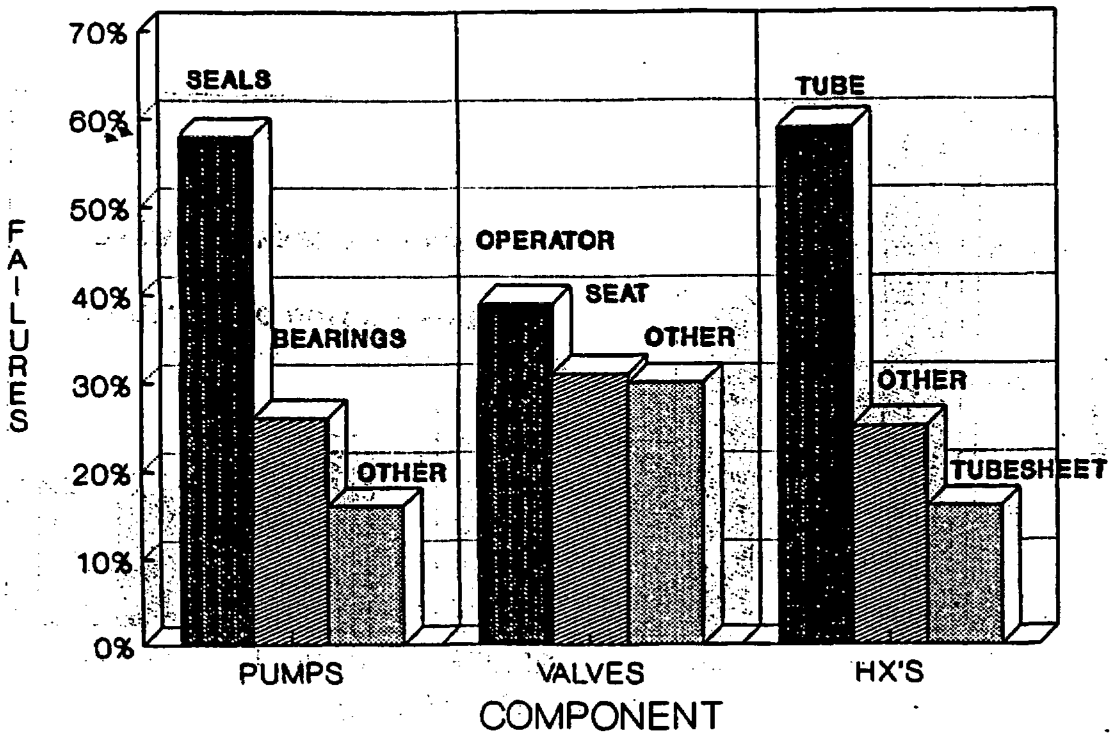

Figure 4-20 Subcomponent failures

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 CCW 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 4-21 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 4-22, 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. With the exception of pumps, all of the components examined showed the most failures for components which are not check-tested. However, the number of failures occurring for components which are frequently checked also is significant.

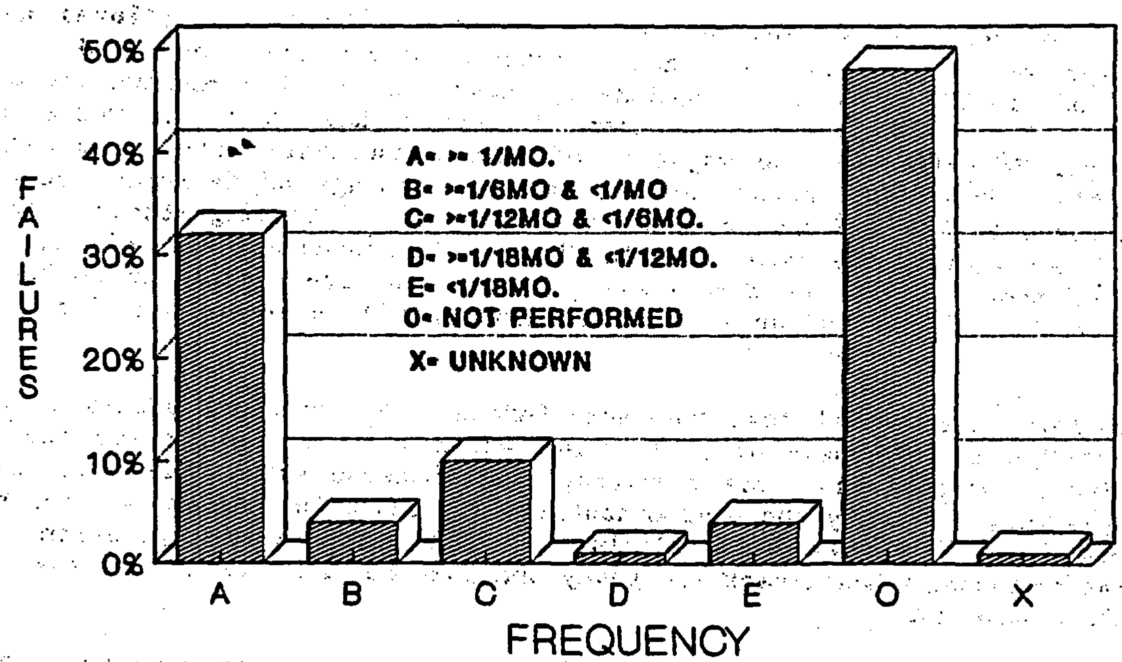

Figure 4-21 Check test frequency for all CCW system components

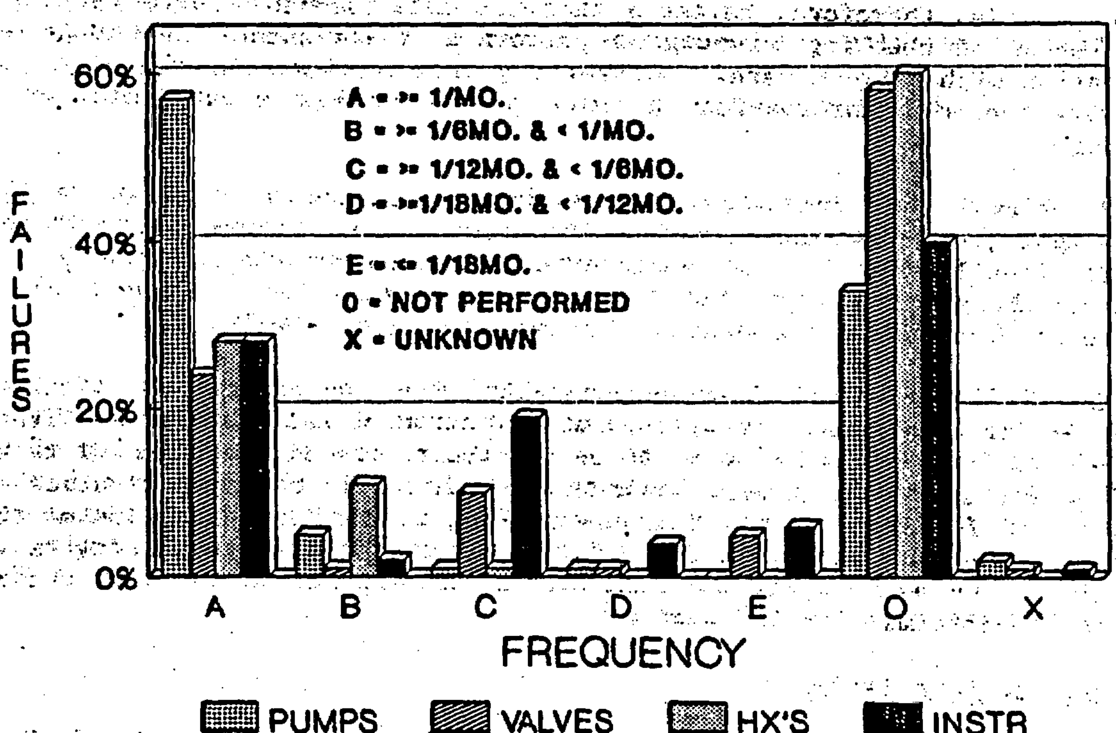

Figure 4-22 Check test frequency for specific components

As previously mentioned, data on pumps indicate that most failures occur for pumps which are frequently check-tested. Pumps are recognized to be costly and important pieces of equipment; consequently, they receive the most attention of all components, regardless of whether they are considered safety or non-safety related. The majority of pumps, therefore, are check-tested frequently. This may account for the large percentage of failures occuring for frequently tested pumps, and not the frequency of testing itself. It is noted, however, that the data shown in Figure 4-22 indicate that a significant number of pumps are not check-tested. Since pumps have been shown to have a significant number of failures, check-testing should be considered for these pumps.

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 CCW 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.

Functional test frequency for CCW system components is shown in Figure 4-23. Again, the predominant number of failures occur in components which are not 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 4-24.

The functional test frequency results show no correlation to failure rate. As for check-test frequency, the distribution of failures for the functional test frequencies is believed to be related to the distribution of testing frequencies at the plants. Since there is a fairly even distribution of failures for functional tests that are performed, further investigation may be warranted by normalizing this data with the number of plants performing tests at each of the various frequencies. This is recommended as part of future work to be performed for CCW systems.

4.4 Failure Rate Analysis

The previous sections presented a qualitative analysis of the failure data which provided insights into the effect of aging on CCW system performance. This section discusses the results of a quantitative analysis to identify aging effects on failure rates. The objective was to determine the

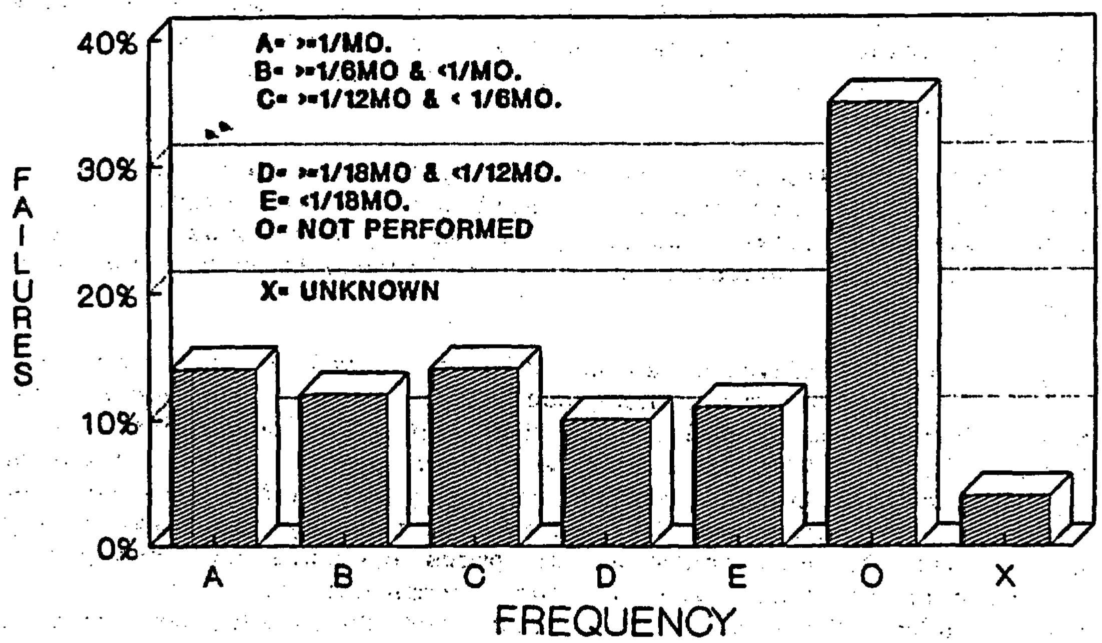

Figure 4-23 Functional test frequency for all CCW system components

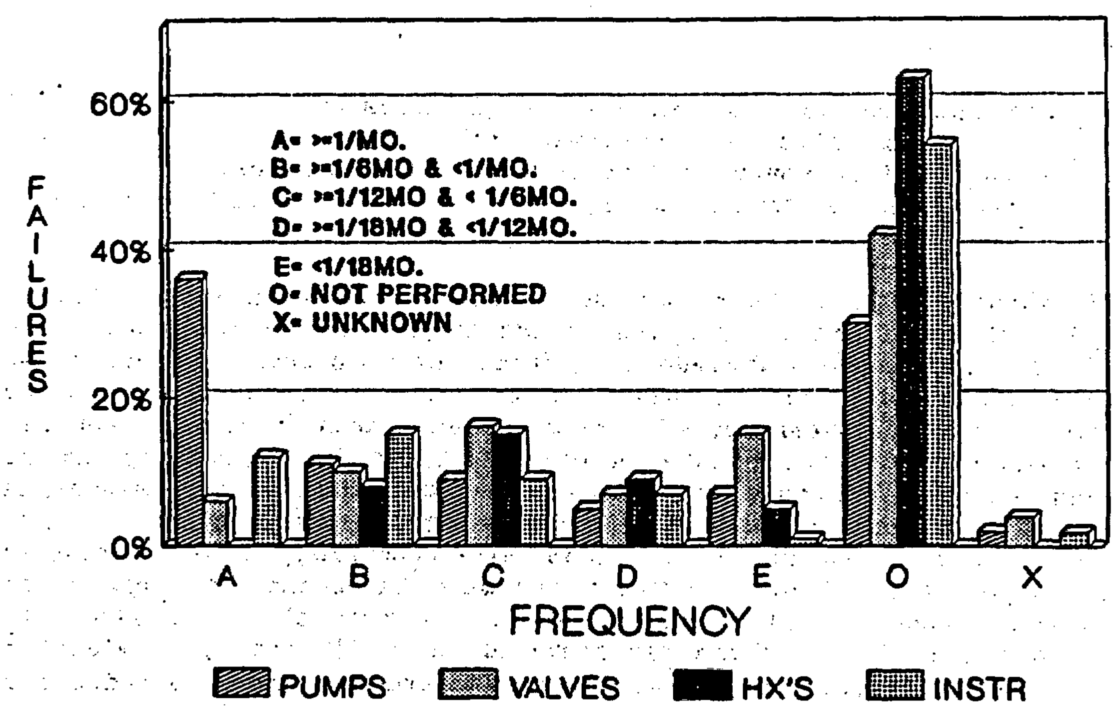

Figure 4-24 Functional test frequency for specific components

effect of aging on component failure rates and then use these rates to quantitatively evaluate the aging effect on current PRA analyses. Detailed analyses and calculations yielded time-dependent failure rates, which were not available previously.

To estimate time-dependent failure rates, the data were first sorted to determine the number of failures as a function of age for pumps, heat exchangers, motor operated valves (MOVs), check valves, and manual valves. The NPRDS data were the sole source of information because they were the most comprehensive.

To normalize the failure data, an estimate was made of the component population for each plant. The methods used in determining the populations are discussed in Appendix D. Together with the plant ages and the date each plant started reporting to NPRDS, this information was used to determine the number of operating hours falling in the NPRDS reporting period for each component as a function of age. Since the CCW system is a normally operating system, it was assumed that all components except pumps operated continuously. For pumps, the most common design includes three pumps with two normally operating and one in standby. Therefore, it was estimated that each pump operates 65% of the time.

Failure rates were then calculated from the equation

$$\lambda_i = \frac{n_i}{T_i} \qquad (4.4\text{-}1)$$

where:

λ_i = Failure rate in time interval i (failures/hr)
n_i = Number of failures in time interval i
T_i = Number of component operating hours in time interval i
i = One-year interval ending at age i years (i.e., for age 1.1 to 2.0 i=2)

It should be pointed out that the failure rates calculated in this manner are only estimates for the purpose of identifying trends with age. The data used in the calculations include the effects of maintenance, testing, inspection and other periods of down-time. Information is not available to separate out these effects, therefore, effective failure rates (sometimes termed failure frequencies) were calculated and used. There is uncertainty in the numbers caused by such factors as component down-time due to maintenance, testing or refueling outages, component replacement with new equipment (believed to be a small effect), inaccurate or incomplete reporting, and degraded versus failed components. Also, no attempt was made to characterize the quality of maintenance. Despite these shortcomings, however, the data base is large enough, and the effect of the deficiencies is small enough, to yield reasonable, usuable results. Sensitivity studies were performed to address uncertainties in the data. They identified areas where the results are particularly sensitive to the failure rates. In general, results were not especially sensitive to variations in the calculated numbers. These studies are discussed in detail in Section 7.

It should be noted that if the effects of test and maintenance in the data are considered, the resulting component failure rates may be higher than shown here. This could occur, for example, if a component is refurbished at some point in its life making it "good-as-new." The operating hours used in calculating a failure rate then would have to start when the component was refurbished, resulting in a smaller number of operating hours than if the component's entire life was used. If, however, the test and maintenance activities returned the component to a "good-as-old" condition (i.e., component returns to same point on its failure rate curve), there would be no change to the failure rates calculated here.

In future work, the feasibility of removing testing and maintenance effects from the failure data should be investigated. This would allow a more detailed analysis of their effects on component failure rate. Since this was not the primary goal in this study, the test and maintenance effects were included in the failure data. Although the resulting failure rates are appropriate for investigating aging effects on system unavailability and component importance, they should be used with caution for other applications.

The time-dependent NPRDS failure rates are shown in Figures 4-25 through 4-29 for pumps, heat exchangers, MOVs, check valves, and manual valves, respectively. The figures also show constant failure rates from various other sources. The curves used in the figures to represent the failure rates calculated from NPRDS data are curve fits generated by the computer software program (Harvard Graphics)*. These curves are shown only as an aid in visualizing and interpreting the data and are not meant to indicate any specific relationship between the data points. Application of these failure rates is based on models, which are discussed latter in this section.

Although there is uncertainty and scatter in the seen from the NPRDS data, some components show a trend toward increasing failure rate with advancing age. This indicates the possibility that as components age, the chance of a failure occurring in any given period of time increases. The result is that component, and therefore system unreliability will also increase with age. This is a significant result since current risk analyses assume a constant failure rate with time. With an increasing failure rate, reliability could reach an unacceptable level as a plant approaches the end of its design life.

Various humps and valleys exist in the failure rate curves. These can be attributed to numerous causes including errors in determining or reporting the component's age, incompleteness of the data, or uncertainties in the population estimates. They could also be due to periodic maintenance which improves the condition of the component. Since there are many uncertainties in the data, no firm conclusions can be drawn regarding the local variations. The overall trends, however, are expected to be representative of actual conditions. The effects of the local variations are accounted for by the sensitivity studies discussed in Section 7.

* Harvard Graphics, Version A, Copyright 1986 Software Publishing Corporation

As a check on the failure rates calculated from the database information, actual plant data from Indian Point-2 was used. The amount of data was not large enough to determine failure rates as a function of time; however, average values were obtained. The Indian Point data is discussed in more detail in Section 5.

In Figure 4-25, the pump failure rate calculated from the NPRDS data is compared with failure rates obtained from the Indian Point plant data (discussed in Section 5), the WASH-1400 report, and a typical plant PRA analysis. All values show good agreement and are within an order of magnitude of each other. It is noted, however, that the value used in the plant PRA was the lowest of all the failure rates and probably represents a very optimistic estimate. The same is true for the manual valve failure rate, as shown in Figure 4-29.

To apply the failure rates calculated from the data to a PRA analysis, failure rate models using one- or two-line approximations for the failure rate curves were established for each of the components. These models were used as input to the PRAAGE computer program to analyze the effects of aging on system unavailability and component importance versus time.

In modeling the failure-rate curves, only the data from ages two through fourteen were used. The data for one-year old components were not used to assure that infant mortality would not be included. Failure data for components older than fourteen years were not used since there were data only on six plants. It was felt that this data was insufficient for statistical purposes and could be biased by plant reporting characteristics. Some of this data from older plants was examined in the sensitivity studies (Section 7).

Not all failures are reported to NPRDS, therefore, the failure rates calculated from these data are non-conservative. An attempt was made to account for this by adjusting the number of failures reported by dividing by a "reporting factor." Comparison of the Indian Point data (Section 5) with the NPRDS records showed that on average, 31% of all failures were reported to NPRDs. A reporting factor of 0.31, therefore, was used to obtain a "best estimate" failure rate vs. age curve for each component. As will be discussed in Section 7, sensitivity studies showed that the results are not particularly sensitive to the correction for the reporting factor used.

The "best estimate" failure rate curves were then examined to determine the best model for each. For heat exchangers, check valves and MOVs, a single straight line was fitted through the data points by least squares analysis. The one-line model was selected based on subjective judgement of the data. With the variations in the failure rates and the uncertainties in the data, it was felt that a more complex model was not justified and would not provide any additional accuracy. Sensitivity studies were performed to verify this. The failure rate models for heat exchangers, check valves, and MOVs are presented in Figures 4-30 through 4-32.

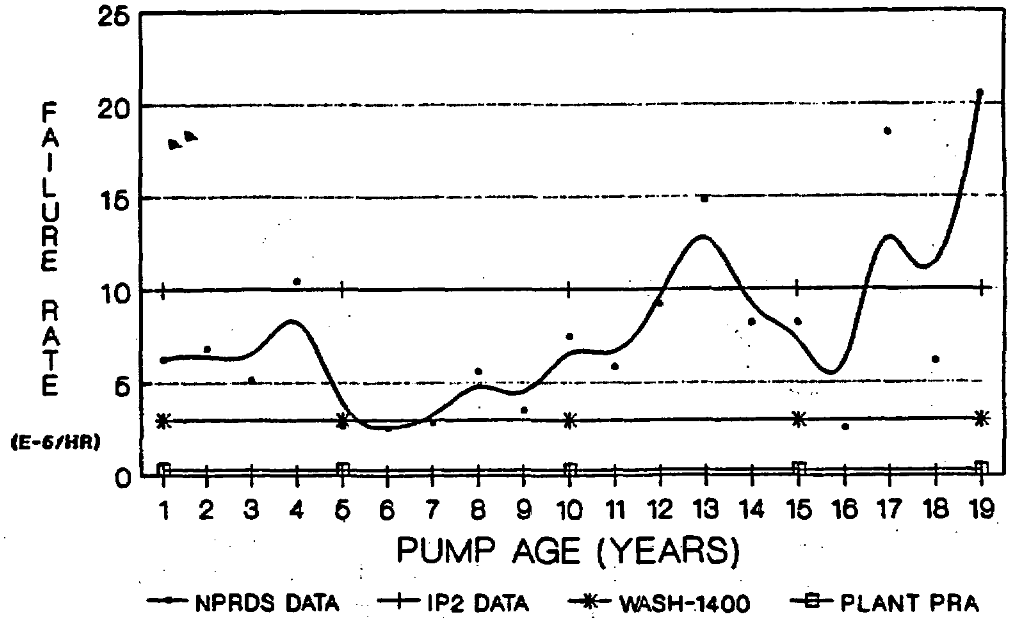

Figure 4-25 Pump failure rate (failure to continue to run) vs. age

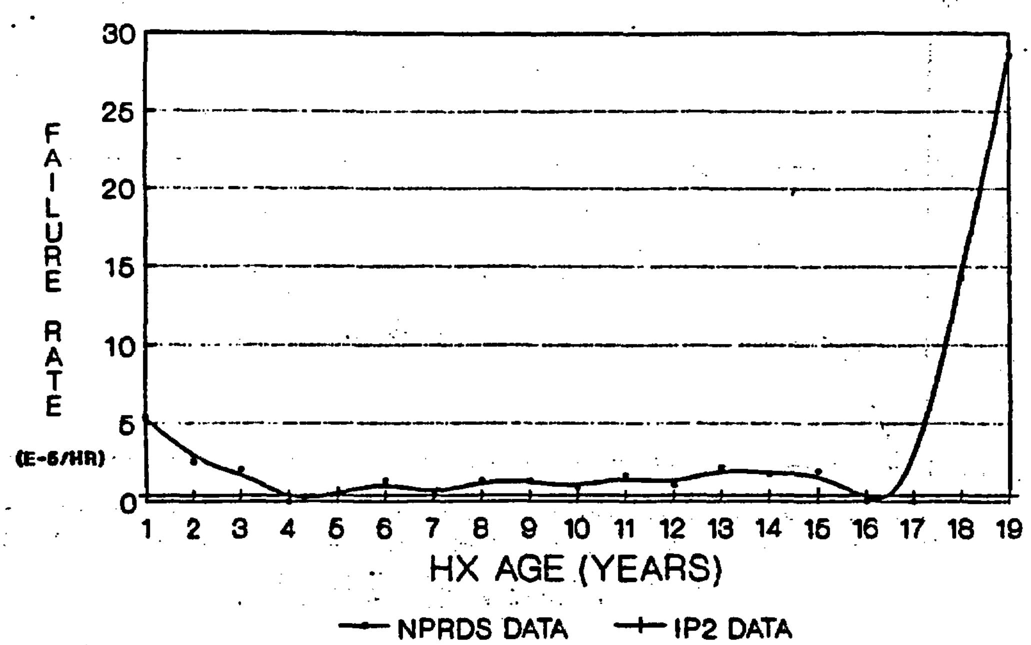

Figure 4-26 Heat exchanger failure rate vs. age

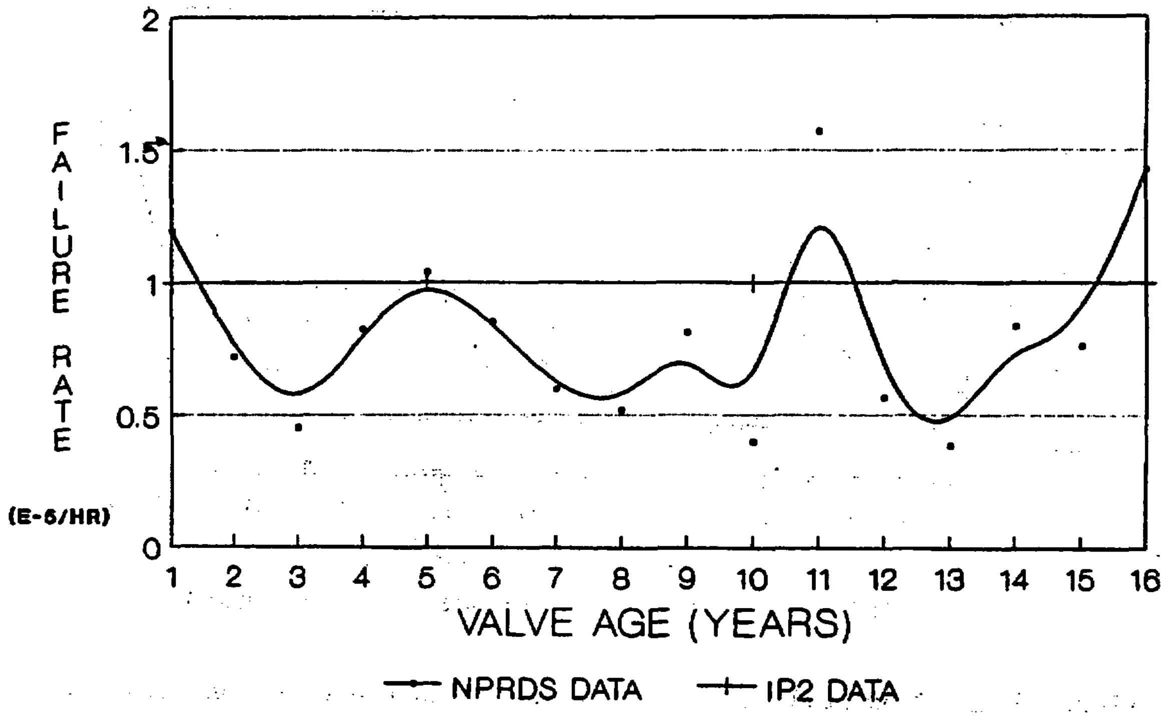

Figure 4-27 Motor operated valve failure rate vs. age

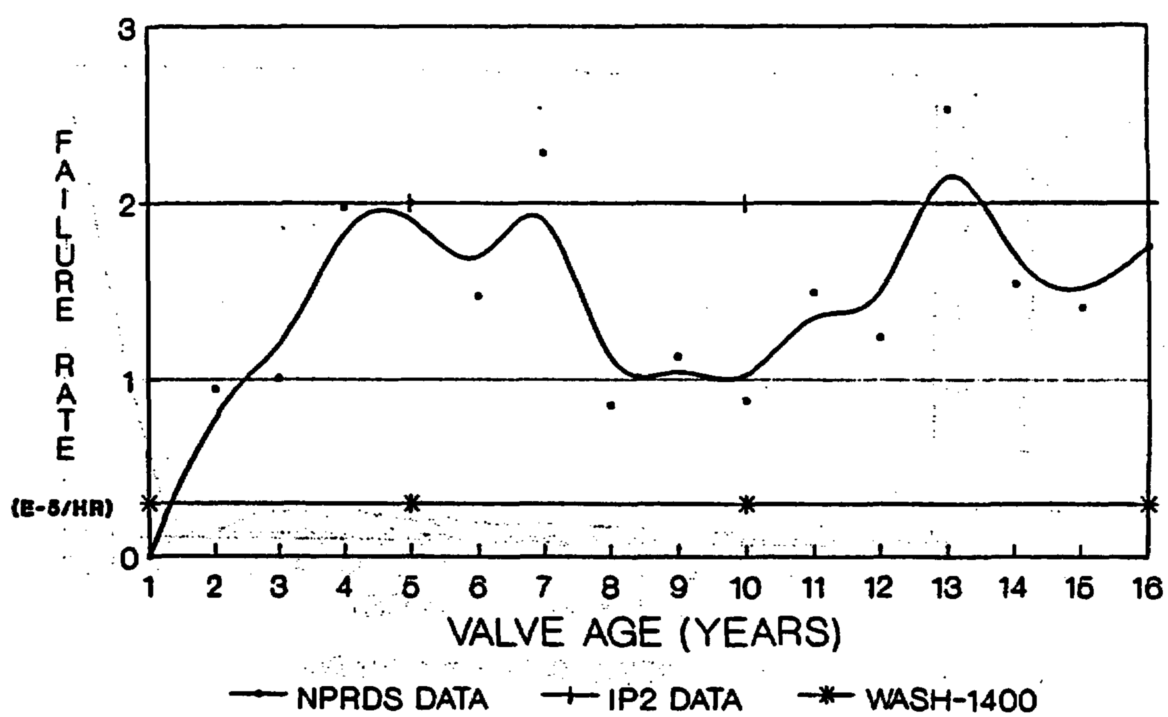

Figure 4-28 Check valve failure rate vs. age

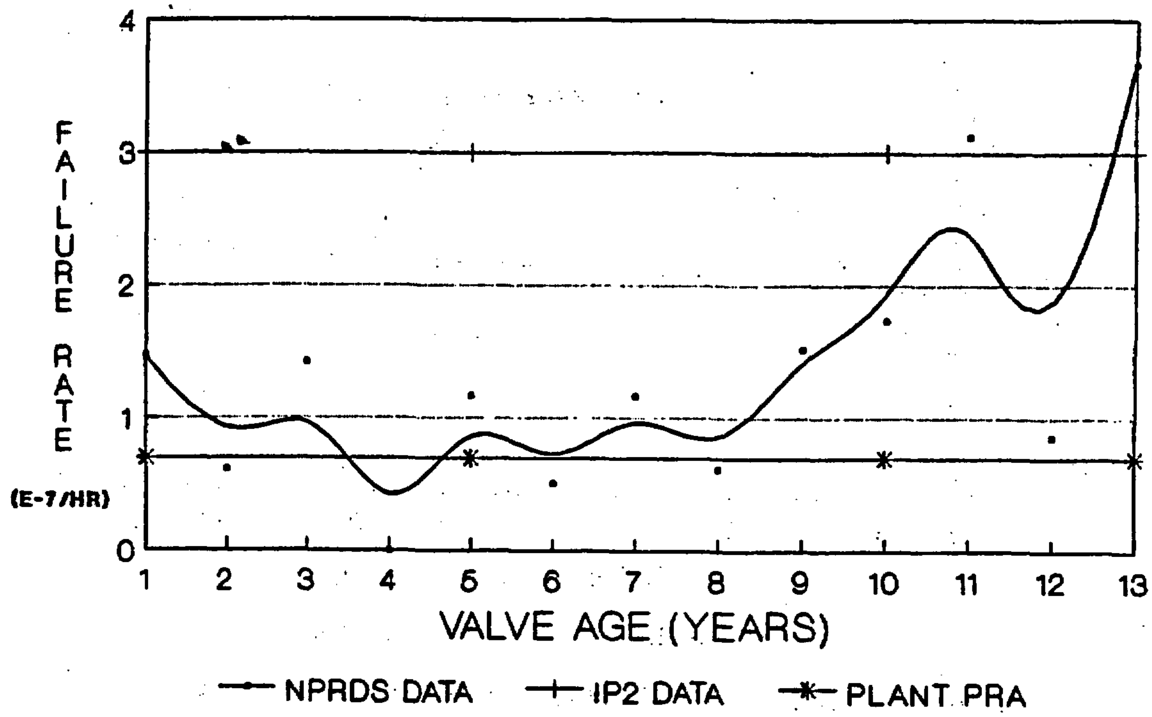

Figure 4-29 Manual valve failure rate vs. age

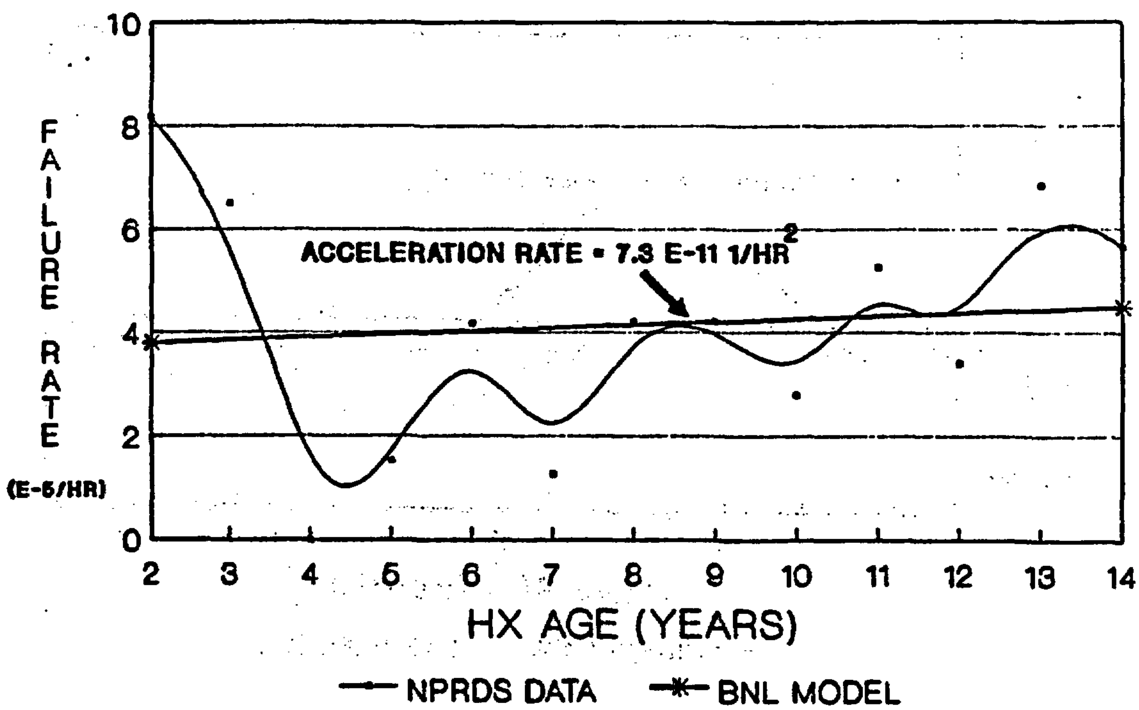

Figure 4-30 BNL model of heat exchanger failure rate vs. age

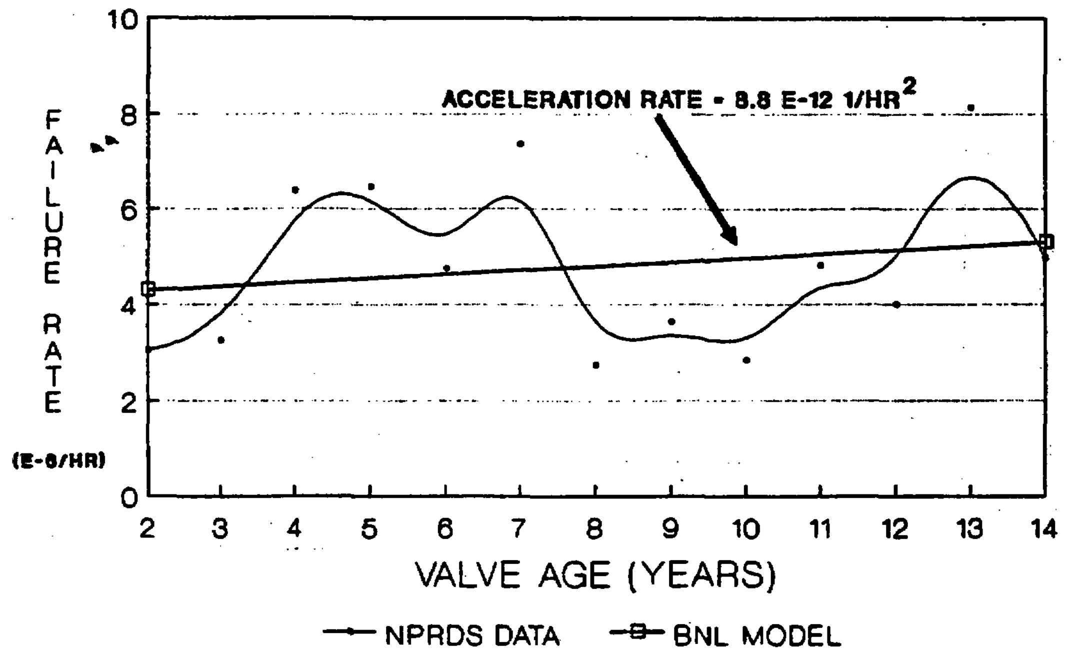

Figure 4-31 BNL model of check valve failure rate vs. age

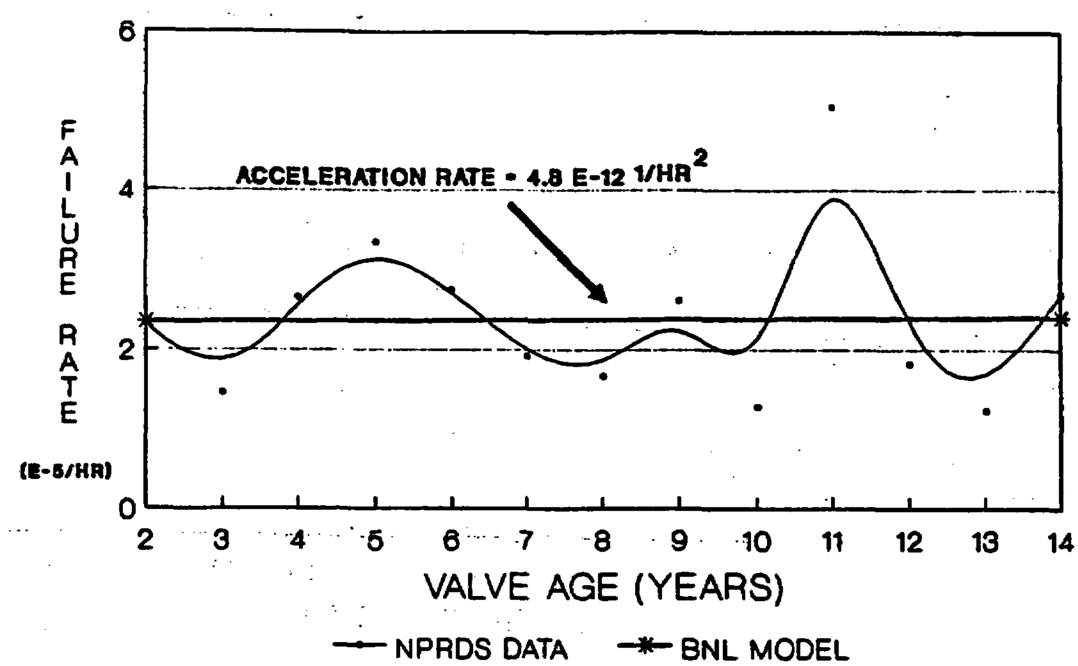

Figure 4-32 BNL model of MOV failure rate vs. age

The acceleration rates indicated on each figure represent the rate at which the failure rate increases with component age, or the slope of the failure rate curve. An acceleration rate of zero would indicate no change in failure rate with time. Another way of expressing the acceleration rate is the Aging Fractional Increase (AFI), which is the ratio of the slope of the failure rate curve to the initial, constant failure rate:

$$\text{Aging Fractional Increase (AFI)} = \frac{\text{Slope of Failure Rate Curve}}{\text{Initial Constant Failure Rate}} \qquad (4.4\text{-}2)$$

The relation between the two parameters is:

$$\text{AFI} = \frac{\text{Acceleration Rate } (1/\text{hr}^2)}{\text{Initial Constant Failure Rate } (1/\text{hr})} \qquad (4.4\text{-}3)$$

The AFI was used as input to the PRAAGE code for the PRA calculations. Table 4-1 shows the acceleration rate and AFI for the various CCW system components.

Table 4-1 Component Aging Acceleration Rates and Aging Fractional Increases

Component	Aging Acceleration Rate	Aging Fractional Increase
Heat Exchangers	7.3×10^{-11} 1/hr^2	1.9×10^{-6} 1/hr
Check Valves	8.8×10^{-12} 1/hr^2	2.0×10^{-6} 1/hr
MOV	4.8×10^{-12} 1/hr^2	2.0×10^{-7} 1/hr
Manual Valve	5.4×10^{-12} 1/hr^2	2.5×10^{-5} 1/hr
Pump	2.9×10^{-9} 1/hr^2	3.2×10^{-5} 1/hr

The failure rate curves for pumps and manual valves (Figures 4-25 and 4-29) show that failure rate remained relatively constant for some period prior to increasing. For these two components it was felt that a two-line model would best represent the failure rate. The model was patterned after the traditional bathtub curve which includes an initially decreasing failure rate, representing infant mortality, followed by a constant failure rate and then an increase in failure rate during the wearout period. Since infant mortality failures were effectively eliminated from the data, only the constant failure rate and wearout portions of the bathtub curve were modeled.

To construct the two-line models, an estimate was made for the age at which the failure rate began to increase with time (estimated aging start time). All failure rates prior to the Aging Start Time (AST) were then averaged to obtain a constant failure rate line. After the AST a least squares analysis of the failure rates was applied to obtain the best straight line. The intersection of the two lines (actual AST) was then determined and compared to the initial estimate. An iteration process was then performed until the estimated and actual AST's converged. The results for manual valves and pumps are shown in Figures 4-33 and 4-34, respectively.

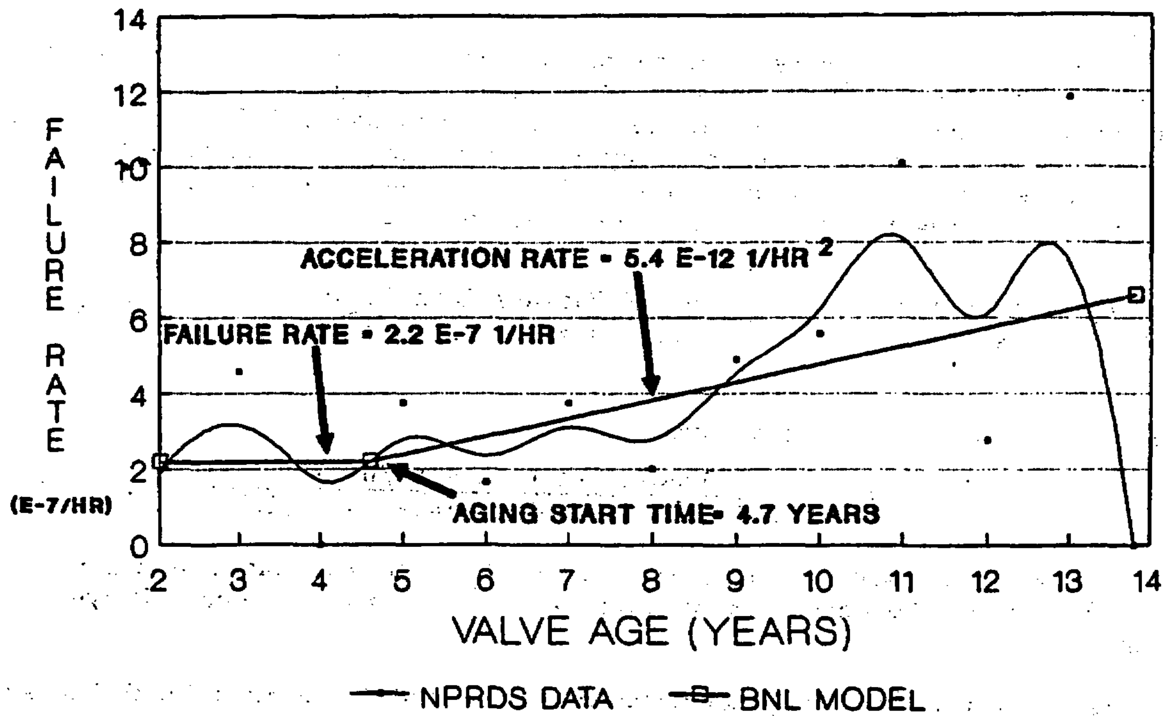

Figure 4-33 BNL model for manual valve failure rate vs. age

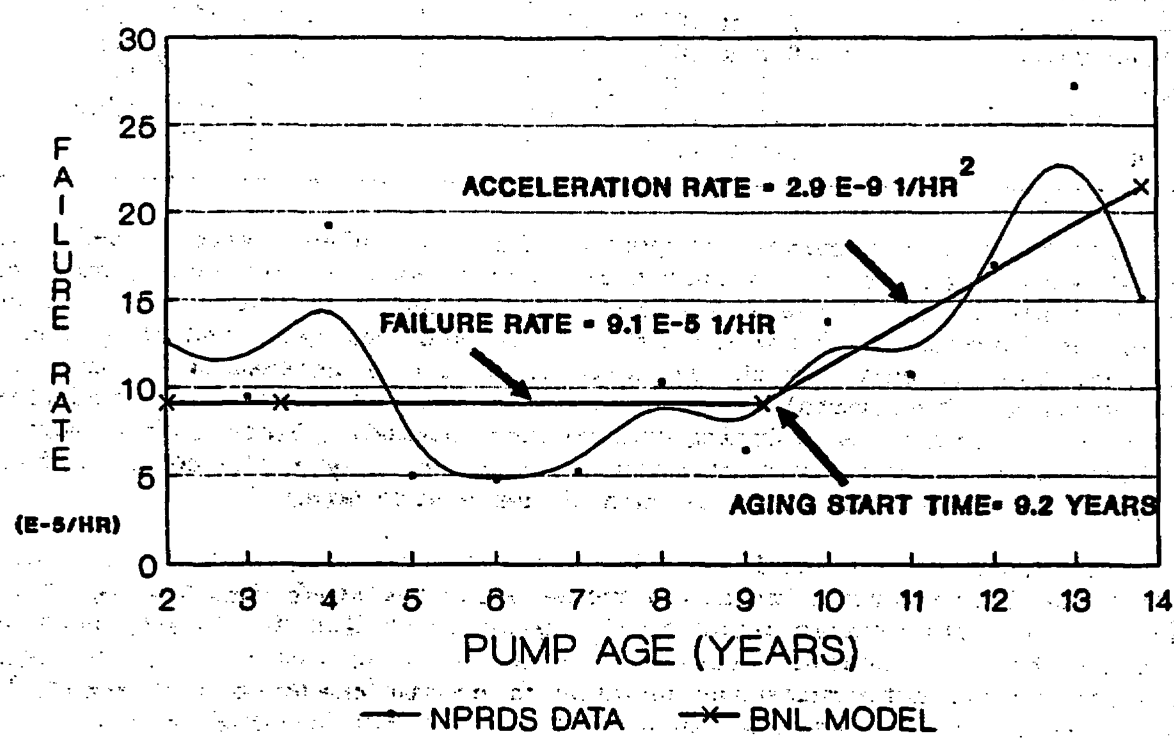

Figure 4-34 BNL model for pump failure rate vs. age

Each of the failure rate models was input to the PRAAGE program to evaluate the effects of the increase in failure rate with age on system unavailability. Since the models have significant uncertainties, parametric studies were performed relative to the acceleration rates and the AST's obtained. The methods used for developing models for the parametric studies were the same as those previously discussed, with the exception of piping.

Since there were only four piping failures in the data, failure rates could not be calculated on a year-by-year basis. As a more meaningful indication of aging effects, the failure rate for piping was calculated for five-year intervals. Each failure rate was assumed to occur at the midpoint of the interval. A least squares analysis was then performed to fit the best line through the failure rates. To avoid having a zero failure rate until the midpoint of the first interval (2 1/2 years), a baseline constant piping failure rate of 8×10^{-10} failures per hour, taken from the Indian Point-2 PRA, was used. The resulting model is shown in Section 7 of this report. Results from the PRAAGE runs, using various parametric models, are discussed in Sections 6 and 7 of this report.

4.5 Summary of Conclusions

Analysis of the failure data provided a great deal of information on aging of CCW systems. The following conclusions were drawn from the data analysis:

(1) At the system level, 72% of all failures in the NPRDS data base were related to aging, indicating that the CCW system is susceptible to aging degradation. This finding is supported by the PRA-type analyses, which are discussed in Sections 6 and 7. Therefore, it is important to monitor and control aging phenomena.

(2) CCW system failures are most often detected when the system is in service, as expected for a normally operating system. This result confirms the need for good functional indicators to detect and monitor aging effects.

(3) Current monitoring methods detect approximately 65% of all failures before the operation of the system is affected. However, improvements to current monitoring methods should be considered so that they can detect failures that result in operational abnormalities or unacceptable levels of performance.

(4) The majority of the CCW system failures resulted in degraded operation of the system or in a loss of redundancy. A reduction in the number of failures resulting in a loss of redundancy would improve system availability. Complete loss of CCW system function occurred rarely (there was only one incident in the data bases) indicating that current system designs have sufficient redundancy to provide good reliability. This data backs up the PRA calculations, which also indicate good reliability.

(5) The predominant cause of failures was shown to be normal service which supports the large fraction of failures due to aging. The second largest cause of failure was human error, most of which were related to maintenance. Efforts to lower the number of failures caused by human error should be directed toward this area.

(6) Leakage was the predominant mode of failure. Inspections for leakage should be considered an important part of surveillance, inspection and monitoring. Aside from leakage, numerous other failure modes are present. Monitoring programs, therefore, must be diverse to detect all failures. This conclusion is supported from examination of the failure mechanisms. Wear was the predominant failure mechanism, but other mechanisms occur in the CCW system.

(7) On a component level, valves were the most commonly reported component to fail. Pumps, instrumentation/controls, and heat exchangers also provided a significant number of CCW system failures. All components had a large aging fraction, indicating that aging occurs in all components of the CCW system.

(8) Looking at the subcomponents, valve failures were dominated by the failure of valve operators, followed by wear of the valve seats. Pump failures were dominated by seal and bearing failures, while heat exchanger failures most frequently involved the tubes. These subcomponents are areas where surveillance and monitoring should be stressed.

(9) Component test frequency was also reviewed as part of this analysis. Results showed that components failed where there was no testing and also where testing was frequent. There was no direct correlation between frequency of testing and failure rate.

(10) As a final part of the analysis, failure rates were calculated as a function of age for several components. In general, failure rates tended to increase with component age. Three components showed increases beginning immediately, while two had a constant failure rate period before the rate increased. An increasing failure rate is significant since it shows that system unavailability will increase as the plant ages, a fact which is not modeled by current PRA analyses. Also, since the failure rates of different components do not increase at the same rate, the relative importance of the components varies with age.

(11) There is good agreement between the calculated failure rates and other sources of failure data. However, CCW system failure rates used in a typical plant PRA analysis are lower than those calculated from the data. This is partially accounted for by the increasing failure rate with time effect. This discrepancy may have implications for many PRA studies.

Models were constructed for each of the failure rate curves for use in the PRAAGE computer code. Results showing the effect of aging on system unavailability are discussed in Sections 6 and 7 of this report.

5. INDIAN POINT UNIT 2: CCW SYSTEM ANALYSIS

5.1 Background

This section of the report describes the detailed analysis performed on the CCW System at Indian Point Unit 2 (IP-2). The analysis consisted of a review of the CCW System design, failure data, maintenance records, testing, operation, procedures, and hardware. The analysis was conducted to understand in depth the operations of a typical CCW System, to correct deficiencies in the various industry-wide data bases, to identify any developing problems in CCW Systems and to document the actions being taken by utilities to ensure proper operation of the system. By reviewing actual site records, "walking-down" the equipment while in operation, and discussing the system with site personnel, a more complete picture could be obtained than by reviewing the nationwide data bases. Also, the complete site failure and maintenance history was compared to the information in the national data bases, to obtain correction factors to be applied when utilizing these data bases and to validate the database findings. IP-2 was selected because it has a typical CCW system and it is more than 14 years old.

The IP-2 CCW System is a normally operating, safety-related system with three main CCW pumps and two CCW heat exchangers cooled by Service Water. Figure A-2 of Appendix A gives a brief description of the IP-2 CCW System; Appendix B describes it in full.

The data presented in this section was collected on two site visits to Indian Point-2. Personnel from the following departments within IP-2 were interviewed: management, maintenance, administration, test performance, quality assurance, operations, and instrumentation and control. A detailed walk-down of the CCW System was performed with utility personnel. Areas of particular interest were the CCW pumps, heat exchangers, surge tank, piping, valves, motor control centers, instrumentation, pipe supports and selected loads. Copies of the procedures, P&IDs, maintenance records, test results and various computer listings were obtained for analysis.

5.2 CCW Failure Data

5.2.1 Discussion

A failure of a component is considered to be a loss of function, either directly due to the circumstances of the failure, or because the component had to be taken out of service for repair. The failures of CCW System components were determined from maintenance records, test records, and the NPRDS data base for IP-2. Failures were sorted by component and arranged chronologically. Comparisons were made between the data bases to determine typically which items and what percent of items are reported to the different data bases. The most complete data came from site maintenance records, as the reporting requirements and time frames for the other data bases varied and did not require that all items be reported. The site maintenance records became more complete after 1984 when they were computerized. Determinations were also made of the predominant failure modes and causes for each component. Failure rates were calculated for each component for comparison to the industry wide data of Section 4 and for use in the PRA analysis of Section 6.

5.2.2 Pumps

The three CCW pumps at IP-2 are horizontal, S-line pumps made by Ingersoll Rand, rated at 3600 gpm and 220 feet TDH. The pumps are driven by Westinghouse, Life Line A motors rated at 250 horsepower and 480 volts. Failure data was obtained for about 14 years, from early 1973 to early 1987. The predominant failures were water leakage at the mechanical seal and failure of the pump bearing (Table 5.1). In the case of pumps, pump motors, and circuit breakers, a single event sometimes included the failure (and replacement) of more than one item; for example, the inboard mechanical seal and inboard bearing. In most cases, the severity of a particular failure was undiscernable, due to the brevity of the descriptions on the Maintenance Work Requests. Hence, it was assumed (for example) that a leak large enough to require replacement of the mechanical seal constituted failure. In actuality, the pump may have been able to continue operation with this leakage.

Table 5-1 Indian Point-2 CCW Pump Failures

Component		Number of Failures**
Mechanical Seals -	Inboard	13
	Outboard	7
Pump Bearings -	Inboard	7
	Outboard	6
	Thrust	1
Pump	- Element	2
	- Alignment	2
	- Coupling	2
	- Packing	1
	- Vibration	1
	- General	1
Motor		5*
Circuit Breaker		1

*Three of the five motor failures constituted one common mode failure of all three motors. This failure is included because it was reported to the database, however, it occurred during a refueling outage when maintenance was being performed.

**The total of this table is larger than the total number of actual failures due to multiple items failing in one event.

Figure 5-1 illustrates the number of pump failures versus plant age. The number is generally quite low and calculation of failure rates is somewhat uncertain. The single failure events (see below for multiple events) appear to occur in three time frames:

- The first two years (burn-in) with a failure rate of 1.9×10^{-4} pump failures per operating hour.
- The middle seven years (mid-life) with a failure rate of 4.4×10^{-5} pump failures per operating hour, and
- The final five years (aged) with a failure rate of 1.3×10^{-4} pump failures per operating hour.

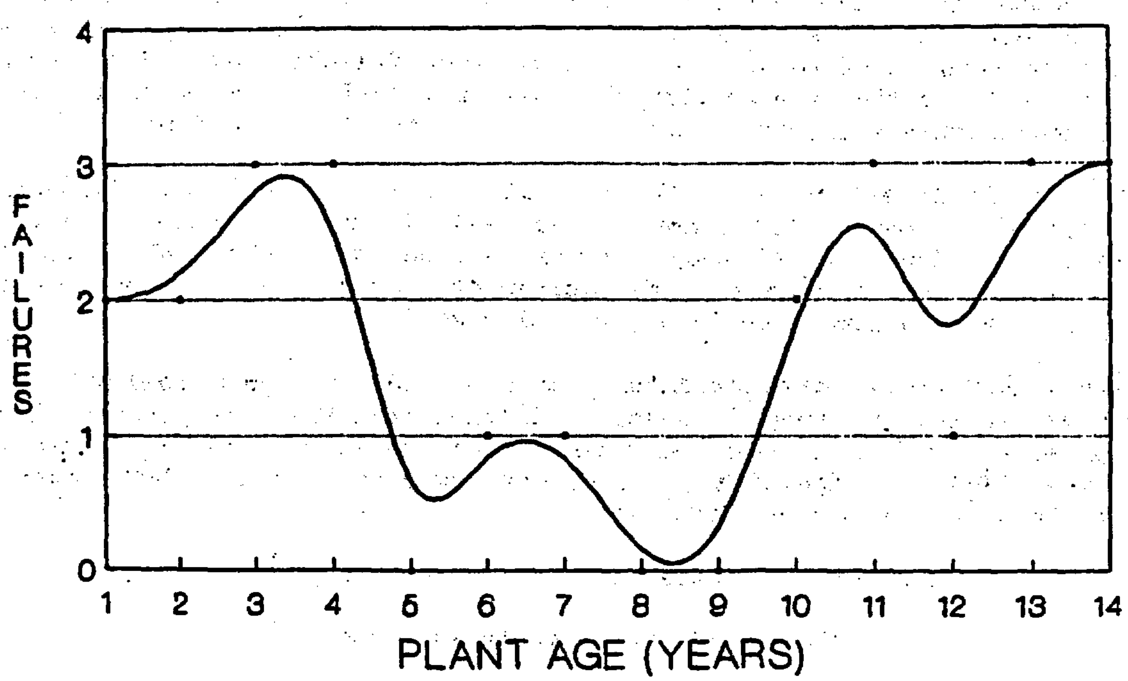

Figure 5-1 Indian Point-2 CCW system pump failures

Due to the difficulty in collecting the maintenance data before 1982, there may be some missing data in the middle years. If so, then the pumps could be exhibiting a relatively constant failure rate with time. Taking the data at face value, we see a higher initial failure rate (or burn-in period), a lower rate for about seven years, and then a higher failure rate as the pumps age. The failure rate does not appear to increase continuously. This seems reasonable in that, as failures occur in the dominant subcomponents (mechanical seals and bearings), they are replaced. Thus, the failure rate may be held at a relatively constant level by effective maintenance.

The average failure rate, about 10^{-4} pump failures/operating hour, compares reasonably well to the general rate for centrifugal pumps given in WASH-1400 of $3X10^{-5}$ failures/hour with upper and lower error bounds of $3X10^{-4}$ and $3X10^{-6}$ respectively. The Indian Point Probabilistic Safety Study (PRA) used a lower value for CCW pump failure-to-run of $2.76X10^{-6}$ failures/hour. This value was derived from the LER data base, which did not contain most of the failures in the maintenance records since failure of one CCW pump was generally not reportable via an LER.

One event, noted in Table 5-1, had multiple failures. Through an operational/maintenance error, Service Water was sprayed into the three CCW pump motors, causing all three to fail. This was a common-cause failure of the full CCW System. Utilizing this one event, a CCW System common cause failure rate of $6.7X10^{-6}$ failures/operating hour was calculated. This value is similar to common-cause failure probabilities used in some PRAs. However, it should be noted that this event occurred during a refueling outage while maintenance was being performed. This common-cause failure rate, therefore, is not expected to be the same as when full system functionality is required. It is included in this analysis because it was reported to the database.

One should note that the failure to start on demand of a standby pump is typically quite large. For example, the IP-2 PRA uses a value of $6.4X10^{-3}$ failures per demand for CCW pumps should they be in standby and required to start. No data was available on the number of pump starts so a new value of failure to start on demand could not be calculated.

Two other items were determined from the data on pump failure. Each failure was evaluated using the aging definition from the NPAR program plan [9], and 89% were found to be aging-related. Each failure occurring during the period when IP-2 was reporting to NPRDS was compared with the NPRDS data and 57% were found to be in the NPRDS data.

Table 5-2 summarizes the data for the IP-2 CCW pumps.

Table 5-2 CCW Pump Failure Data

<u>Failure to run</u> (failures per operating hour)

WASH-1400 upper bound	3.0 E-4	Highest
IP-2 Burn-in data	1.9 E-4	
IP-2 Aged Data	1.3 E-4	
IP-2 Mid life data	4.4 E-5	
WASH-1400 mean	3.0 E-5	
WASH-1400 lower bound	3.0 E-6	
IP-2 PRA	2.76 E-6	Lowest

<u>Aging-Related Failures:</u> 89%

<u>Percent Reported to NPRDS:</u> 57%

<u>Dominant Failures:</u> Mechanical Seals and Bearings

5.2.3 Valves

Indian Point-2 primarily uses manual valves in its CCW System (Figure 5-2). However, the system also has four Air-Operated Valves (AOVs) and six Motor-Operated Valves (MOVs) as containment isolation valves. In addition, two MOVs initiate CCW flow to the Residual Heat Exchangers (HX) and one AOV acts as a temperature control valve for the non-regenerative HX. Check valves are used at each CCW pump and in the cooling loop to the Safety Injection Pumps. Relief valves are in the loops to many loads, but not all.

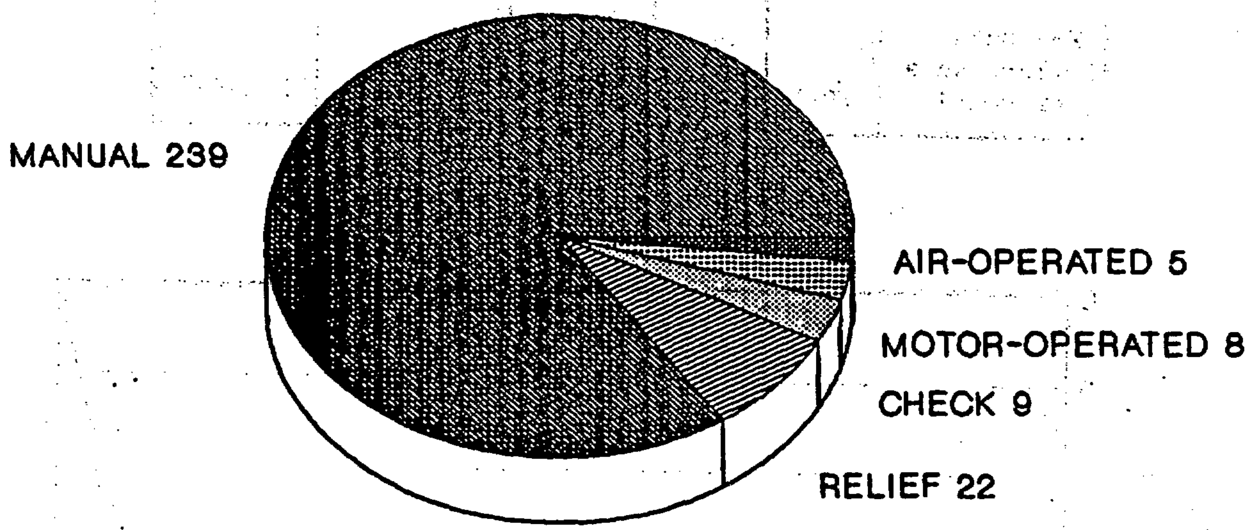

Figure 5-2 Indian Point-2 CCW system valves

Table 5-3 summarizes the failure data for the valves at IP-2. The failures are plotted against the plant's age in Figure 5-3. The failures tend to follow the typical bathtub shape, showing a wear-in period of four years, a low failure rate for four years, followed by an aging or wear-out period. From eight years on, there appears to be an increasing rate of failure, but (as with pumps) the amount of data is not sufficient to reliably quantify this increase. However, a rough aging acceleration rate can be determined by taking a linear approximation to the increase in failures. If this is done over the final eight years, an aging acceleration rate of 1.75 E-3 failures/yr^2 or 2.3 E-11 failures/hr^2 is obtained.

Table 5-3 Indian Point-2 Valve Failure Data

Valve type:	MOV	AOV	Check	Manual/RV	Total
Number	8	5	9	261	283
# in PRA	0	0	3	20	23
Failure Rate used in PRA (per hour)	-	-	7E-5	7.4E-8	
# Failures IP-2 Data	11	3	2	9	25
IP-2 Data Failure Rate (per hour)	1E-5	5E-6	2E-6	3E-7	7.2-E-7

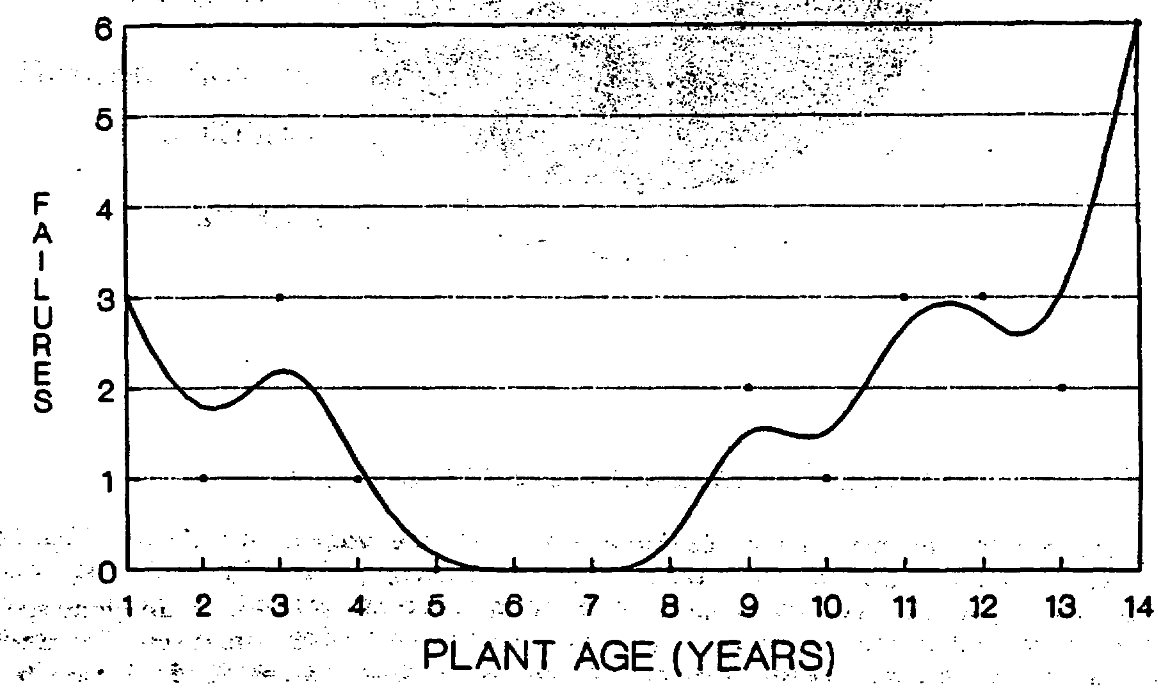

Figure 5-3 Indian Point-2 CCW system valve failures

As with pumps, the aging fraction and percent reported to NPRDS were calculated as follows:

aging-related valve failures: 92%

valve failures in NPRDS: 15%

The types of failures varied in their mode (Figure 5-4). There were also certain valves that failed more than once. For manual valves, the failures were usually caused by broken components such as the valve stem, gate, yoke and handwheel. The check valves failed by seat or flange leakage. Failures of the MOV's generally involved the valve operator due to torque switch failures, limit switch out of adjustment, worn gears and shorted wires. Stem wear, a clutch problem and one stuck valve, were also noted. The AOV failures were due to a limit switch, a broken wire on the transducer, and a stuck valve.

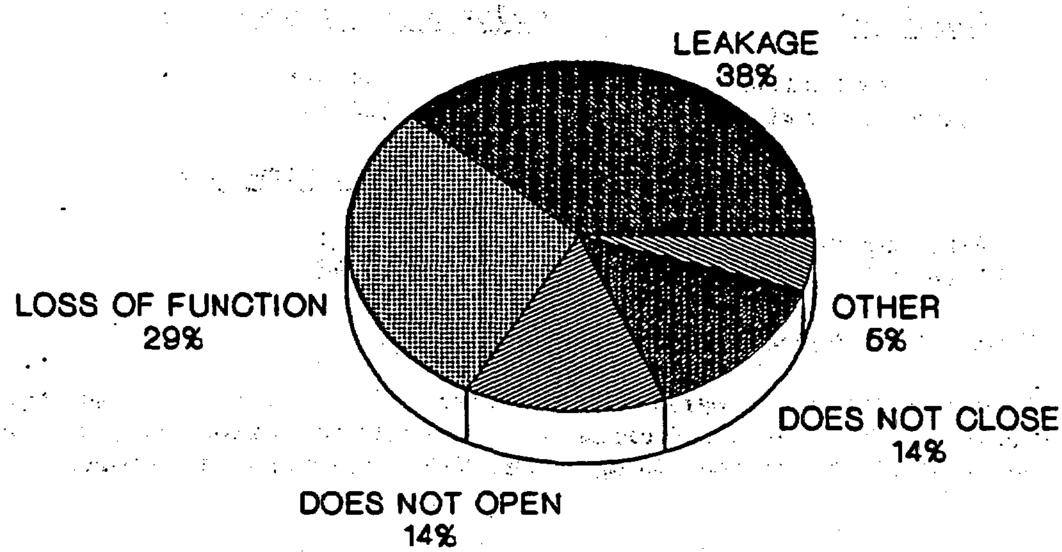

Figure 5-4 Indian Point-2 CCW system valve failure modes

5.2.4 Heat Exchangers (HX)

In fourteen years of operation there has only been one HX failure, which occurred at 10.5 years. This was a pinhole leak in the HX head on the Service Water side, resulting in a spray of Service Water. Although operation could have continued with this minor leak, it was considered to be a failure for this study since the HX was taken out of service for repair. This one event translates to a failure rate of 4E-6 failures per HX per year assuming a constant failure rate. There is not enough data to establish a trend of the failure rate with age, although the mechanisms involved (corrosion and wear) and the component's age at the time of failure would imply an increasing failure rate with age.

5.2.5 Miscellaneous Failures

In addition to the failure of pumps, valves and HXs, 15 other component failures occurred between 1981 and 1987, as follows:

Instrumentation - 8
Pipe supports/restraints - 4
Circuit breakers - 2
Pipe - 1 (at age 11)

No data was available prior to 1981 for these components.

Of the above failures, twelve out of the 15 (80%) were aging-related and two out of the 15 (13%) were reported to NPRDS.

Failure rates were computed as follows:

Component	Failures per component year
Instrumentation	5.4 E-6
Circuit Breakers	1.4 E-5
	Failures per system year
Pipe Supports/restraints	8 E-5
Pipe	2 E-5

5.2.6 Summary of Component Failures

This subsection summarizes the data given in the preceding four subsections. Table 5-4 summarizes the percent of failures at Indian Point that were related to age according to the NPAR definition of aging.

Table 5-4 IP-2 Aging-Related Failures

Component	# Failures	#Aging-Related	%Aging-Related
Pumps	27	24	89
HXs	1	1	100
Valves	25	23	92
Misc.	15	12	80
Total:	68	60	88

The aging-related fraction is consistently in the 80 - 100% range.

Table 5-5 summarizes the failures that were reported to NPRDS.

Table 5-5 IP-2 NPRDS Summary

Components	Total # Failures in NPRDS Time Frame	# Failures Reported to NPRDS	% Reported to NPRDS
Pumps	23	13	57
Valves	20	3	15
HX	1	0	0
Misc.	15	2	13
Total:	59	18	31

As discussed before, the low percent of failures reported to NPRDS is probably due to several reasons including the difference in the definition of failure and the variability in reporting over the years. This percentage was used as a correction factor for the data in Section 4. Table 5-6 summarizes the average failure rates of components determined from the Indian Point site data and, where it was possible to establish, it indicates the trend in failure rate with time.

Table 5-6 IP-2 Failure Rate Summary

Component	Average Failure Rate (failures per component operating hour)	Trend
Pumps	1E-4	Steady with age
MOVs	1E-5	Bathtub curve
AOVs	5E-6	-
Check Valves	2E-6	-
Manual Valves	3E-7	Increasing with age
Total Valves	7E-7	Bathtub curve
HXs	4E-6	Probably increasing with age
Instrumentation	5E-6	-
Circuit Breakers	1E-5	-

The CCW test, maintenance and calibration programs for IP-2 are discussed in Appendix B, along with additional details of the system design and operation.

6. PRA MODEL OF CCW SYSTEM

In this section of the report, the probabilistic work will be discussed. This work included the use of a computer program to perform PRA calculations at various plant ages. The effects of aging on system unavailability and component relative importance were investigated using this program.

Section 6.1 describes the theory used in modeling aging. In Section 6.2, the PRA model of the Indian Point-2 CCW system used in this study is discussed. The computer program developed for this study is described in Section 6.3, while Section 6.4 presents the baseline results obtained. Section 6.5 discusses the projections of system unavailability to 40 year plant life and component prioritization.

6.1 Mathematical Models of Aging

6.1.1 Definition of Aging

Aging is universally familiar but lacks an operational definition for use in a mathematical model. A common definition of aging, which relates to the aging of equipment, is the length of time during which a being or thing has existed; this refers to the total life of the equipment but not to life-shortening processes. Another definition of age is the latter period of a natural term of existence.

Figure 6-1 shows a curve commonly used to represent the failure rate of equipment. This is characterized by a region of early failures called "burn-in" or "infant mortality" followed by a region in which the probability of failing per year is relatively constant called "mid-life" and then an end of life region often called "wearout". The wearout region is of primary concern since aging can cause significant increases in failure rate. For large equipment, the sparseness and heterogeneity of data result in large uncertainties and the form of these distributions are poorly known, but it is generally assumed to be of approximately the same form. Electronic failure rate data, based on many failures, exhibit a region of constant failure rate, but mechanical data, although sparse, seem to show a much shortened mid-life region. In nuclear power plants, the burn-in region may be difficult to discern because of the extensive preoperational testing.

The aging curve may be better understood by a simple model for which the rate of components failing, $dN(t)/dt$, is related to the number, N, that can fail through a parameter $\lambda(t)$:

$$\frac{dN(t)}{dt} = -\lambda(t)\, N(t) \qquad (6.1\text{-}1)$$

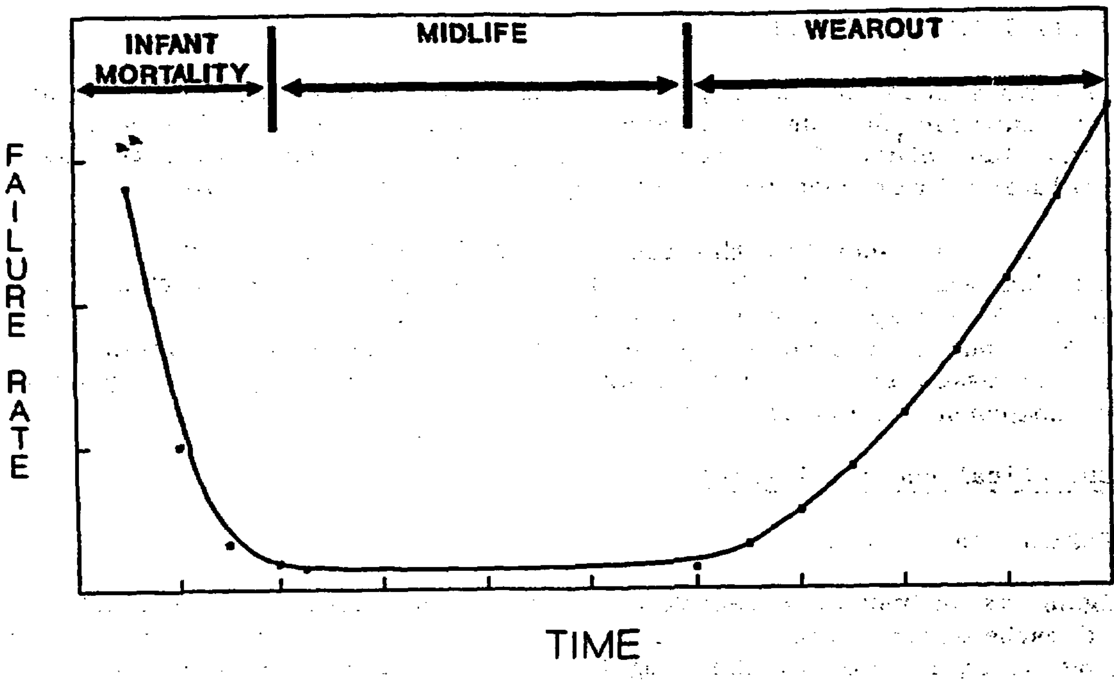

Figure 6-1 Representation of Failure Rate

If $\lambda(t)$ is constant, i.e., no aging, λ is a proportionality parameter. To solve the differential equation, assume N_o is the number of components at the beginning. Then divide through by N_o and take the limit as N_o goes to infinity, so that the ratio N/N_o is the probability of success (p):

$$\frac{dp}{dt} = -\lambda(t)p \qquad (6.1\text{-}2)$$

The resulting equation may be integrated to give:

$$p = e^{-\int_0^{\tau} \lambda(t)dt} \qquad (6.1\text{-}3)$$

The basic question of concern is "what is the probability that this type of component will fail during the critical time in which it is needed (mission time), usually designated as "tau" (τ). In this CCW study, tau is taken to be 24 hours, which is much shorter than the time scale in which aging takes place.

The boundary conditions for Equation (6.1-3) are p=1 at t=0; i.e., that it is initially working at the beginning of the mission time and at the end of the mission time it is restored good as it was at the beginning of the mission. Because the mission time is much less than the aging time ($\tau \ll t$), the failure rate (λ) may be taken to be the average value (λ_o) over the mission.

With these understandings, and using the fact that the probability of failing is one minus the probability of success, the integration of Equation (6.1-3) may be performed to give the probability of component failure as:

$$1-p = q = 1-e^{-\lambda_o \tau} \approx \lambda_o(t)\tau \qquad (6.1\text{-}4)$$

where the average failure rate (λ_o) over the mission time (τ) is indicated as being slowly time dependent.

Equation (6.1-4) shows that the unavailability (q) is a function of both the failure rate and the time. Thus, a piece of equipment is more likely to fail during a long time period than a short time period, and this likelihood is an exponential function of time. This is the nature of normally working equipment, not regarded as failing due to aging. Thus, it would seem that "normal," i.e., expected wear, is not an aging phenomenon, but aging begins when $\lambda_o(t)$ begins to increase above its value in mid-life when it is relatively constant.

Assigning the cause of failure to aging or non-aging causes by inspection depends on the ability to distinguish between normal random causes of failure and those causes indicative of departure from a constant failure rate. A safety concern related to plant operation is whether or not the time-dependent effects of aging on the components result in significant increases in failure rates. Similarly, life extension addresses whether the increases in failure rates due to aging are acceptable and whether the aging mitigation procedures allow safe operation beyond the age limit of the original design.

6.1.2 Aging Model

Precise data on wearout are generally unavailable and may be different for different components and types of components. Rothbart et al, 1981 [15] constructed a reliability simulator in which "burn-in" and "wearout" were modeled as exponentials. Vesely, 1987 [11] modeled wearout as a failure rate that linearly increases with time based on a statistical rationale and on the fact that the shortage of data will not support a more complex representation.

Another approach is to note that any well-behaved function may be expanded in a Taylor's series - the second term of which is the linear dependency suggested by Vesely. The disadvantage of a power series is its generality which provides no physical insight (a priori information) into the expected form of aging.

For generality, the power series expansion of the failure rate $\lambda(t)$ is:

$$\lambda_o(t) = \sum_{i=0}^{\infty} a_i t^i = a_0 + a_1 t + a_2 t^2 + \ldots \qquad (6.1\text{-}5)$$

This may be substituted into equation 6.1-3 and integrated to give:

$$q_a = 1-e^{-\sum_{i=0}^{\infty} \frac{a_i t^{i+1}}{i+1}} \qquad (6.1\text{-}6)$$

The data presented in Section 5 lacks the statistical accuracy to justify more than a linear model. The model used in the PRAAGE code for modeling the aging effect is

$$\lambda_o(t) = \lambda\ (1+at) \qquad (6.1\text{-}7)$$

where λ is the constant failure rate and a is the rate of change of λ with age.

6.1.3 Repaired "Good-as-Old" or "Good-as-New"

Vesely, 1987 [11] presents two models for the probability of a component failure. The "good-as-new" model assumes that when a component is repaired, it is essentially replaced, i.e. time is set back to zero, to start a new path through its lifetime curve without the period of "burn-in". The "good-as-old" model assumes that a component is restored to operation without replacement so the component is at the same place on the wearout curve as it was before failure. The reality of an actual repair is someplace between these extreme models, but precisely where, is unknown.

For a better understanding of aging, the age-dependent and age-independent parts may be separated in equation 6.1-6 by ignoring the first, non-aging term (a_o), and terms higher than the second to give the contribution to the probability of failure by aging alone as:

$$q(\tau) = 1-e^{-\frac{a_1\tau^2}{2}} \qquad (6.1\text{-}8)$$

For "good-as-new," time (t_N) is the time of the last good as new test or maintenance. The probability of failure is then:

$$q(\tau) = 1-e^{-\frac{a_1}{2}(\tau-\tau_N)^2} \qquad \text{"good-as-new"} \qquad (6.1\text{-}9)$$

For "good-as-old," the time interval is encompassed in the limits of the integral from the beginning t_N to give:

$$q(\tau) = 1-e^{-\frac{a_1}{2}(\tau^2-\tau^2_N)} \qquad \text{"good-as-old"} \qquad (6.1\text{-}10)$$

where only the aging aspect of the failure process is being modeled.

The PRAAGE code, used in this work assumes the latter model ("good-as-old") by restoring the failure rate to the value it had at the beginning of the mission time thereby allowing aging to continue according to the aging model.

6.1.4 Applicability of the Linear Approximation of an Aging Curve

Figure 6-2 is the wearout curve previously presented as Figure 6-1, with two simplified models superimposed on it. Line A approximates the aging curve in the wearout region and line B approximates the full curve. Line A is a reasonably good approximation of wearout but it must start at a retarded time, therefore, it must be specified by at least two parameters: the time at which wearout begins and the slope of the line. Line B requires only one parameter: the slope of the line, however, this model does not closely match the actual curve over most of the plant's life.

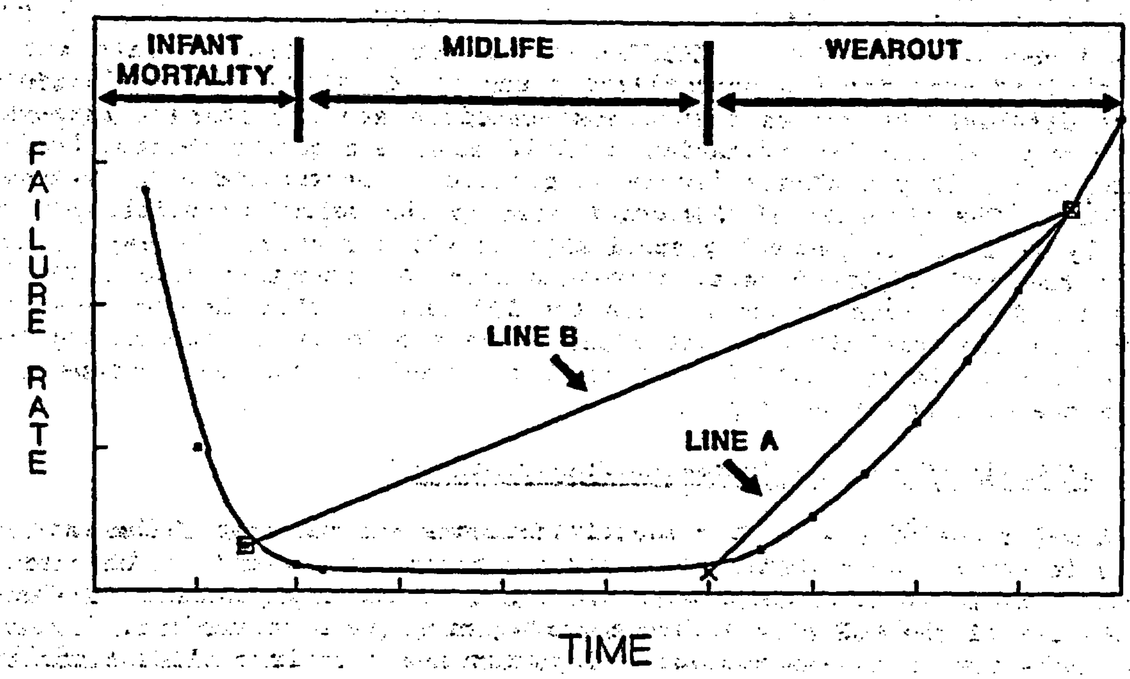

Figure 6-2 Approximations of the Failure Rate Curve

This work began with a one-parameter model. When the field data were analyzed (Section 4) it was found that there was a region of constant failure rate followed by linear aging for some components, thus requiring two parameters (Line A). These parameters are in addition to the non-wearout failure rate. With these considerations, the aging model (6.1-7) becomes:

$$\lambda_o(t) = \lambda \qquad \text{for} \quad t \leq t_o \qquad (6.1\text{-}11)$$

$$\lambda_o(t) = \lambda(1+at) \qquad \text{for} \quad t > t_o \qquad (6.1\text{-}12)$$

where t_o is the time of aging onset.

6.1.5 Test and Maintenance

The work presented here uses the operating history of CCW components at nuclear power plants. As such, the effect of test and maintenance (T&M) practices is reflected in the data. In this sense, the failure rates obtained from the field data are the effective failure rates and include the effects of T&M when used in the previously presented models of aging.

It may be possible to mitigate the effects of aging through improved T&M procedures and practices. To study this possibility, it would be necessary to remove the effects of T&M from the failure rate data, and use the resulting "purified" data in reliability codes such as FRANTIC (Vesely, 1980[16]; Ginzburg and Powers, 1984[17]) or SOCRATES (Wagner et al., 1985[18]) with the T&M model that is proposed for accomplishing the aging mitigation. This process is made difficult by the fact that T&M practices are not uniform throughout the industry, so this "purification" must be done on a plant-by-plant basis or by categories, if the plants may be so grouped or approximated by a generic T&M model. The theoretical T&M model used in the selected reliability code similarly will not be generally applicable to the industry. It may be noted that in the work reported here, there has been no attempt to remove T&M from the data and then reintroduce it into the PRA model. This is due to several reasons: 1) the complexities just cited, 2) the effects of removal and reinsertion are at least partially compensating, and 3) for our purposes the effects are considered secondary.

6.2 PRA Model of the CCW System at Indian Point-2

A major use of PRA is in the identification and ranking of the many components in power plants according to their importance to safety. An objective of this work is to rank components according to their importance to the unavailability of the CCW system with consideration given to their age. First, it is necessary to review measures of importance for their applicability to aging, and then obtain the information necessary to calculate the importances, namely the cutset models and the nominal failure rates, as used in the IPPSS.

6.2.1 Measures of Aging Importance

Fullwood, 1987[19] presents seven measures used in assessing the importance of nuclear plant components to safety. To use these for aging, it is necessary only to use time-dependent failure rates in the CCW model and the results will be the aging importance of the components. Not all of these importance measures need be calculated because Fullwood showed that the measures are interrelated and two fundamental measures are sufficient. These are the Birnbaum and Inspection importances. The Birnbaum Importance (BI) is the fractional change in risk for a fractional change in the failure rate of the component:

$$BI = \frac{\partial R}{\partial p_\ell} \qquad (6.2\text{-}1)$$

where R is risk and p_ℓ is the probability of the ℓ-th component failing. For aging analysis at the system level, risk is not the measure of concern. Instead, it is system unavailability.

Fullwood showed that the linearity of a cutset model leads to the Birnbaum Importance being the sum of the cutset probabilities involving the particular component with that component in the failed state. This may be interpreted as the conditional probability of system failure, given that component 1 has failed. The disadvantage of this importance measure is that it may emphasize important but highly reliable components while the less reliable components may be more important to the system's unavailability. This problem may be circumvented by multiplying the Birnbaum Importance by the probability to give a measure called Inspection Importance (II):

$$II = p_\ell \frac{\partial R}{\partial p_\ell} \qquad (6.2\text{-}1)$$

It may be shown (op. cit.) that this is the sum of the cutset probabilities involving the component of interest. It is clear that if the system unavailability is represented by a model having only single cutsets, the probability of failure of a component is its Inspection Importance. Its fractional contribution to the system unreliability is:

$$\text{Fractional Contribution}_\ell = \frac{II_\ell}{U} \qquad (6.2\text{-}3)$$

where the denominator (U) is the system unavailability determined by combining the cutset probabilities in union. It is desirable to measure the component's contribution to the unavailability as in equation 6.2-3 for components that appear in redundant trains. Inspection Importance is a measure of this, but the sum of all Inspection Importances does not add up to the system unavailability because of double counting. This problem is circumvented by defining a new importance measure called Normalized Inspection Importance (NII) as

$$NII = \frac{II_\ell}{\sum_\ell II_\ell} \qquad (6.2\text{-}4)$$

where the denominator is the sum of Inspection Importances for all components. The sum of the Normalized Inspection Importances of all components adds up to "1". They can also be expressed in percentages. The percentage representation is used in the PRAAGE program. If the Normalized Inspection Importance for a component is multiplied by the system unavailability, the result is the component's contribution to the system's unavailability.

6.2.2 Cutset Model of the CCW system

As a demonstration of the methodology for prioritizing the safety significance of component aging, the process begins with the CCW system cutsets. Table 6-1 presents the first, second and third order cutsets of the Component Cooling Water system from the Indian Point Probabilistic Safety Study (IPPSS- chapter 1.5.2.3.7) using the component identifiers used in the IPPSS (see Table 6-3).

Table 6-1 Cutsets for the CCW System in a Matrix Format

First Order	UXV734AC+
	UTK0021L+
	UXV734BC+
	UPPFAILS+
	TSW1NOFL+
	TXV32--C+

Second Order	TXV34--C+	*	TXV35-1C+
	TXV35--C+		UXV766BC+
	UXV766AC+		UHE0022L+
	UHE0021L+		UXV765BC+
	UXV765AC+		TXV34-1C+
			TXV33--C+
			TXV31-1C+

Third Order	JBS-25AD+	*	JBS-22AD+	*	JBS-23AD+
	UXV760CC+		UXV760BC+		UXV760AC+
	UMO0021S+		UMO0022S+		UMO0023S+
	UPM0021S+		UPM0021S+		UPM0023S+
	UCV761CQ+		UCV761BQ+		UCV761AQ+
	UCC0021F+		UCC0022F+		UCC0023F+
	UXV762CC+		UXV762BC+		UXV762AC+
	4BS-25AD		4BS-22AD		4BS-23AD

By expansion of the matrices according to the operations in Table 6-1, all 554 cutsets of the IPPSS model are represented, i.e. no truncation has taken place. Furthermore, there was no truncation in the IPPSS model to represent the CCW system.

This work departs from the conventional manner of presentation of the cutsets and the manner in which IPPSS presents the cutsets, through the use of a matrix grouping. The reasons for using the matrices are that presentation of the 554 cutsets in the conventional form would take considerable space and would convey less information. Another reason is that matrix manipulation greatly simplifies the mathematics of the importance calculations.

6.2.3 Importance Calculations

The base-case calculations of importance were performed using the IPPSS probabilities presented in Table 6-2 as the initial non-aged failure rates (λ_o) to which the linear aging model is applied, as described in Section 6.1. When the field data became available reflecting actual performance, the importances were recalculated and the piece-wise linear aging curve was used. Results from these calculations are presented in Section 6.4.

6.2.4 Unavailability Calculations

System unavailability is the probability that the system will be inoperable when required and is calculated as the sum of the probability for failing in a first, second or third order cutset. This involves the probabilistic summation of the matrix elements shown in Table 6-1, and is done using deMorgan's theorem as the product of the success probabilities for the elements in a matrix. If the sums are combined as shown for the matrices, then the probability of failing in that order of cutset is obtained, thus the total probability of failure is the sum of the probability of failing in any of the three orders. This final combination is the unavailability.

If probabilities are small, they may be added instead of summed probabilistically. PRAAGE was designed for answering "what if" studies of plant availability which often involves setting the probability of failure of one or more systems to "one" to represent the failed condition. This would lead to erroneous results if the approximation of simple addition were used.

6.3 Interactive Computer Model of Aging

6.3.1 Purpose

The computer code PRAAGE (PRA + AGE) was developed to provide a mathematical model for studying the effects of aging in a complex system. While it was written explicitly for the Indian Point-2 CCW, using the model in the IPPSS, it can be modified for studying the aging and unreliability of other systems at Indian Point-2 or other plants.

PRAAGE was designed to interact in real-time so an operator can readily assess the aging importance of components, correct preassigned parameters from values determined from plant specific data or better, generic data, and study methods for mitigating aging without redesign of the system. Details of the PRAAGE code and how to use it are presented in Appendix C.

Table 6-2 Component Identifier, Non-Aged Failure Rate and Description of the Component and Failure Mode From IPPSS

Identifier	Failure Rate	Component and Description of Failure Mode
JBS-25AD	4.1E-5	No power at switchgear bus 5A
JBS-22AD	4.1E-5	No power at switchgear bus 2A
JBS-23AD	4.1E-5	No power at switchgear bus 3A
TSW1NOFL	2.5E-4	SW availability
TXV31-1C	7.4E-8/H	SW supply isolation, 31-1 transfers closed
TXV32--C	7.4E-8/H	SW supply isolation, 32, transfers closed
TXV33--C	7.4E-8/H	SW supply isolation, 33 transfers closed
TXV34-1C	7.4E-8/H	SW inlet to heat exchanger 22 transfers closed
TXV34--C	7.4E-8/H	SW outlet from heat exchanger 21 transfers closed
TXV35-1C	7.4E-8/H	SW outlet from heat exchanger 22 transfers closed
TXV35--C	7.4E-8/H	SW outlet from heat exchanger 21 transfers closed
UCC002?X	Include in UPM002?S	
UCV761AQ	7.0E-5/D	Pump 23 discharge check valve fails to open
UCV761BQ	7.0E-5/D	Pump 22 discharge check valve fails to open
UCV761BQ	7.0E-5/D	Pump 21 discharge check valve fails to open
UHE0021L	8.42E-7/H	CC heat exchanger 21, loss of cooling capability (leak or rupture)
UHE0022L	8.42E-7/H	CC heat exchanger 22, loss of cooling capability (leak or rupture)
UM0002?S	Include in UPM002?S	
UPM0021S	2.8E-6/H 6.4E-3/D	CC pump 21 does not start/does not continue to run

Table 6-2 (continued)

Identifier	Failure rate	Component and Description of Failure Mode
UPM0022S	2.8E-6/H 6.4E-3/D	CC pump 22 does not start/does not continue to run
UPM0023S	2.8E-6/H 6.4E-3/D	CC pump 23 does not start/does not continue to run
UPPFAILS	8.6E-10/H	CC pipe failure
UTK0021L	8.6E-10/H	CC surge tank, leak or rupture
UXV734AC	7.4E-8/H	Valve 734A transfers closed
UXV734BC	7.4E-8/H	Valve 734B transfers closed
UXV760AC	7.4E-8/H	Pump 23 suction valve transfers closed
UXV760BC	7.4E-8/H	Pump 22 suction valve transfers closed
UXV760CC	7.4E-8/H	Pump 21 suction valve transfers closed
UXV762AC	7.4E-8/H	Pump 23 discharge valve transfers closed
UXV762BC	7.4E-8/H	Pump 22 discharge valve transfers closed
UXV762CC	7.4E-8/H	Pump 21 discharge valve transfers closed
UXV766AC	7.4E-8/H	Heat exchange 21 inlet valve transfer closed
UXV766BC	7.4E-8/H	Heat exchange 22 inlet valve transfer closed
UXV765BC	7.4E-8/H	Heat exchange 22 outlet valve transfer closed
UXV765AC	7.4E-8/H	Heat exchange 21 inlet valve transfer closed
4BS-22AD	Included in JBS-2?AD	

Note: For ? substitute 1, 2, or 3.

6.3.2 Modeling Details

Attention should be called to certain modeling details which are important for interpreting results from the PRAAGE code.

The ratio (R) of the unavailability after aging begins to the unavailability before aging begins is a measure of the affect of aging. For the three levels of redundancy this is:

$$R_1 = \frac{q_1(\text{with aging})}{q_1\ (\text{without aging})} = 1 + at \tag{6.3-7}$$

$$R_2 = \frac{q_2\ (\text{with aging})}{q_2\ (\text{without aging})} = (1+at)^2 = (R_1)^2 \tag{6.3-8}$$

$$R_3 = \frac{q_3\ (\text{with aging})}{q_3\ (\text{without aging})} = (1+at)^3 = (R_1)^3 \tag{6.3-9}$$

As stated, a triply redundant system ages as the cube of a singly redundant system with the same aging parameters. Similarly, a doubly redundant system ages as the square of a singly redundant system.

The physical reason why in redundant systems the ratio (R) of the unavailabilities increases faster than in systems of less redundancy, is that age acts as a common cause affecting the systems in a redundancy. This conclusion is not original. Vesely has reported similar results.[11] Figure 6-3 presents the percent contribution to unavailability caused by aging which, in addition to the fact that this is a semi-logarithmic plot, causes the roll over in the curves around 20 years. The reason for the drop in the percentage contribution from pipe is because check valves and pumps are coming into dominance. The rapid rise in check valves and pumps in the region around 10 to 18 years is caused by a combination of the cubic nature of aging when it is affecting a triple redundancy and because of the significant slope of aging for these components, as shown in Figures 4-31 and 4-34, respectively. However, in this case the latter cause predominantly affects the total system unavailability.

In conclusion, although redundancy results in a system of relatively small unavailability, the common cause effect of aging causes a redundant system to age faster than if the sub-systems/components making up the redundancy were considered separately. This is not intended as a criticism of the principle of redundancy but to call attention to the need for additional attention as aging takes place. Because a redundant system was highly reliable at the beginning of life is no reason for complacency in assuming it will remain so when subjected to aging.

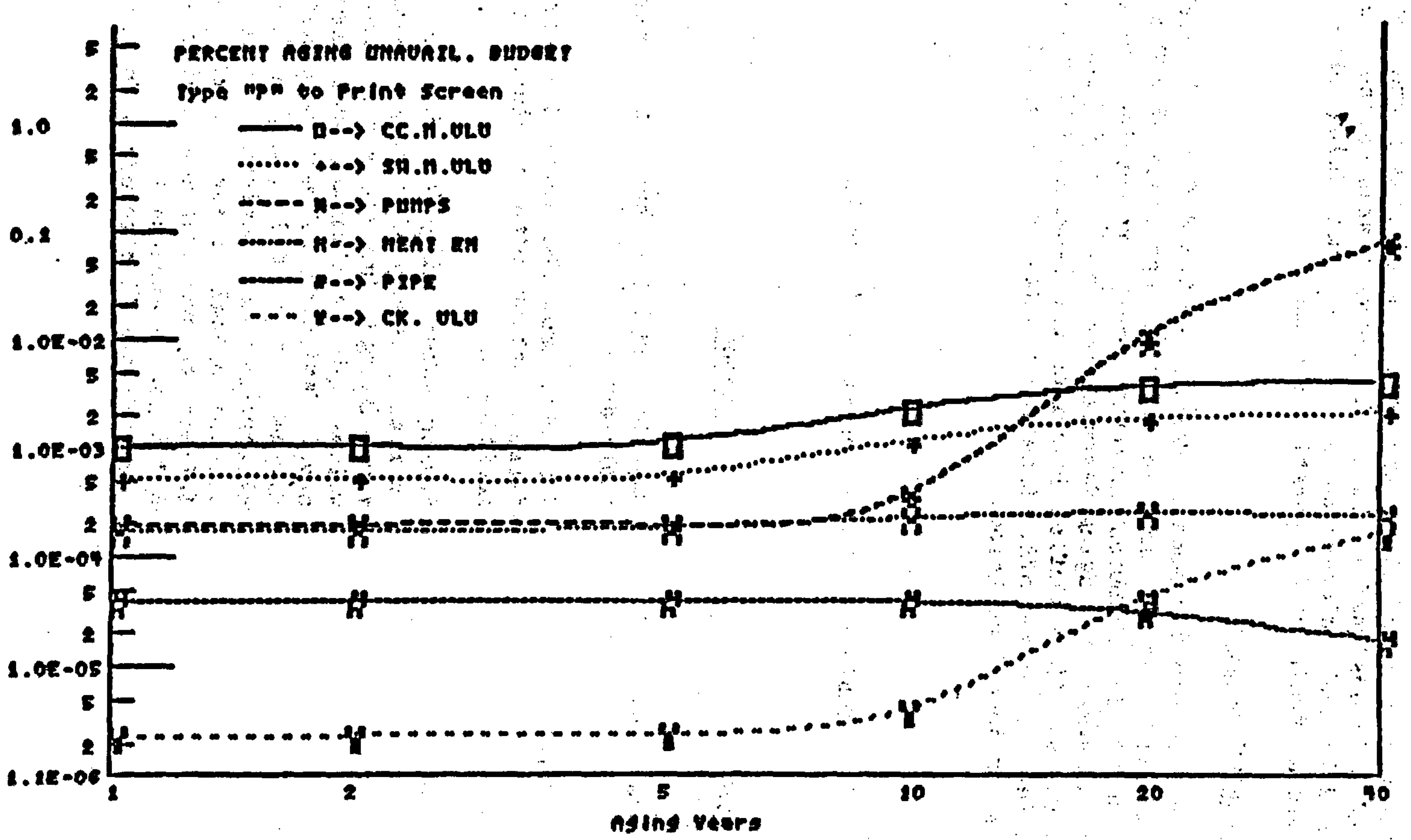

Figure 6-3 PRAAGE Graphical Output - System Unavailability

6.3.3 Characteristics of PRAAGE

A convenient summary of the characteristics of PRAAGE are presented in Table 6-3. PRAAGE is designed to be an independent stand-alone program that contains the models and information needed to study aging effects, while requiring no special training in operating the code.

6.4 Baseline PRAAGE Results

The interactive computer model, PRAAGE, developed for this study calculates CCW system unavailability and the relative importance of CCW system components. This model also computes baseline results by turning the aging feature off, or by utilizing the aging feature, the model can project system performance to 40 years. This section describes the baseline CCW system results with the aging feature turned off. Baseline results are presented for several different cases, as described below.

Table 6-3 Characteristics of PRAAGE

Characteristic	Comments
40 year aging analysis	May be extended but set primarily by tableformat.
Generic default data	The data from the IPPSS is provided as a default option with complete freedom for modification.
Data modification	*Aging menu: aging fraction, aging annual increase, Vesely algorithm. *Outage time menu: outage time for the generic classes: valves, pipe and tanks, heat exchanger, CC pump. *Generic failure rate menu for the classes: pipe and tank, service water, electrical bus, valves: motor operated, valves: manual, valves: check, CC pump - failure to run, CC pump failure to start and heat exchanger. * Individual components - 33 components in 2 tables.

Table 6-3 (cont.)

Characteristic	Comments
System model	All of the 554 cutsets in the IPPSS CCW model are implemented by using the matrix method, and there is no need to truncate cutsets to accommodate the code.
Linear-aging model	PRAAGE uses the linear model but the time of aging onset may be specified. A non-linear model may be implemented if the data support it.
No small-probability approximation	Avoidance of probability approximations allows deterministically failing components to determine the effects of operation in these degraded states.
Importance measures	PRAAGE presents Birnbaum and a newly defined Normalized Inspection Importance. The normalization allows the interpretation in terms of unavailability contribution.
Tabular data presentation	Three generic menus allow the selection of: fractional contribution to unavailability, fractional unavailability contribution per component and incremental contribution to unavailability. Individual component importances are presented by Birnbaum and Normalized Inspection Importance (fractional unavailability contribution).
Printed output	Each of the output menus may be printed.
Plotted output	The generic output may be plotted on an automatically adjusted log-log scale, 6 items at a time with different lines and symbols. Cubic spline fitting is used to show the curvature. Paper copies of the graphs may be plotted on an Epsom or Epsomlike printer.
Typo trapping	PRAAGE uses typo-trapping to avoid or reduce system crash from this cause.

Since PRAAGE has the capability to easily change the component's failure rate data on which system unavailability is computed, PRAAGE results were obtained using three different sets of component failure rate data:

1. Constant failure rates from the Indian Point-2 PRA,

2. Constant failure rates computed from the review of actual Indian Point plant data, described in Section 5.2.

3. Time-dependent failure rates computed from the industry-wide NPRDS review, described in Section 4.

The results of PRAAGE calculations using failure rate data from 1 and 2 above are discussed in this section. These failure rates were fixed numbers and were not time-dependent. They were computed by averaging failures over time and hence represent CCW system experience to date; about 10 years for the Indian Point-2 PRA data and about 14 years for the Indian Point-2 plant data analyzed for this study. The failure rate data derived from NPRDS is time-dependent and is discussed in Section 6.5.

Figure 6-4 lists the failure rate input data from PRAAGE corresponding to the Indian Point-2 PRA.

#3 Menu of Generic Failure Rates

No.	Generic Component	Value
1	Pipe and Surge Tank	8.6E-10/hr
2	Service Water	2.5E-04
3	Electrical Bus	4.1E-05
4	Valve: Manual CCW	7.4E-08/hr
5	Valve: Manual SW	7.4E-08/hr
6	Valve: Check	7.0E-05/hr
7	CC Pump: Failure to Run	2.8E-06/hr
8	CC Pump: Failure to Start	6.4E-03
9	Heat Exchanger: Leak or Rupture	8.4E-07/hr
10	Pump Maintenance Unavail. Contribution	8.5E-02
11	Common Cause Unavail. Contribution	0.0E+00

Figure 6-4 Failure Rates From Indian Point-2 Input to PRAAGE

Most of the failure rates are given in failures per hour. Pump failure to start is per demand. Failure of the supporting systems (service water and electrical power) were converted to failures in 24 hours, and the contribution of pump maintenance to unavailability is a fraction. Using this data, PRAAGE computes a system unavailability of 2.56×10^{-4}. This matches closely with the value calculated in the Indian Point-2 PRA using the same failure rate data but different methods of computation. Table 6-4 lists the percent contribution of each major component to system unavailability.

Table 6-4 Component Importance IP-2 PRA Data

Component Type	Percent Contribution
Service Water (SW)	97.1
CCW - Manual Valves	1.4
SW - Manual Valves	0.7
Pumps	0.5
Pipe	0.2
Check Valves	0.1
Electrical Power	<0.1
Heat Exchangers	<0.1

Failure of the service water system dominates the CCW system unavailability for Indian Point. This was true for all the cases analyzed (Table 6-5). The dependence of plants other than Indian Point on the service water system would depend on the reliability of their specific systems, whose designs vary even more than the CCW systems. Therefore, in the majority of the following analyses the probability of failure of the service water system was set equal to zero, thus allowing an analysis of CCW system component effects. The specific SW system manual valves to the CCW Heat Exchangers were kept in the analysis (see Figure B-3 in Appendix B).

Table 6-5 Comparison of Unavailability Calculations

Case	CCW System Unavailability
Unavailability due to SW system failure alone	2.50 E-4
IP 2 PRA Baseline Model	2.56 E-4
NPRDS Data (Age=0)	2.68 E-4
IP-2 Plant Specific Data	2.73 E-4

Excluding the effect of service water, the importance of the components appear as shown in Table 6-6.

The dominant components are CCW manual valves. This is an aggregate of all manual valves included in the PRA model, however, it is dominated by failure of two manual valves, 734A and 734B, in the header to the Safety Injection Pumps and Residual Heat Removal Pumps (see Figure B-1, Appendix B). Failure of either of these two valves causes a total loss of CCW to these loads. The SW manual valves are dominated by failure of one valve, SWN-32, which would result in a loss of SW to both CCW heat exchangers and a failure of the CCW system. It should be noted that the Indian Point-2 CCW system has no motor-operated valves (MOVs) in crucial portions of the CCW system whose failure could disable the entire system. Hence, MOVs do not appear in the importance rankings.Other NPPs have MOVs in key positions; hence, some of the conclusions concerning the critical IP-2 manual valves would apply to MOVs that are located in critical positions (i.e., such that one valve failure would disable the system).

Table 6-6 Component Importance - IP-2 PRA Data, SW=0

Component	Percent Contribution
CCW - Manual Valves	48
SW - Manual Valves	24
Pumps	18
Pipe	5.3
Check Valves	4.6
Electrical	0.1
Heat Exchangers	<0.1

Figure 6-5 illustrates the input data for PRAAGE derived from the IP-2 plant analysis of Section 5.2. Figure 6-6 illustrates the component importances based on this input data.

#3 Menu of Generic Failure Rates

No.	Generic Component	Value
1	Pipe and Surge Tank	8.6E-10/hr
2	Service Water	2.5E-04
3	Electrical Bus	4.1E-05
4	Valve: Manual CCW	3.0E-07/hr
5	Valve: Manual SW	3.0E-07/hr
6	Valve: Check	2.0E-06/hr
7	CC Pump: Failure to Run	1.0E-04/hr
8	CC Pump: Failure to Start	6.4E-03
9	Heat Exchanger: Leak or Rupture	4.0E-06/hr
10	Pump Maintenance Unavail. Contribution	8.5E-02
11	Common Cause Unavail. Contribution	0.0E+00

* Service water effects eliminated in some runs by setting to zero.

Figure 6-5 Indian Point-2 plant specific data.

Component	Importance (%)
SVC WTR	91.1919
TOT VLV	7.8878
CC.M.VLV	5.2553
SW.M.VLV	2.6284
PUMPS	0.7656
PIPE	0.1430
HEAT EX	0.0092
CK. VLV	0.0042
ELECT.	0.0024

Note: Service Water - Nominal

Component	Importance (%)
TOT VLV	89.5514
CC.M.VLV	59.6638
SW.M.VLV	29.8401
PUMPS	8.6923
PIPE	1.6240
HEAT EX	0.1049
CK. VLV	0.0474
ELECT.	0.0274
SVC WTR	0.0000

Note: Service Water = 0

Figure 6-6 Component importances - IP-2 plant-specific data.

These results again show the dominance of failures of the service water system and the CCW manual valves. Next in importance for the base case are CCW pump failures, but these are considerably lower. The following section discusses the results of PRAAGE using the NPRDS data as input and projecting the results out to 40 years of system operation.

6.5 PRAAGE Projections and Component Prioritization

The time dependent failure rates were used as the input for PRAAGE to project performance of the CCW system into the future. The basis for the time-dependent failure rates is the NPRDS study of Section 4, as validated by the plant specific reviews of Indian Point Unit 2. The results of the projections are system unavailability versus time and also component prioritizations versus time. Since the input data or failure rates increase with time, system unavailability also increases with time. Figures 6-7 and 6-8 list the input failure rate. Figure 6-10 shows the component prioritization including service water (SW) failure: Figures 6-11 and 6-12 show the prioritization with SW failure rate set to zero. Finally, Figure 6-13 plots the CCW system unavailability versus time and shows the rate at which it increases. These figures are discussed in more detail in the following paragraphs.

The combination of PRAAGE Menus #1 and #3, as shown in Figures 6-7 and 6-8, specify the three parameters needed to define the time-dependent failure rate for each component. As an example, Figure 6-9 shows the failure rate function of the CCW pump specified by the three key parameters given in these two menus.

#1 Menu of Aging Parameters

No.	Parameter	Value
	Analysis Time of this Aging Study	40.0 yrs
1	Manual Valve Aging Fract. Increase/Yr	0.21
2	Manual Valve Aging Start Time in Yr	4.70
3	Check Valve Aging Fract. Increase/Yr	0.02
4	Check Valve Aging Start Time in Yr	2.00
5	Pipe and Tank Aging Fract. Increase/Yr	0.00
6	Pipe and Surge Tank Aging Start Time in Yr	2.50
7	Heat Exchanger Aging Fract. Increase/Yr	0.02
8	Heat Exchanger Aging Start Time in Yr	2.00
9	CC Pumps Aging Fract. Increase/Yr	0.28
10	CC Pumps Aging Start Time in Yr	9.20
11	Service Water Aging Fract. Increase/Yr	0.00
12	Service Water Aging Start Time in Yr	10.00
13	Switchgear Aging Fract. Increase/Yr	0.00
14	Switchgear Aging Start Time in Yr	10.00
15	Turn Off(0)/On(1) Aging	1

Figure 6-7 PRAAGE Input - Aging Parameters

From examining Figures 6-7 and 6-8, one can note that manual valves, check valves, and heat exchangers have a two-part failure rate curve similar to that for pumps. Pipes, Service Water, and Switchgear have constant failure rates with time for the base case.

#3 Menu of Generic Failure Rates

No.	Generic Component	Value
1	Pipe and Surge Tank	8.6E-10/hr
2	Service Water	2.5E-04
3	Electrical Bus	7.1E-05
4	Valve: Manual CCW	2.2E-07/hr
5	Valve: Manual SW	2.2E-07/hr
6	Valve: Check	4.3E-06/hr
7	CC Pump: Failure to Run	9.1E-05/hr
8	CC Pump: Failure to Start	6.4E-03
9	Heat Exchanger: Leak or Rupture	3.8E-05/hr
10	Pump Maintenance Unavail. Contribution	8.5E-02
11	Common Cause Unavail. Contribution	0.0E+00

Figure 6-8 PRAAGE Input-Generic Failure Rates

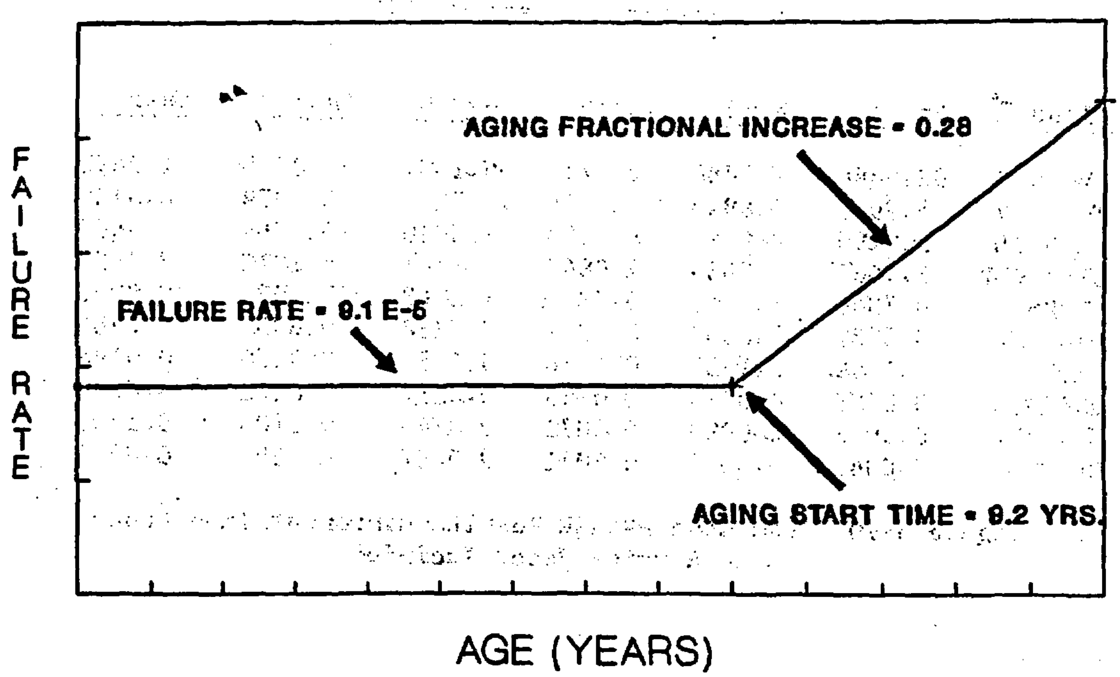

Figure 6-9 CCW Pump Failure Rate Model Input to PRAAGE

Figure 6-10 shows that the SW system initially dominates unavailability at 92%, but that pumps increase in importance over the forty years until they become dominant at 81% in year 40. In Figures 6-11 and 6-12, the effects of SW are removed. The initial dominance by manual valves at 80%, which increases up to year 10 is shown. Between years 10 and 20, pumps pass valves in importance and continue to increase up to year 40. The initial increase in valves is due to the fact that for valves the aging start time is at 5 years, while the aging start time for pumps is 9 years. The pumps, however, age at a faster rate and they are in triple redundancy. This causes a three-fold effect due to the increase in their failure rate. At age 40, pumps comprise 92% of the unavailability. This information would be useful in later years for prioritizing maintenance actions if resources are limited.

#8 System Unavail. and Generic Inspection Aging Importance
All Numbers in Percent (%)

Class	Year1	Year2	Year5	Year10	Year20	Year40
SVC WTR	92.5888	92.5888	92.1753	86.1031	55.8725	11.9693
TOT VLV	5.8945	5.8945	6.2476	11.7310	15.3979	6.8109
CC.M.VLV	3.9192	3.9192	4.1537	7.8010	10.1864	4.4467
SW.M.VLV	1.9665	1.9665	2.0845	3.9161	5.1010	2.1936
PUMPS	0.7340	0.7340	0.7318	1.2463	27.9093	80.9056
HEAT EX	0.6333	0.6333	0.6967	0.7788	0.6935	0.2480
PIPE	0.1452	0.1452	0.1446	0.1351	0.0876	0.0188
UNAVAIL	0.0268	0.0268	0.0269	0.0287	0.0362	0.0954
CK. VLV	0.0088	0.0088	0.0093	0.0140	0.1105	0.1706
ELECT.	0.0041	0.0041	0.0041	0.0057	0.0391	0.0474

Figure 6-10 Base Case PRAAGE Results-Component Importance with Service Water Included

#8 System Unavail. and Generic Inspection Aging Importance
All Numbers in Percent (%)

Class	Year1	Year2	Year5	Year10	Year20	Year40
TOT VLV	79.5348	79.5348	79.8440	84.4147	34.8942	7.7370
CC.M.VLV	52.8820	52.8820	53.0849	56.1346	23.0840	5.0513
SW.M.VLV	26.5336	26.5336	26.6405	28.1793	11.5598	2.4919
PUMPS	9.9046	9.9046	9.3524	8.9680	63.2470	91.9061
HEAT EX	8.5456	8.5456	8.9035	5.6044	1.5717	0.2817
PIPE	1.9597	1.9597	1.8479	0.9719	0.1986	0.0213
CK. VLV	0.1191	0.1191	0.1185	0.1008	0.2504	0.1938
ELECT.	0.0553	0.0553	0.0522	0.0410	0.0885	0.0539
UNAVAIL	0.0018	0.0018	0.0019	0.0037	0.0112	0.0705
SVC WTR	0.0000	0.000	0.0000	0.0000	0.0000	0.0000

Figure 6-11 Base Case PRAAGE Results-Component Importance with Service Water Excluded

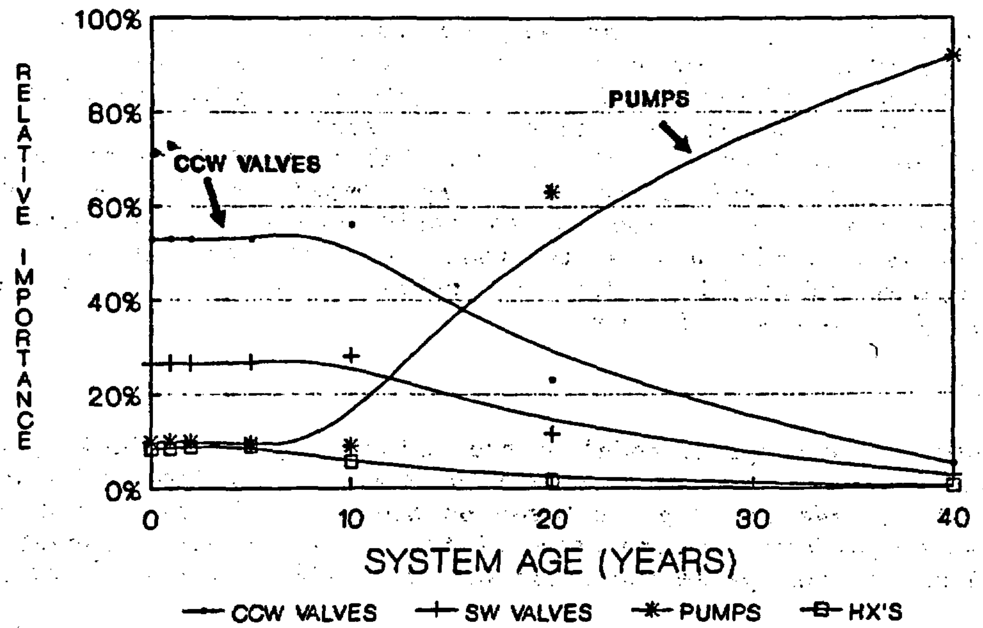

Figure 6-12 Component Importances Versus Time

Figure 6-13 shows CCW unavailability increasing by a factor of 40 over its lifetime to a value of 7×10^{-4} (without considering service water effects). Fullwood[20] discusses the effects on system unavailability from the degradation of multiple redundant components.

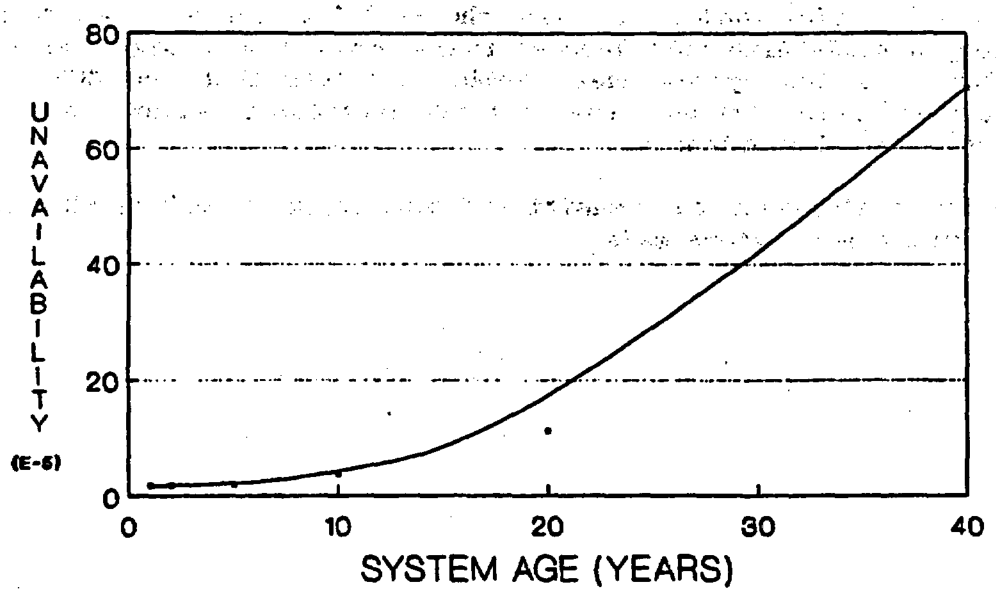

Figure 6-13 CCW System Unavailability Versus Time

The projections by PRAAGE probably overestimate the increase in system unavailability since they assume a constant level of maintenance and testing. As failure rates increase with time, the plants would probably identify this and compensate with increased maintenance. However, with projections such as PRAAGE available, trending of system unavailability over the long term may be feasible as a functional indicator. By comparison to an established unavailability alert level, the plants could compensate for problems and increasing unavailability much earlier than they might otherwise. This would keep the system reliability high and prevent unacceptable increases in unavailability or core melt frequency.

6.6 Summary

The interactive computer model of the CCW system, called PRAAGE, was used to evaluate the current status of CCW systems and to project CCW system performance to an age of 40 years based on trends in the failure data. The program also calculates the importance of individual system components, based on their necessity for system operation and their likelihood of failure.

These studies showed that the typical CCW system has an unavailability (or probability of failure) in the 10^{-3} to 10^{-4} range. The values for system unavailability computed from real data on component failures are slightly but not significantly higher than CCW system unavailabilities used in contemporary PRAs. Projections out to an age of 40 years show that the unavailability of the CCW system could increase noticeably into the 10^{-2} to 10^{-3} range, if additional actions to address system failures are not taken.

System unavailability is currently dominated by failure of the Service Water System and then by failures of key CCW valves. The valves shown to be important are normally open, manual, isolation valves in the main supply header to the crucial loads. Although the IP-2 CCW system had no MOVs in such a location, a plant with MOVs located there would show a high importance for these MOVs. As the system ages, PRAAGE predicts that the CCW pumps will become more and more dominant, due to their increasing failure rate with time and to their triple redundancy.

Section 7 discusses the sensitivity studies performed on the input data and the various assumptions made.

7. SENSITIVITY STUDIES

In this section, the interactive computer model of the CCW system, PRAAGE, is used to test the effects of variations of the input data. The failure rates of the individual components are varied based on different interpretations of the data analysis to determine sensitivity of the results obtained in Section 6. Also, some of the assumptions used in the analysis are varied to determine if the results are particularly sensitive to such changes. In most cases, no significant differences in the results were seen, although in a few cases, the importance of certain components increased markedly.

7.1 Check Valves

Figure 7-1 shows the check valve failure rate data versus age and the straight line model used in the baseline PRAAGE study. The baseline model of the check valve failure was constant at 4.3×10^{-6} failures/year from age 0 to age 2 and then increased at an acceleration of $8.8 \times 10^{-12}/\mathrm{hr}^2$ (fractional increase per year of 0.018).

Another interpretation of the data is a two-line model with an aging start time in the 8 to 10 year range. This interpretation was investigated as Case 1 of the sensitivity studies. The Case 1 actual AST, initial failure rate, and slope were obtained using the method described in Section 4. This two-line model is also shown in Figure 7-1. The initial constant failure rate is 4.53×10^{-6} failures/hr, the start time is 10.4 years, and the slope has a fractional increase per year of 0.10 (units of failures/hr per year). This revised data was input to PRAAGE, keeping all other data at the baseline values and setting the service water failure rate to zero. Figure 7-2 shows the PRAAGE results.

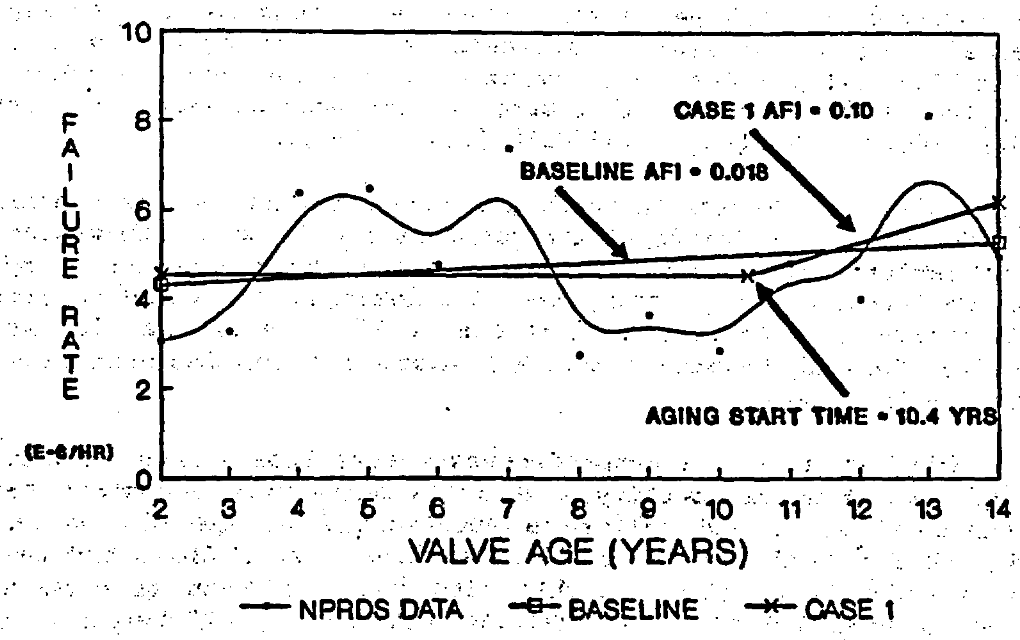

Figure 7-1 Check Valve Failure Rates Examined in Sensitivity Studies

#8 System Unavail. and Generic Inspection Aging Importance
All Numbers in Percent (%)

Class	Year1	Year2	Year5	Year10	Year20	Year40
TOT VLV	79.5263	79.5263	79.8441	84.4265	34.8928	7.9618
CC.M. VLV	52.8720	52.8720	53.0850	56.1478	22.9897	5.0111
SW.M. VLV	26.5286	26.5286	26.6406	28.1860	11.5124	2.4717
PUMPS	9.9151	9.9151	9.3523	8.9547	63.2557	91.6839
HEAT EX	8.5440	8.5440	8.9035	5.6057	1.5652	0.2794
PIPE	1.9593	1.9593	1.8479	0.9721	0.1978	0.0212
CK. VLV	0.1256	0.1256	0.1185	0.0927	0.3906	0.4790
ELECT.	0.0554	0.0554	0.0522	0.0410	0.0885	0.0538
UNAVAIL	0.0018	0.0018	0.0019	0.0037	0.0112	0.0709
SVC WTR	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000

Figure 7-2 System Unavailability and Component Inspection Importance for Check Valve Case 1 Failure Rate

Neither system unavailability nor the dominant components change from the baseline results. A noticeable difference is that at age 40, check valves become more important than heat exchangers, whereas in the base case they do not. Using the baseline check valve failure rate, which has a more conservative (lower) aging acceleration rate, does not significantly affect the results of this study.

7.2 Pumps

Figure 7-3 shows the failure data and the base case two-line model used for the CCW pumps. The aging acceleration rate of $2.9E-9/hr^2$ corresponds to a fractional increase of 0.28 per year. Pumps were found to be very important in the baseline PRAAGE analysis, particularly as they age. As a result, several different aging rates were examined in the parametric studies. The following aging rates, expressed in fractional increase per year were studied: 0, 0.10, 0.15, 0.20, 0.28 (base) and 0.40 (Figure 7-3).

Figure 7-4 shows the large effect that pump aging has on the unavailability of the CCW system. If aging of the pumps is set to zero, CCW unavailability remains essentially constant, even with the other components aging at their baseline rates. If the aging rate of the pumps is allowed to increase, CCW unavailability can increase by a factor of 20 times at age 40. As noted in Section 6.5, for the baseline case, CCW pumps are the dominant components from about year 15 onward.

As the aging rate of the pumps decreases they become less important and less dominant to CCW unavailability. When the aging rate decreases to 0.10, the importance of pumps and valves is about equal at age 40, as shown by Figure 7-5. At age 40 the total of all valves contributes 50.2% and the pumps 47.7% to unavailability. To reach this situation, the pump aging rate would have to drop to about one third of the baseline value (.10/.28=.36). Such a large drop is unlikely, hence, it is reasonable to conclude that pumps will remain the dominant component in the future as the CCW systems age.

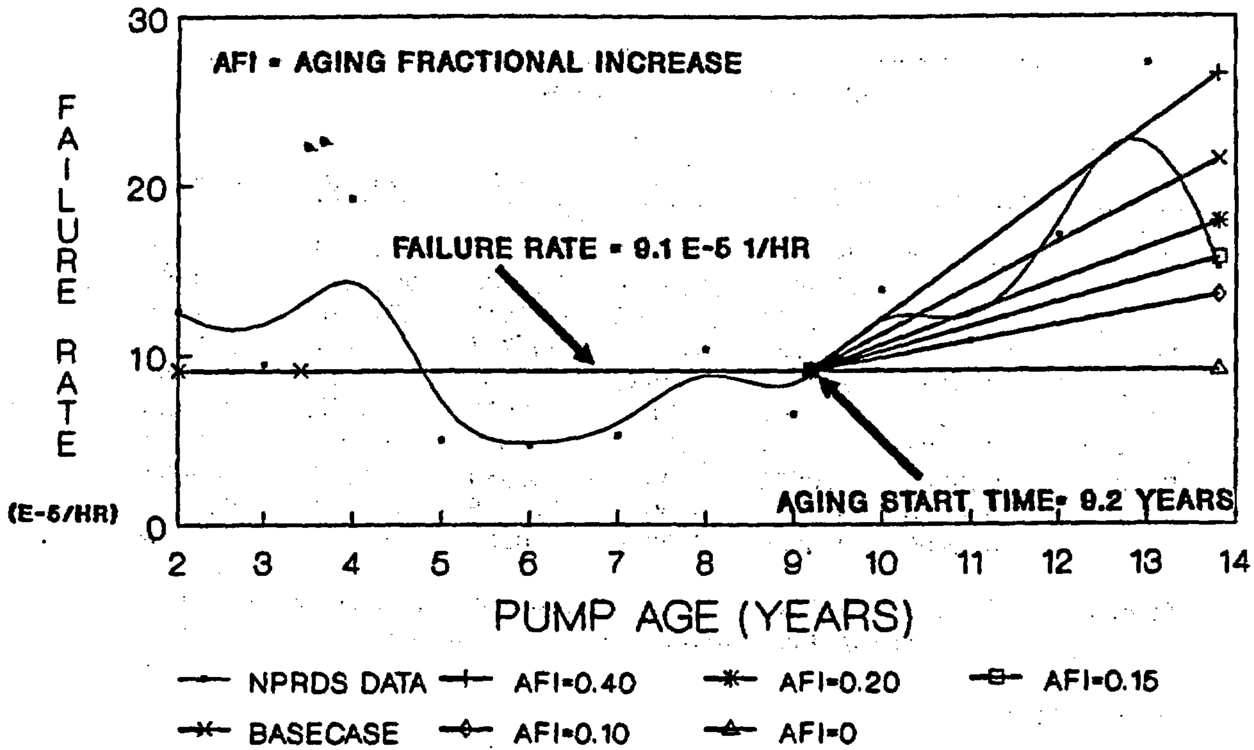

Figure 7-3 Pump Failure Rates Examined in Sensitivity Studies

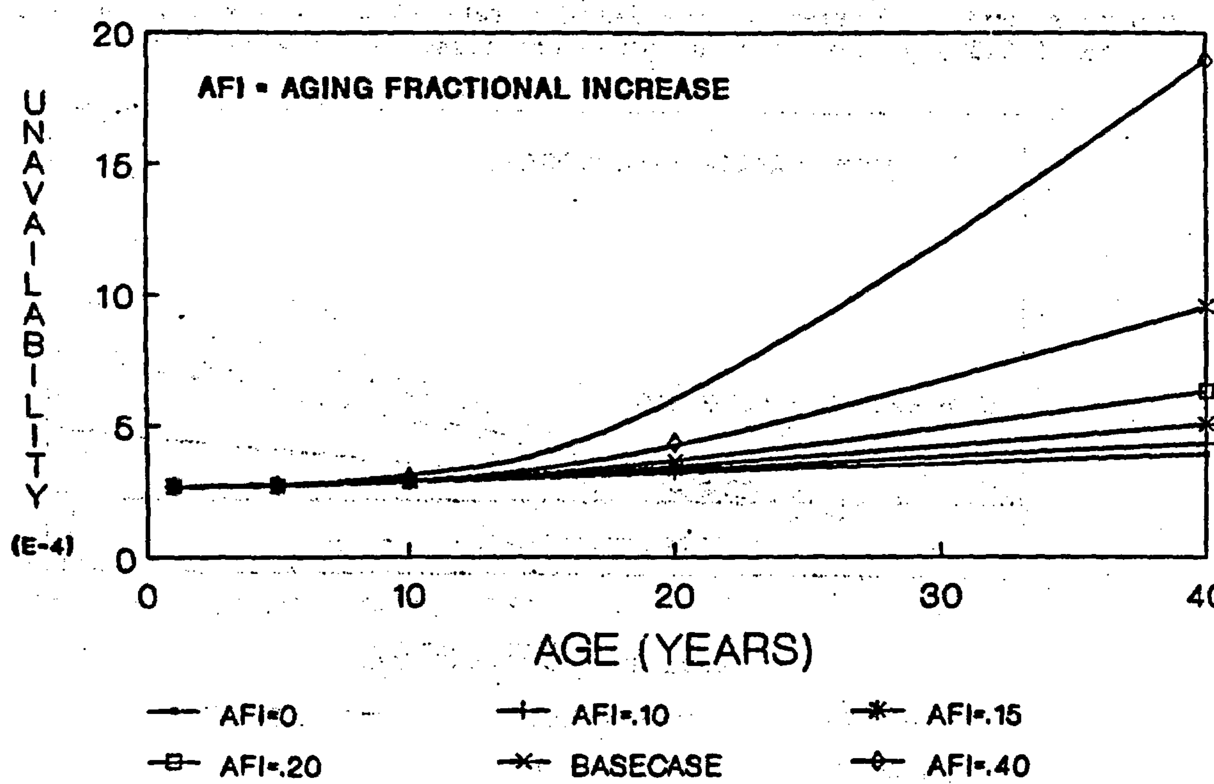

Figure 7-4 CCW System Unavailability for Various Pump Failure Rate Curves

#8 System Unavail. and Generic Inspection Aging Importance
All Numbers in Percent (%)

Class	Year1	Year2	Year5	Year10	Year20	Year40
TOT VLV	79.5348	79.5348	79.8440	86.8233	76.4689	50.2454
CC.M. VLV	52.8820	55.8820	53.0849	57.7498	50.8087	33.3052
SW.M. VLV	26.5336	26.5336	26.6405	28.9923	25.5103	16.7035
PUMPS	9.9046	9.9046	9.3524	6.3777	19.5716	47.6577
HEAT EX	8.5456	8.5456	8.9035	5.7661	3.4684	1.8882
PIPE	1.9597	1.9597	1.8479	0.9999	0.4383	0.1430
CK VLV	0.1191	0.1191	0.1185	0.0812	0.1499	0.2367
ELECT	0.0553	0.0553	0.0522	0.0330	0.0529	0.0658
UNAVAIL	0.0018	0.0018	0.0019	0.0036	0.0076	0.0183
SVC WTR	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000

Figure 7-5 CCW Unavailability and Component Importance for Pump Aging Fractional Increase = 0.10

7.3 Manual Valves

Figure 7-6 shows the data on manual valve failure and the base case two-line model used for the PRAAGE analysis. The initial failure rate is 2.2E-7 failures/hour until 4.7 years. The failure rate then increases at a fractional rate of 0.215 per year or an aging acceleration of 5.4E-12/hr^2.

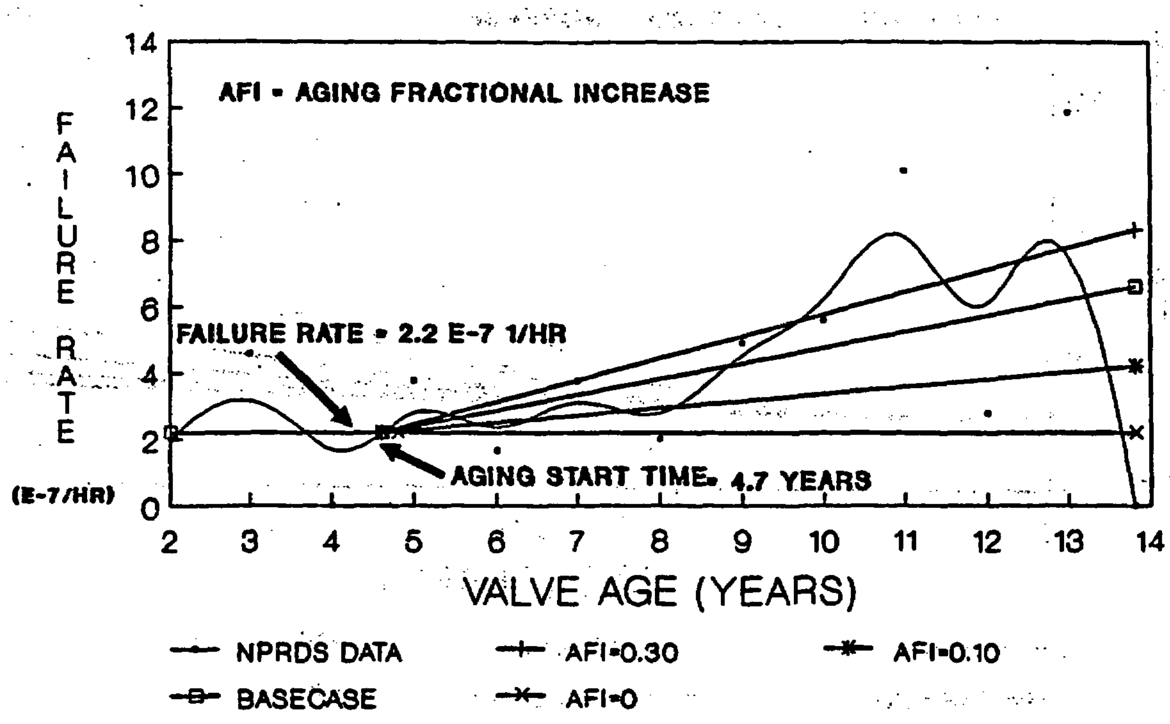

Figure 7-6 Manual Valve Failure Rates Examined in Sensitivity Studies

In a manner similar to that used in the parametric study of pump failure rate, the manual valve aging fractional increase per year was varied to study the effect of different aging rates. Aging rates of 0, 0.10, 0.21 (base case) and 0.30 were selected (Figure 7-6).

Figure 7-7 shows the relative or percent importance of manual valves in the CCW system for the four cases studied from zero to 40 years of age. All follow the base case of a high, early importance and then decreasing importance as the pumps take over at about 15 years. The small increase in the importance of the valves until shortly after age 10 is due to the fact that the aging start time for valves is 4.7 years and that for pumps is 9.2 years. These results show that, CCW system unavailability and component importance are relatively insensitive to the aging acceleration rate of manual valves.

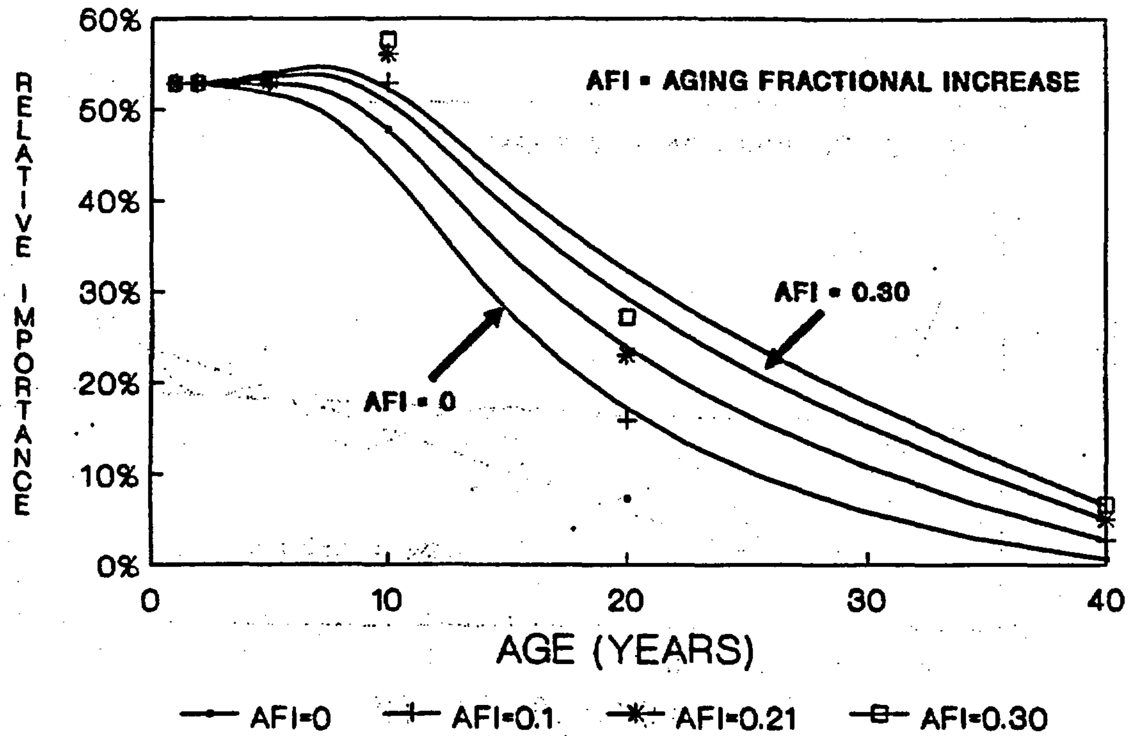

Figure 7-7 CCW System Manual Valve Importance for Various Aging Acceleration Rates

7.4 Heat Exchangers (HX)

Figure 7-8 shows the failure data on heat exchangers and the baseline model used for the PRAAGE analysis. The baseline model was $3.8x10^{-5}$ failures/hour up to year 2, and then a fractional increase of 0.017 per year (which corresponds to $7.37x10^{-11}$ failures/hr^2). Two different cases besides the base case were selected for sensitivity analysis. Case 1 is also shown on Figure 7-8 and consists of a two-line model with a break point at age 5 and then a steeper slope than the base case. The Case 1 fractional increase per year is 0.381.

The Case 1 results, given in Figure 7-9, show an increase in HX importance over the base case as the plant ages, from an initial 0.2% to 2.2% at age 40. This value is still quite small compared to the importance of the pumps at age 40, but is now approximately the same as the second most important component; namely, CCW-manual valves.

A second case was also analyzed for CCW-HXs. Case 2 utilized all of the failure data for HX's, even beyond the fourteen-year mark. After 15 years the data is sparse and not statistically reliable; however, for sensitivity analysis it can be informative. Figure 7-10 shows this data and illustrates a potentially large increase in HX failures after year 16. This is physically possible, since HXs tend to operate for a long time with no leaks until corrosion and erosion have reached a stage when leaks are beginning, and then many leaks may appear in quick succession.

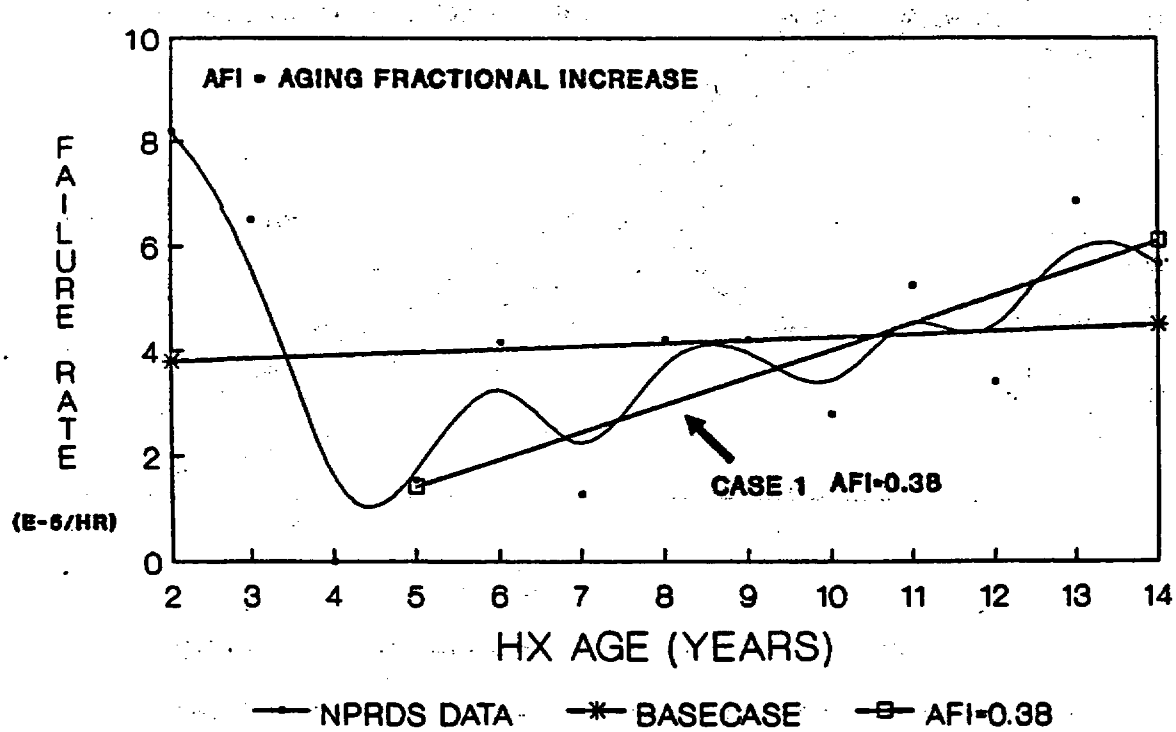

Figure 7-8 Heat Exchanger Failure Rates Examined in Sensitivity Studies

#8 System Unavail. and Generic Inspection Aging Importance
All Numbers in Percent (%)

Class	Year1	Year2	Year5	Year10	Year20	Year40
SVC WTR	93.1062	93.1062	92.7510	86.1871	54.9329	11.7225
TOT VLV	5.9160	5.9160	6.2735	11.7399	15.1911	6.7400
CC.M. VLV	3.9365	3.9365	4.1475	7.8075	10.0359	4.3828
SW.M. VLV	1.9706	1.9706	2.0897	3.9183	5.0465	2.1901
PUMPS	0.7382	0.7382	0.7364	1.2475	27.4399	79.2373
PIPE	0.1460	0.1460	0.1455	0.1352	0.0862	0.0184
HEAT EX	0.0894	0.0894	0.0895	0.6846	2.3115	2.2354
UNAVAIL	0.0267	0.0267	0.0268	0.0287	0.0366	0.0976
CK VLV	0.0089	0.0089	0.0093	0.0140	0.1087	0.1671
ELECT	0.0041	0.0041	0.0041	0.0057	0.0384	0.0465

Figure 7-9 HX Parametric Study Results - Case 1

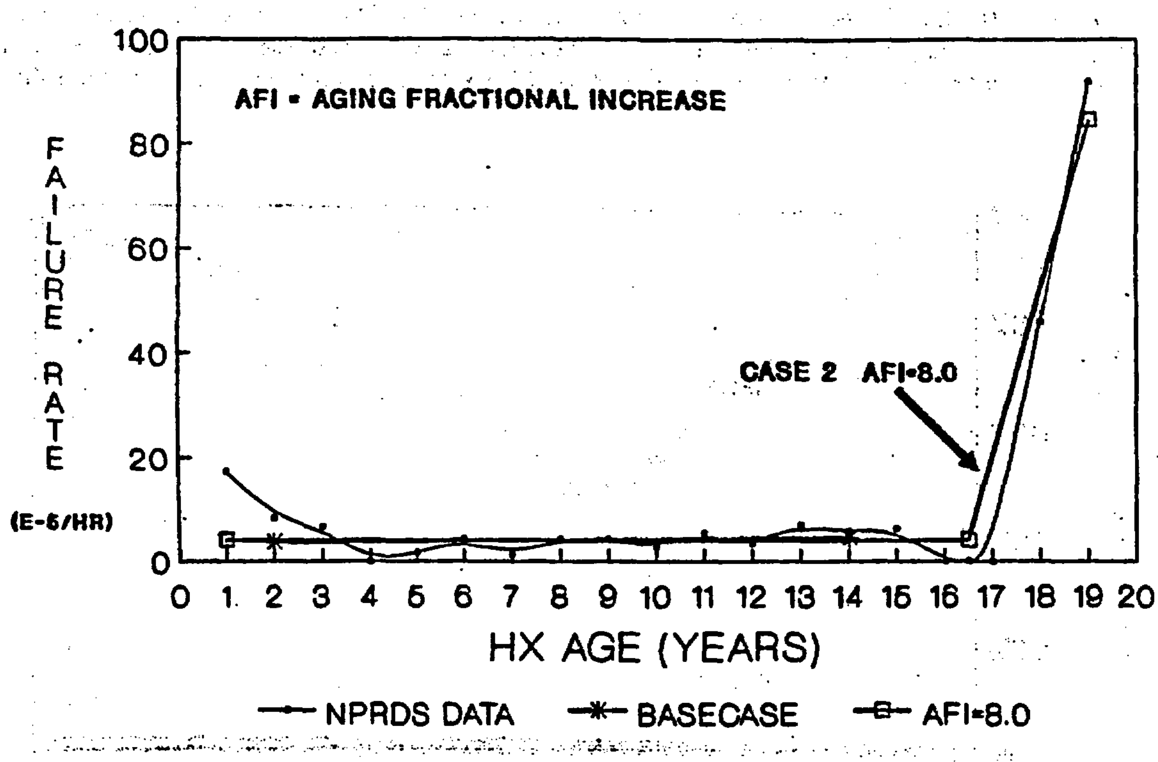

Figure 7-10 Heat Exchanger Case 2 Failure Rate

Case 2 uses a two-line model with an initial failure rate of 4.01×10^{-5} failures/hour, a break point of 16.5 years and after that a fractional increase per year of 8.035. A failure rate of 7.6×10^{-3} failures/hr would result at age 40 if the trend continued; this is abnormally high, but is useful as a bounding example. Figure 7-11 shows the results of the PRAAGE analysis for Case 2.

#8 System Unavail. and Generic Inspection Aging Importance
All Numbers in Percent (%)

Class	Year1	Year2	Year5	Year10	Year20	Year40
SVC WTR	92.5223	92.5223	92.1701	86.1952	12.5997	0.4312
TOT VLV	5.8912	5.8912	6.2473	11.7407	3.7742	0.3748
CC.M. VLV	3.9168	3.9168	4.1535	7.8082	2.4178	0.2119
SW.M. VLV	1.9656	1.9656	2.0845	3.9185	1.3314	0.1567
PUMPS	0.7335	0.7335	0.7318	1.2476	6.2938	2.9146
HEAT EX	0.7037	0.7037	0.7022	0.6755	77.3038	96.2771
PIPE	0.1451	0.1451	0.1446	0.1352	0.0198	0.0007
UNAVAIL	0.0268	0.0268	0.0269	0.0287	0.1130	2.8867
CK. VLV	0.0088	0.0088	0.0093	0.0140	0.0249	0.0061
ELECT	0.0041	0.0041	0.0041	0.0057	0.0088	0.0017

Figure 7-11 PRAAGE Results for Heat Exchanger Case 2 Study

By age 20 HXs have increased dramatically in importance to 77%. At age 40, they constitute 96% of the unavailability of the system. Figure 7-12 summarizes the three HXs cases examined and illustrates the dominance of HXs for Case 2.

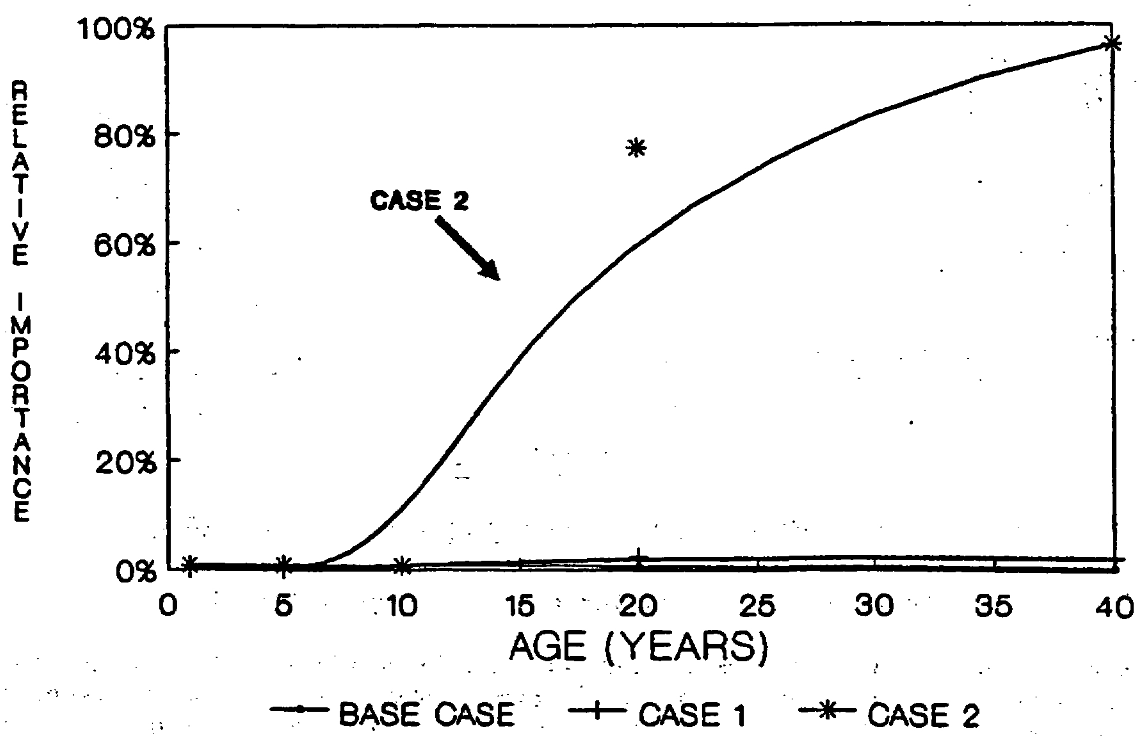

Figure 7-12 Heat Exchanger Importance Versus Time

7.5 Piping

Two parametric cases were analyzed for the failure data on piping in addition to the base case. The base case used the value from the IP-2 PRA of 8×10^{-10} failures per pipe section per hour. The other two cases used data reported to NPRDS pipe leakage. Both cases assumed that a reported leak constituted a failure, which is a conservative assumption and gave the piping a higher importance than it actually has. All the PRAAGE analyses divided the CCW system into 18 pipe sections and treated each equally, as did the IP-2 PRA. Case 1 averaged all reported instances of CCW pipe leaks and obtained a constant failure rate of 2.69×10^{-7} failures per pipe section per operating hour. Case 2 was based on analyzing the data in 5-year increments (as described in Section 5) and obtained an increasing failure rate with a slope of 5.6×10^{-8} failures/hr-yr. All three failure rates are shown in Figure 7-13. Case 1 and Case 2 cross at about age 7.

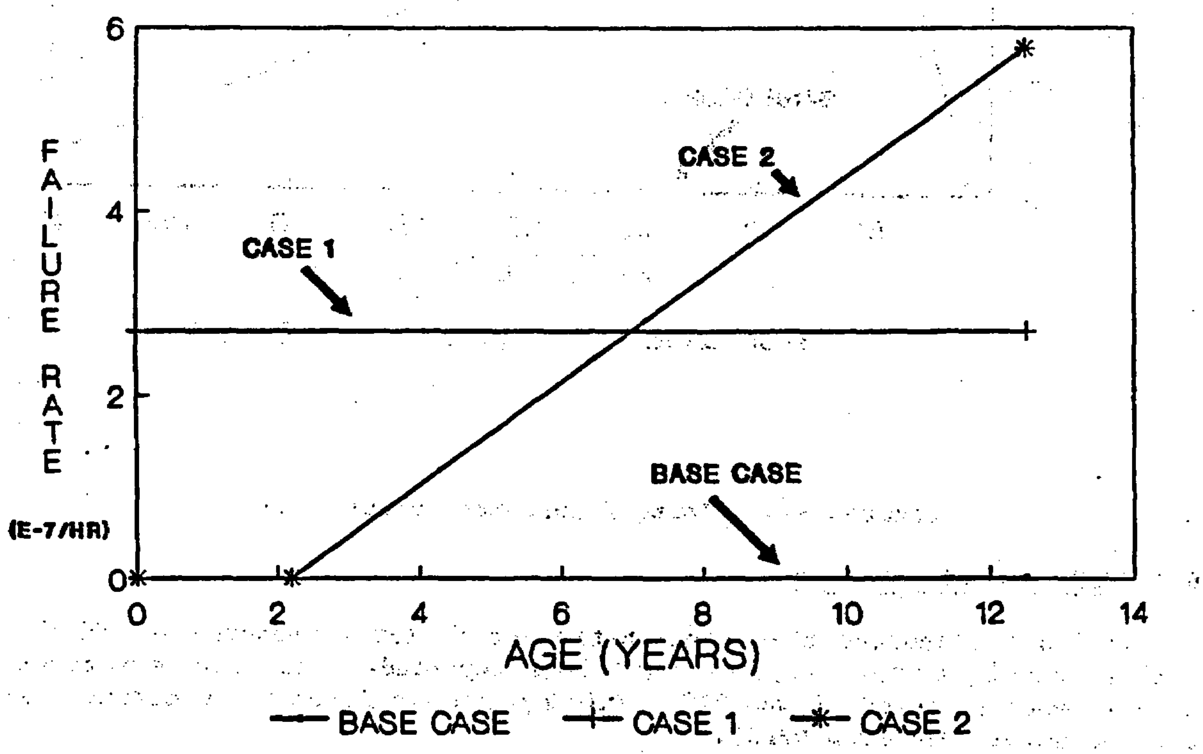

Figure 7-13 Piping Failure Rates Examined in Sensitivity Studies

When these three cases are analyzed with PRAAGE, notably different results are obtained. These are presented in Figure 7-14. For the base case, the importance of piping is less than 1% for all 40 years. For case 2, pipe importance starts at the base value and increases to around age 10, then it decreases as the importance of the pumps takes over. For case 1, pipe importance starts high then decreases and crosses the curve for Case 2 at about age 7, the same time at which the failure rate curves for pipes cross. The very high importance of piping shown in this parametric study for case 1 and 2 implies that the assumption that leaks constitute a failure is probably unreasonable. However, it shows a bounding case, such that if pipe leaks are developing and not addressed, they can significantly affect the availability of the CCW system.

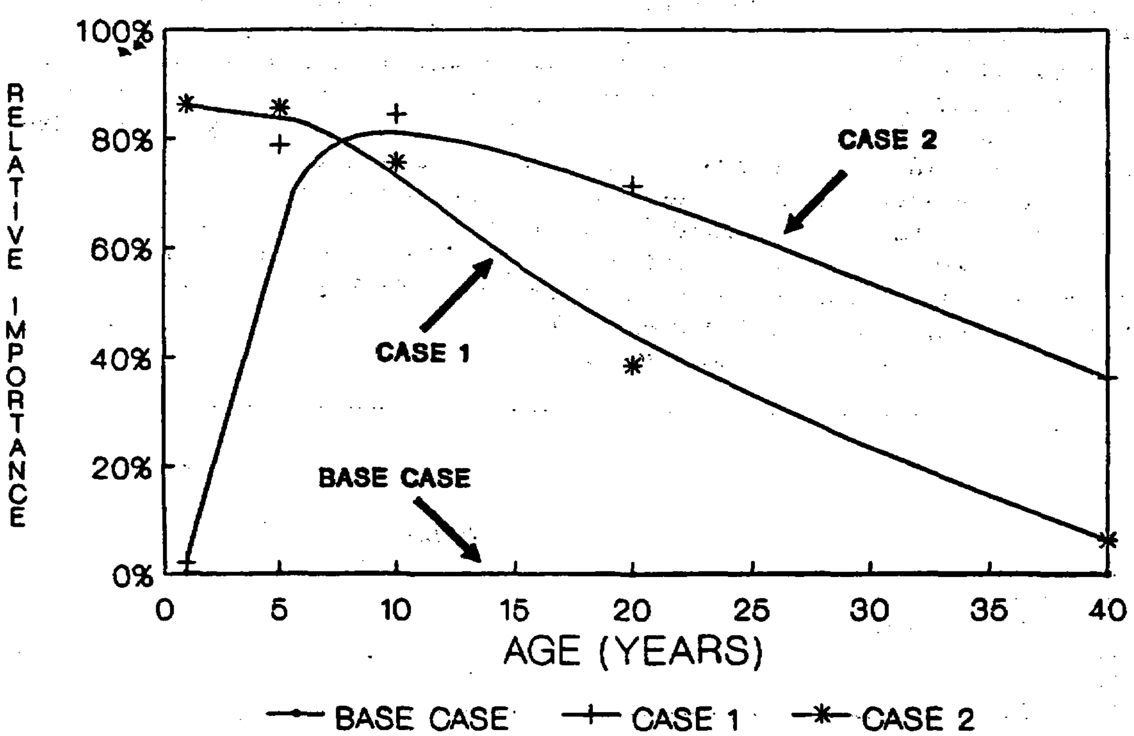

Figure 7-14 Piping Importance Versus Time

7.6 NPRDS Reporting Factor

Section 4 discusses the determination of failures reported by NPRDS and compares actual failures at IP-2 with those reported. The average percent of failures reported to NPRDS was 31%. This percentage was used to correct failure rates determined from NPRDS.

An attempt was made to establish a reporting factor dependent on calendar year, since reporting increased after 1975. However, there was insufficient data to do this, therefore, the reporting correction factors are averaged over all years.

To determine the sensitivity of the results to this factor, calculations were performed using the following NPRDS reporting percentages: 100%, 50%, 31% (base case), 20%, 10% and a variable percentage. For each one, except the variable case, all component failure rates were adjusted by the same value. For the variable case, pumps and valves were adjusted by the individual amounts indicated in the IP-2 review, 57% and 15%, respectively. Figure 7-15 illustrates the PRAAGE results for the various cases analyzed.

NPRDS Reporting Percent	CCW Unavailability		Dominant Component		% Contribution	
	Year 1	Year 40	Year 1	Year 40	Year 1	Year 40
100	6.0E-6	3.6E-4	CCW-V	Pumps	50	95
50	1.1E-5	5.0E-4	CCW-V	Pumps	52	93
31	1.8E-5	7.1E-4	CCW-V	Pumps	53	92
20	2.8E-5	1.0E-3	CCW-V	Pumps	51	91
10	6.1E-5	2.4E-3	CCW-V	Pumps	44	92
Variable	3.4E-5	6.8E-4	CCW-V	Pumps	60	80

Note: CCW-V = CCW Manual Valves

Figure 7-15 NPRDS Reporting Factor Sensitivity Study Results

Figure 7-15 shows that as the percent of failures reported decreases, the component failure rates, and the calculated CCW unavailability increase. Although the relationship is not strictly linear, in this example, various competing non-linear effects have balanced out and the CCW unavailability has increased by a factor of 10 as the NPRDS reporting percent varied from 100% to 10%. An important point to note is that the dominant component to unavailability did not change as reporting percent varied. In fact, even the percent contribution of the dominant component varied very little. The only noticeable variation was when the variable (between components) reporting percentages were applied. As a separate comparison, NPRDS reporting factors for inverters were analyzed (as determined during the aging of battery chargers and inverters study [1]). An upper bound on the reporting factor for inverters was determined to be 61%.

In conclusion, although the 31% NPRDS reporting correction factor is not firm, it is the best estimate available. The results are not sensitive to variations in the reporting factor if an average value is used. If reporting factors for actual components deviate significantly from the average, however, this could affect results of component importances as seen in this study.

7.7 Summary

The sensitivity studies showed that with a few exceptions the baseline results described in Section 6 are generally applicable and not excessively sensitive to reasonable variations in input data and assumptions. Specifically, the results are not sensitive to changes in the data on failure rate for valves (check valves, manual valves, or MOVs) or to changes in the assumed NPRDS reporting correction factors. The dominance of pumps in later years is unaffected by reasonable changes in their failure rates, but the system unavailability is sensitive to increases in the failure rate of pumps above those determined.

The heat exchanger (HX) data indicates sharply increasing failure rates in the 15-20 year period. If this is true, the results are sensitive to this and would result in a much higher dependence of system unavailability on HXs.

The data on failure of CCW pipes is quite sparse. The analyses performed indicate that the results are somewhat sensitive to changes in pipe failure rates. For more detail the reader should refer to the pertinent subsection of Section 7. The results of the sensitivity studies are summarized in Table 7-1.

Table 7-1 Summary of Sensitivity Study Results

	Item/Component	Parameter Varied	Sensitivity
1.	CCW Unavailability	Pump AFI	High
		Check Valve AFI	Low
		Check Valve AST	Low
		Manual Valve AFI	Low
		NPRDS Reporting Factor	Moderate
2.	Check Valve Importance	Check Valve AFI	Low
		Check Valve AST	Low
3.	Manual Valve Importance	Manual Valve AFI	Low
4.	Heat Exchanger Importance	HX AFI	High
5.	Piping Importance	Piping AFI	High
6.	General Component Importance	NPRDS Reporting Factor	Low

AFI = Aging Fractional Increase

AST = Aging Start Time

8. RESULTS

8.1 Discussion

The deterministic and probabilistic work showed that aging degradation is a concern for CCW systems and can adversely affect the performance and availability of the system. An analysis of past operating experience indicates that the dominant cause of failure is "normal service", while the predominant mechanism of failure was "wear." These findings support the conclusion that aging contributes to a significant portion of CCW system failures. Monitoring methods must, therefore, include good functional indicators which will detect aging effects while the system operates normally. It would then be possible to mitigate aging degradation.

The data showed that there were numerous failure modes with "leakage" the most common. The components most frequently found to be failed were valves and pumps. For valves, there was internal leakage through valve seats and external leakage from seals. Instrumentation/controls and heat exchangers also have a significant number of failures.

To quantify the effects of aging, time-dependent failure rates were calculated. These failure rates showed a trend toward increasing with age for most of the components examined. It should be noted that these failure rates include the effects of current testing and maintenance practices, which indicates that improvements in both these areas may be required to effectively mitigate aging effects.

Results from the probabilistic work indicate that if interaction between other plant systems is considered, unavailability of the CCW system is dominated by loss of service water to the CCW HX's. In the CCW system itself, baseline results (without aging effects) identified valves, followed by pumps, as the most important components contributing to system unavailability. These results are consistent with the results from the deterministic work which also identified valves and pumps as key components contributing to CCW system failures.

Incorporation of time-dependent failure rates into the PRA calculations provided two significant results: 1) system unavailability increases with age, and 2) component importances change with time. The increase in unavailability suggests that improvements to monitoring methods and maintenance practices may be required to prevent the performance of the system from reaching an unacceptable low level as plants age. The reliability of the CCW system is one area which should be carefully evaluated in relation to extending the life of the plant.

The change in importance of components with age is significant since it identifies an area where age may need to be considered in developing modifications or improvements to plant surveillance and monitoring. For the system analyzed in this study, the manual series header valves were most important in the early years of plant life. When the effects of aging were accounted for, pumps became the most important component after approximately 20 years, and by age 40 they dominated. Based on these findings, a plant using this design should stress surveillance and monitoring of key valves during early years of plant life. During later years, however, more attention should be focussed on the pumps.

The PRA type analyses performed on the CCW system for this study were at the system and component level and focussed on one particular plant, examining in detail how the CCW system itself could fail. It was seen that specific system designs must be considered in evaluating component importances. For example, series header manual supply valves to key loads were dominant to system unavailability early in plant life for the system studied. At other plants, MOV's may be used instead of manual valves and, hence, MOV's would have high importance due to their location. CCW system failure was not projected to the level of the plant nor to core melt. Other work, for example NUREG/CR-4643, "Evaluation of Core Damage Sequences Initiated by Loss of Reactor Coolant Pump Seal Cooling," examined the effect of loss of CCW on core melt frequency and has found that complete failure of the CCW system can have severe consequences.

The data suggested that piping and heat exchangers can become very dominant in later years if failure rates increase at the higher rates indicated. This is due to their predominant failure mechanisms (corrosion and erosion) which are relatively slow processes. Increased surveillance should be considered for these components in later years of plant life. These components should also be addressed as concerns for plant life extension.

Table 8-1 summarizes the variation in component importance with age from both the deterministic and probabilistic approaches. The importance rankings for the deterministic work were determined from a review of the relative number of failures reported for each component, along with their calculated age-dependent failure rate. Rankings for the probabilistic work were obtained from PRAAGE results using time-dependent failure rates.

The results presented in Table 8-1 show that both approaches led to the same general conclusion, that aging degradation occurs in CCW systems and can increase system unavailability and cause shifts in component importance with age. Both approaches identified valves and pumps as the dominant components involved in CCW failures. However, several of the major components may become important in later years and should be considered in assessments of plant life extension. Further work is required to develop the appropriate techniques for life extension analyses.

This study showed that good functional indicators are required to mitigate the effects of aging. Since numerous aging mechanisms are present, surveillance and monitoring programs must be diverse. Figures 8-1 through 8-3 identify the performance hazards, aging effects and potential functional indicators for the major CCW system components. This information can be used in assessing current practices.

Table 8-1: Comparison of Importance Rankings

Component	Failure Mechanisms	Importance Ranking: Determin.	Importance Ranking: Prob.	Life Ext Concern	Remarks
Valves	Wear, Foreign Material, Vibration	High – All Ages	High < 15 yr. Medium > 15 yr.	Yes	• Valves in critical locations are most important • Other components may become more important than valves in later years due to increasing failure rates
Pumps	Wear Vibration	Medium < 10 yrs High > 10 yrs	Medium < 15 yrs High > 15 yrs	Yes	• Results show pumps have potential to become dominant component in later years
Heat Exch.	Corrosion, Erosion	Medium < 15 yrs High > 15 yrs	Low < 20 yrs High > 20 yrs	Yes	• Data indicates HX's have potential for large increase in failure rate in later years
Piping	Corrosion, Erosion	Low < 20 yrs High > 20 yrs	Low < 20 yrs High > 20 yrs	Yes	• Data indicates piping has potential for large increase in failure rate in later years

The functional indicators presented in Figures 8-1 through 8-3 are recommended as potentially viable methods for monitoring and detecting aging degradation. Some of these indicators may already be commonly used while others may require verification of their effectiveness. These functional indicators can be used to modify and improve current monitoring and surveillance methods. However, routine preventative maintenance, which is currently performed, should not be discontinued.

The functional indicators (FI's) discussed are at the compoent level. The logic associated with this approach is that improved component reliability results in improved system reliability. However, there are also a few select FI's which can be classified as system level: 1) surge tank level, 2) pump discharge flow and pressure, 3) heat exchanger outlet temperature, and 4) system unavailability. The first system level FI, surge tank level, addresses one important failure mode, namely leakage. The second and third FI's monitor the capability of the system to achieve its primary design functions, that is to provide sufficient coolant at an acceptable temperature to all of its loads. The last FI, system unavailability, is an integrated assessment of system performance, which can be analyzed to identify weak links. System unavailability can be evaluated for its suitability as a system level FI in the next phase of the CCW work. Results from the Performance Indicator Program at BNL also can be used to evaluate the effectiveness of unavailability as a system level FI.

8.2 Utilization of Research Results

The value of a research program lies in the degree to which its products are utilized 1) as input to other programs, and 2) as technical information to improve operations and maintenance. Table 8-2 identifies those areas to which the systems aging study of component cooling water will provide useful input.

8.3 Future Work

Future work to be performed in Phase I and Phase II of the CCW system study will include the evaluation of current monitoring methods, regulations, testing and maintenance programs. Their effectiveness in mitigating aging effects will be determined and recommendations will be made for improvements. The potential functional indicators identified in this report will also be examined in more detail.

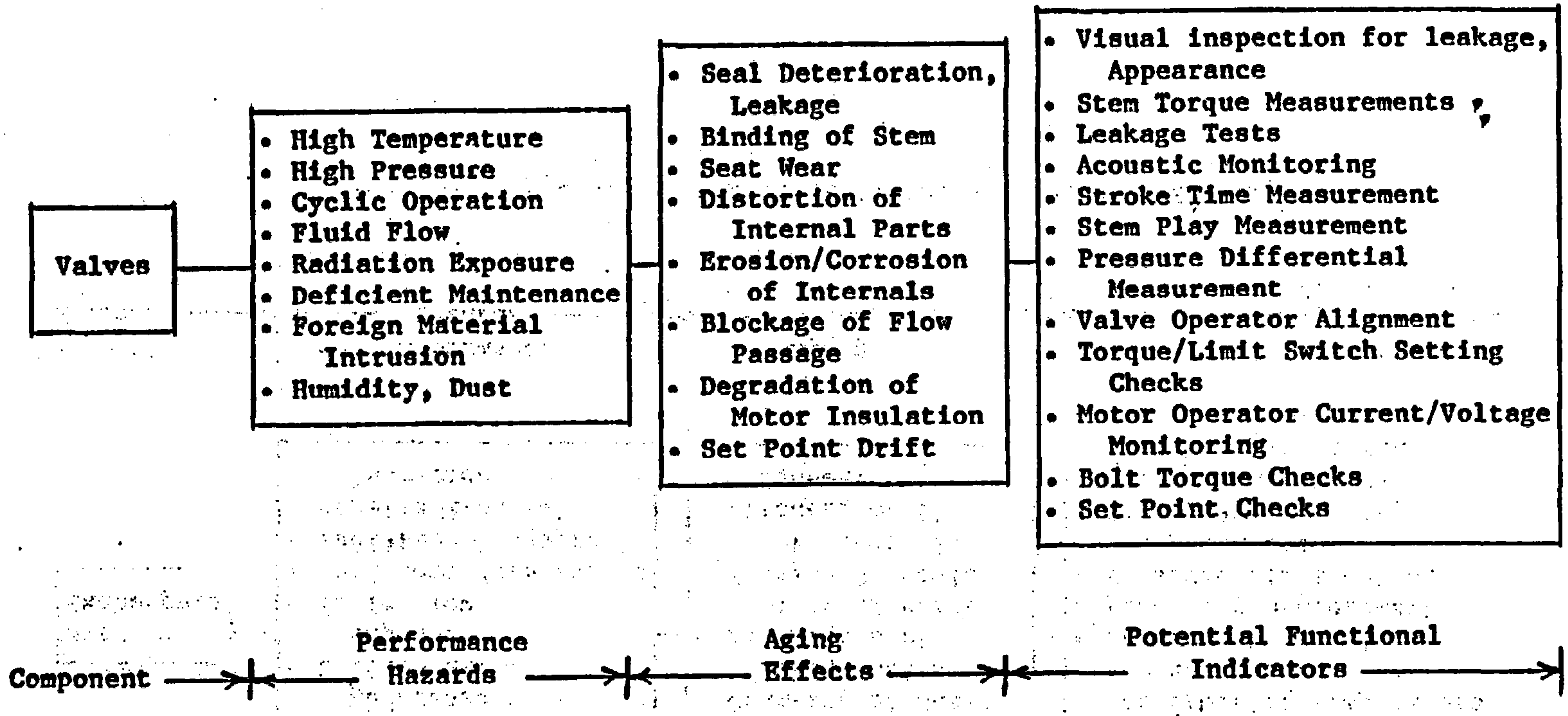

Figure 8-1 Potential Functional Indicators for Valves

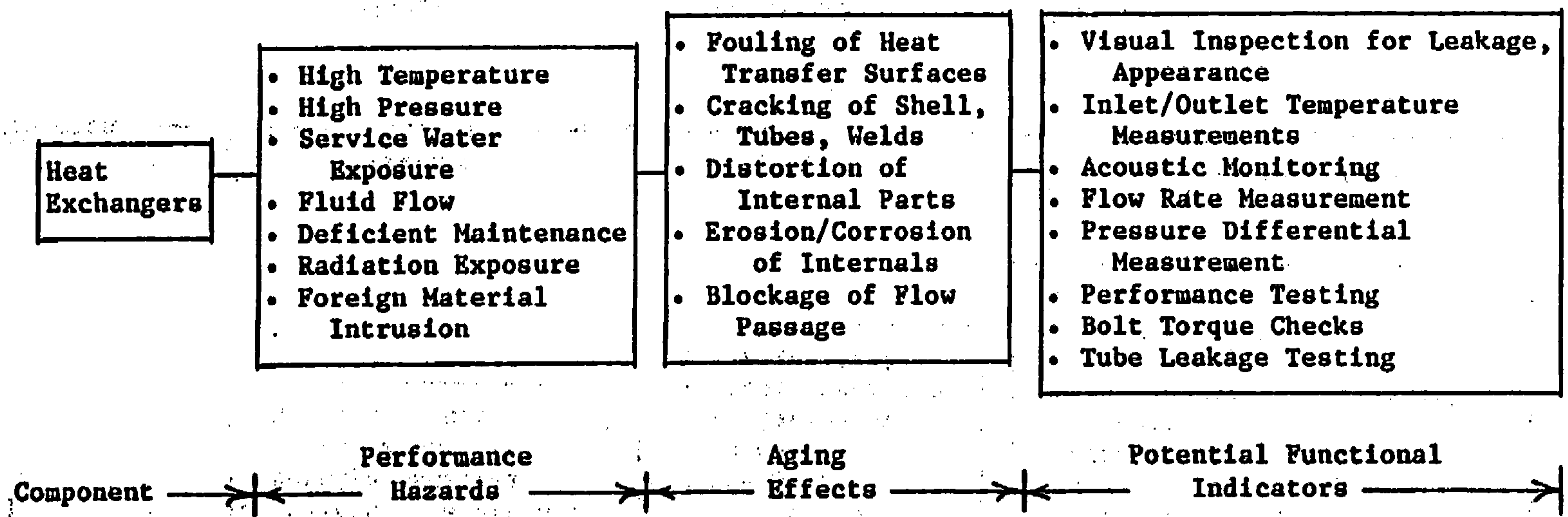

Figure 8-2 Potential Functional Indicators for Heat Exchangers

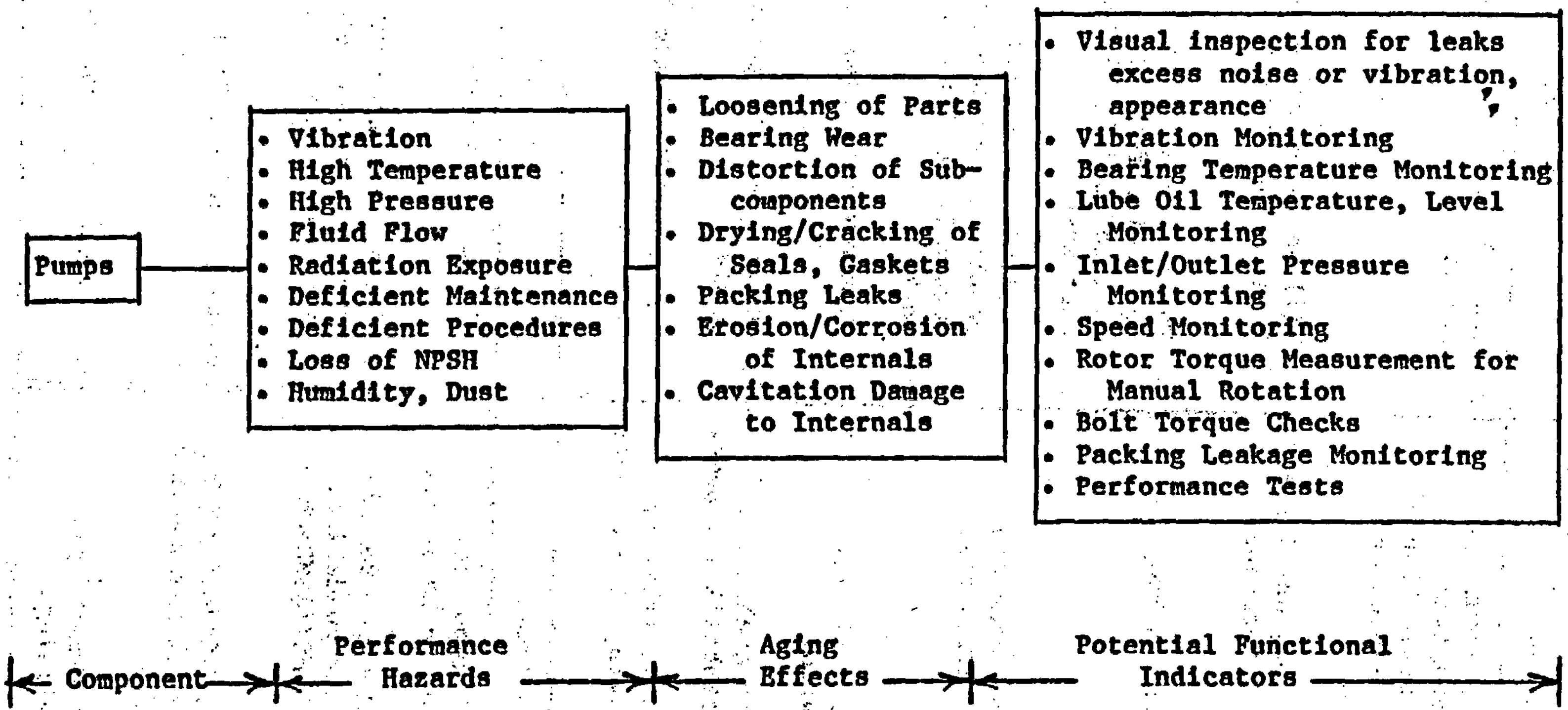

Figure 8-3 Potential Functional Indicators for Pumps

Table 8-2 Utilization of CCW Research Results

General Areas	Remarks
1. Identification of predominant failure modes, mechanisms and causes for use in evaluation of inspection, surveillance, and monitoring methods.	1. Phase I output included in this report.
2. Support NRC in review, development and inspection of maintenance programs.	2. Will be supported by future Phase I and Phase II work.
3. Support NRC Inspection Program	3. Phase I results herein herein are useful for inspection and will be used in FY 1988 inspection task.
4. Identify system and component level functional indicators.	4. Preliminary support with Phase I results; finalized results in future Phase I and Phase II work.
5. Provide technical basis for plant life extension.	5. Complete with Phase I results.
6. Provide support in evaluation of storage and mothballing issues.	6. Phase I results provide technical basis.
7. Determine risk/unavailability associated with aging of components and systems.	7. Complete with Phase I results herein.
Specific Areas	**Remarks**
8. Provide technical input for resolution of Generic Issue 65.	8. Complete with Phase I results.
9. Provide technical input to ASME performance testing guidelines for CCW systems (ANSI/ASME-OM2-1982).	9. Phase I results provided to ASME. Additional interface will take place in phases I and II.
10. Provide input to NRC Reliability Program (Operational Safety Reliability Research).	10. Complete with Phase I results.
11. Provide input to NUREG-1150 Zion Risk Rebaselining.	11. Complete with Phase I results.

9. CONCLUSIONS AND RECOMMENDATIONS

With the findings presented in this report, the first step in understanding and managing aging in CCW systems is complete. The aging phenomenon has been characterized, and a sound technical basis for future work has been established. In addition, several significant conclusions which could influence future NPAR work should be noted.

Conclusions specific to CCW systems include the following:

- The majority of CCW failures are not detected until an operational abnormality occurs or until a test is performed. In addition, only a small percentage are detected by alarms. The CCW operating parameters monitored and alarmed should, therefore, be thoroughly reviewed to ensure that they represent the best choices as indicators of incipient failures. More effective indicators may be required.

- In the CCW data analyzed for this study, piping and heat exchangers were found to have a potentially significant increase in failures during later years. If this occurs, these passive components could become the dominant contributors to system unavailability. Passive components should, therefore, be closely monitored in later years and should not be dismissed as unimportant due to their relatively low failure rates during early age.

Conclusions which are generic in nature but should be considered for all future NPAR work include the following:

- The systems level approach which uses probabilistic as well as deterministic techniques is an effective method of performing systems level aging analyses. It provides a comprehensive means of investigating aging effects and should be used for future system level studies.

- Based on the preliminary findings of this study, current PRAs could be underpredicting long-term plant risk. If PRAs are to be used for plant life extension decisions, the time-dependent effects of aging on component failure rate, component importance and system unavailability should be further examined and addressed.

- Existing national databases are useful for performing aging analyses if appropriate review techniques are employed. However, the database information is difficult to obtain and the required review process is extremely labor intensive. The databases should, therefore, be reviewed to determine if modifications are possible to provide a more accessible and efficient means of analyzing the failure data.

- This study has identified aging trends in component failure rates, component importances and system unavailability that could have adverse impacts on plant safety in later years. Future operating experience should be monitored and periodically checked against these results to ensure that these detrimental trends are not occurring, or that timely preventive actions are taken. The results from this and similar NPAR studies should, therefore, be conveyed to an appropriate group of data analysis personnel to periodically update results using current plant operating data, and check them against predicted trends.

- This study has shown that the potential exists for component failure rates to increase with time and for their relative importance to system unavailability to change. Current test and maintenance activities may not be effective in controlling these trends. Appropriate measures should, therefore, be taken to ensure that test and maintenance actions adequately address the time-dependent effects of aging.

- The more redundant a system is, the faster its relative aging rate, because aging is a common cause effect. Further study of this effect with various quantitative examples is recommended for future system level analyses.

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APPENDIX A

DESIGN REVIEW

CONTENTS

FIGURES

TABLES

A.1 APPROACH

Since the Component Cooling Water (CCW) system design varies between plants and since system design is very important to overall plant performance, it was necessary to perform a detailed design review of each plant's CCW system. This review was important to the NPAR systems study since it provided an understanding of CCW system characteristics, ensured applicability of the selected reference plant (Indian Point-2), aided in the analysis of system failures, and provided the population data necessary for normalizing the failure data. It also provided valuable design insights presented herein.The review was performed by first establishing the basic information desired and then developing a form to be completed for each plant reviewed. The reviews were generally made using the plant's Final Safety Analysis Report (FSAR). A completed form is shown in Figure A-1 for D.C. Cook Unit 1, which is representative of the most prevalent system design. This form was completed for each PWR unit in the United States. Figure A-2 is the data for Indian Point-2, our reference plant. For purposes of comparison, the Reactor Building Closed Cooling Water (RBCCW) system at two BWRs also was reviewed.

A.2 SUMMARY OF DESIGN REVIEW

When all the reviews of CCW systems were completed, they were summarized for each of the three NSSS vendors: Westinghouse, Combustion Engineering (CE), and Babcock & Wilcox (B&W). Then, a final summary of all PWRs was made.Table A-1 provides the summary data for the design reviews of Westinghouse plants, Table A-2 has the summary for CE plants and Table A-3 for B&W plants. Table A-4 lists the abbreviations used in these tables. These tables show the number of each of the major components and the major loads for each plant. Unique features are identified in the Comments column. One should note that there are several plants using shared or cross connected systems and also plants with more than one CCW system. More information on these features is provided in Section A.3.1.

Figure A-1 CCW System Summary

Plant Name: Cook 1 Info Source: FSAR

Pumps: Number (%): 3 (100%) HP: 500HP

Flow Rate: 9000 gpm Head: 190'

Elec Source: Norm or Emerg.

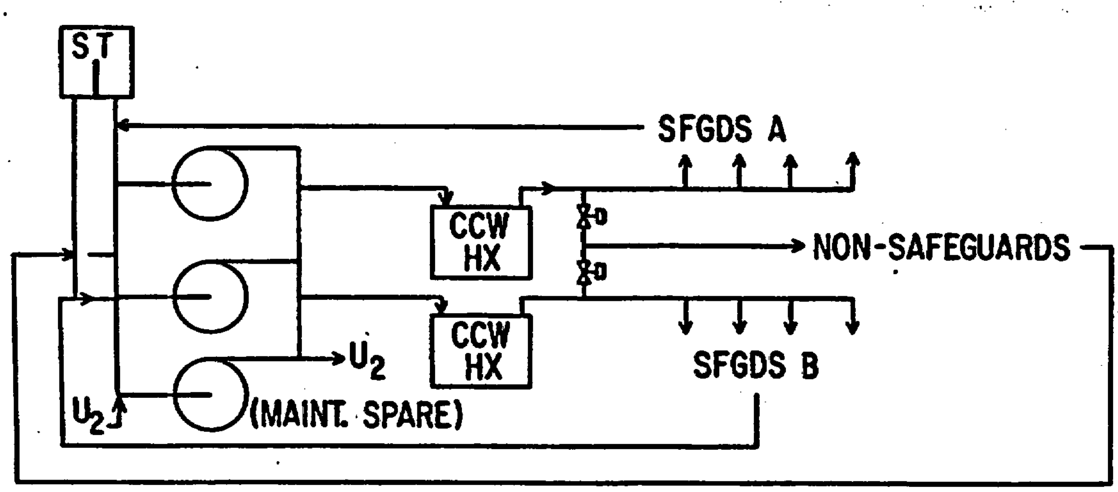
HXs: Number(%): 2 (100%) Surge Tks: 1

Cooling by: Essential Service Water System Designer: American Elec. Power Service Corp

Loads: RHR-HX, Chg-P, SI-P, RHR-P, CS-P, Misc. PDP, SW-HX, LD-HX, XLDHX, RCP m & tb

Notes: One pump for Unit 1 is a maint. spare. Units are cross-connected at pump suction and discharge only.

Instrumentation:	Indication	Alarms	Interlocks
	Rad Mon, T P, Flow, RCP. F&P	Hitemp HX out Flow, ST level	S.T. vent closes on hi-rad, CIV auto isol.



Sketch

Figure A-2 CCW System Summary

Plant Name: Indian Point-2 Info Source: FSAR, PRA, SD

Pumps: Number (%): 3 (100%) HP: 250

Flow Rate: 3600 gpm Head: 220'

Elec Source: 480V buses 2A, 3A, 5A

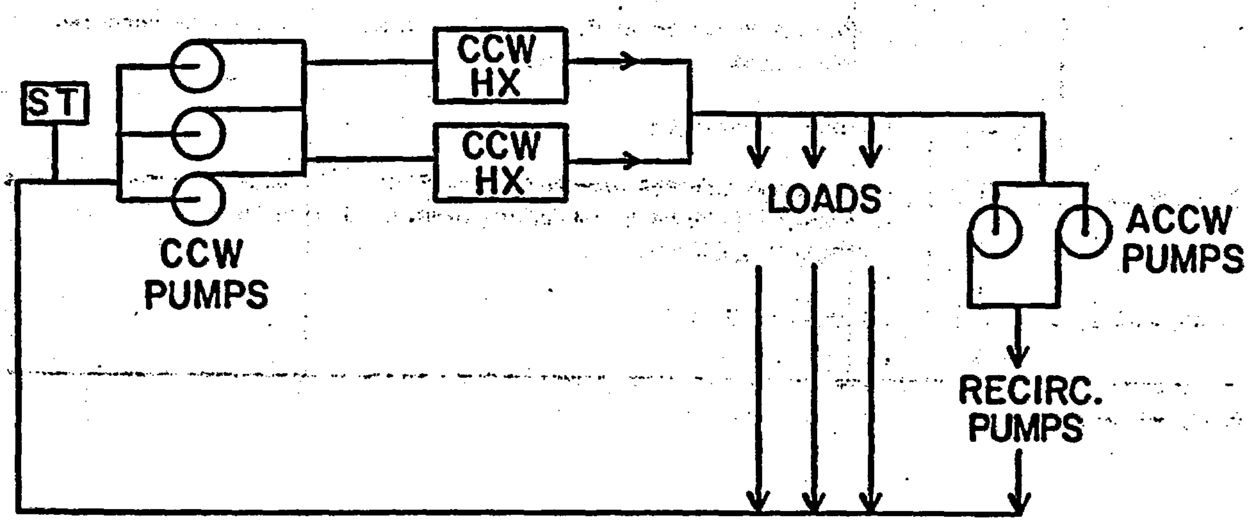
HXs: Number(%): 2 (100%) Surge Tks: 1

Cooling by: Service Water System Designer: West/VE&C

Loads: RCP m & t.b., chg pumps, XLDHX, Misc, SFPHX, SWXH, LDHX, SI pumps, RHX, Recirc. pumps

Notes: 2 pumps needed normally. 1 pump immed. post-acc., then 2 later. Auxiliary Coolant System is CCW & RHR + SFP cooling. Also 2 ACCW pumps at 80 gpm and 100' head.

Instrumentation:	Indication	Alarms	Interlocks
	Pump disch. P HX outlet T & F Pump inlet T & rad mon. Component T & F	Rad mon Lo pump P Lo flow Hi HX T Hi/Lo RCP flow	-Auto close S.T. vent on hi rad -Start pump on lo P -CIV isol. -Close valve on Hi RCP flow -Start ACCW pumps on ESF



Sketch

Table A.1
WESTINGHOUSE PLANTS - CCW SYSTEM SUMMARIES

PLANT	PUMPS	HXs	SURGE TANKS	LOADS *	COMMENTS
Beaver Valley 1	3	3	1	RCPm & t.b. XLDHX Non-Reg.HX SWHX SFP-HX RHR-HX RHR-P Misc.	
Beaver Valley 2	3	3	2	RCPm & t.b. XLDHX Non-Reg.HX SWHX SFP-HX RHR-HX RHR-P Misc.	
Braidwood 1 & 2	5	3	2	RCPm & RCP t.b., SFPHX, LDHX, XLDHX Misc, SWHX, RHR-P&HX, PDP	Shared system
Byron 1 & 2	5	3	2	RCPm & t.b. SFPHX LDHX XLDHX Misc SWHX RHR-P PDP RHR-HX	Shared system
Callaway 1	4	2	2	RHR-HXs LDHX XLDHX RCPm & t.b. SWHX SFP HX Misc RHR-P CS-P SI-P Chg-P PDP	SNUPPS
Catawba 1	4	2	2	RHRHX RHR-P CCW-P AFW-P CS-P SI-P Chg-P LDHX SWHX SFPHX Misc XLDHX RCPm & t.b.	
Catawba 2	4	2	2	Same as Catawba 1	
Comanche Pk.1	2	2	1	RHR-P CS-P RHR-HX CS-HX CW-cond CRAC UPS-AC H_2 Recomb. Misc. PDP LDHX SWHX SFPHX RCPm & t.b. XLDHX	X-connection between units
Comanche Pk. 2	2	2	1	Same as Comanche 1	*
Cook 1	3	2	1	RHR-HX RHR-P Chg-P SI-P CS-P PDP Misc. SWHX LDHX XLDHX RCPm & t.b.	Units are x-connected
Cook 2	2	2	1	Same as Unit 1	
Diablo Canyon-1	3	2	1	Cont.Fan Clr RHR-HX RHR-P Chg-P SI-P CCW-P SFP-HX SW-HX LD-HX XLD-HX Misc. RCPm & t.b. PDP	Some x-connected at loads with Unit 2
Diablo Canyon-2	3	2	1	Same as Unit 1	

* See Abbreviation Sheet

Table A.1 (Cont'd.)

PLANT	PUMPS	HXs	SURGE TANKS	LOADS	COMMENTS
Farley-1	3	3	1	RHR-HX LDHX XLDHX SWHX SFP-HX RCPm & t.b. Misc. RHR-P SI-P Chg-P	No x-connection between units
Farley-2	3	3	1	Same as Unit 1	
Ginna	2	2	1	RHR-HX RCPm & t.b. RHR-P SI-P CS-P SWHX XLD-HX Non Reg.HX Misc.	
Haddam Neck	2	2	1	RHR-HX SWHX Misc	
Indian Pt-3	3	2	2	SI-P RHR-P Recirc-P Chg-P RHX SFP HX SWHX XLDHX Non-regen HX Misc RCPm & t.b.	4-ACCW-P's for recirc.-P loop
Indian Pt-2	3	2	1	Same loads as IP-3	2-ACCW-P's
Kewaunee	2	2	1	RHR-HX RCPm & t.b. LDHX XLDHX SWHX RHR-P SI-P CS-P Misc.	
McGuire 1	4	2	1	RHR-HX SFP-HX LDHX XLD HX SWHX RCPm &t.b. Misc RHR-P Chg-P SI-P	
McGuire 2	4	2	1	Same as unit 1	
Millstone 3 RPCCW	3	3	1	RCPm & t.b. XLDHX SWHX Cont.air clg.SFPHX SI-P RHR-HX RHR-P Misc M/U Water to: SIP Clr CP Clg Chilled Water, etc.	
Millstone 3 SIP Cooling	2	2	1	SI-Pumps	
Millstone 3 Chg. P Cooling	2	2	1	Chg-Pumps	
North Anna 1 & 2	4	4	1	RCPm & t.b. XLDHX Non-Reg. HX SWHX RHRHX RHR-P SFPHX CRDM clr Misc.	Shared system between 2 units
Point Beach-1	2	1 & 1sh	1	RHRHX RCPm & t.b. Non Regen HX XLDHX SWHX RHR-P, SI-P CS-P Misc.	X-connection between units
Point Beach-2	2	1 & 1sh	1	Same as Unit 1	

Table A.1 (Cont'd.)

PLANT	PUMPS	HXs	SURGE TANKS	LOADS	COMMENTS
Prairie Island Unit 1	2	2	1	RHX RHR-P SI-P CS-P RCPm & t.b. LDHX XLDHX SWHX SFPHX Misc.	X-connection between units
Prairie Island 2	2	2	1	Same as P.I. Unit 1	
Robinson 2	3	2	1	RHRHX RCPm & t.b. Non. Reg.HX SWHX XLDHX RHR-P Misc. SI-P CS-P Chg-P SFPHX CRDHX	
Salem 1	3	2	1	RHRHX RCPm & t.b. LDHX SWHX SFPHX Misc RHR-P SI-P Chg-P	No X-connection between units
Salem 2	3	2	1	Same as Unit 1	
San Onofre-1	3	2	1	RHRHX RHR-P SFP-HX RCPm & t.b. XLDHX SWHX Misc Chg-P Recirc-HX	
Seabrook 1	4	2	2	CS-P CS-HX RHR-P RHR-HX SI-P Chg-P Cont. clr RCPm & t.b. Misc SFP-H X LDHX XLDHX	2-RCP t.b. pumps & loop
Sequoyah 1 & 2	5	3	2	RHRHX RCPm & t.b. SWHX SFPHX Misc LDHX RHR-P SI-P Chg-P XLDHX CS-P	Shared between units
Shearon Harris	3	2	1	RHRHX, RHR-P, LDHX, SWHX, XLDHX, SFP-HX Misc, RCPm & t.b.	
South Texas 1	3	3	1	RHRHX RHR-P Cont.Fan clr Boron Inj-P RCPm & t.b. Chg-P LDHX XLDHX SWHX SFPHX Misc	No X-connection between units
South Texas 2	3	3	1	Same as Unit 1	
Summer	3	2	1	RHRHX RHR-P Rx Bldg S-P LDHX XLDHX SWHX SFPHX Misc. RCPm & t.b.	3 RCP t.b. booster pumps
CCW				RHR-P SFPHX Cont. Air Clr Misc.	
Surry 1 Chg-P CW	2	2	1	Chg-P	
Surry 2 Chg-P CW	2	2	1	Chg-P	

Table A.1 (Cont'd.)

PLANT	PUMPS	HXs	SURG TANK	LOADS	COMMENTS
Trojan	3	2	2	RHRHX LDHX XLDHX RCPm & t.b. SWHX Cont. Air Clr SFPHX Misc RHR-P CS-P SI-P Chg-P	
Turkey Pt-3	3	3	1	RHRHX RCPm & t.b. Non.RegHX XLDHX SWHX Misc RHR-P SI-P Chg-P CS-P SFPHX Cont. CRD clr	Some X-connection at loads
Turkey Pt-4	3	3	1	Same as Unit 3	
Vogtle 1 CCW	6	2	2	SFP RHP-P RHR-HX	
Vogtle 1 ACCW	2	2	1	RCPm & t.b. SWHX LDHX XLDHX Misc ACCW-P & m	ACCW system is separate
Vogtle 1 TOTAL	8	4	3	All above two lines	
Vogtle 2	Same as Vogtle Unit 1				
Watts Bar 1&2	5	3	2	RHRHX RCPm & t.b. LDHX XLDHX SWHX SFPHX Misc RHR-P Chg-P SI-P	Common system for 2 units
Wolf Creek	4	2	2	RHRHX LDHX XLDHX RCPm & t.b. SWHX SFPHX Misc. RHR-P CS-P SI-P Chg-P PDP	SNUPPS
Yankee Rowe	2	2	1	MCPm SFPHX Shutdown clr L.P.Surge TK N.Sh TK clr	
Zion 1&2	5	3	2	RHRHX RCPm & t.b. LDHX XLDHX SWHX SFPHX Misc RHR-P Chg-P SI-P	Common system for 2 units

Table A.2
COMBUSTION ENGINEERING (CE) PLANTS - CCW SYSTEM SUMMARIES

PLANT	PUMPS	HXs	SURGE TANKS	LOADS	COMMENTS
ANO-2	3	3	2	RCPm & sc, LDHX, Misc	Non SR only
Cal.Cliff-1	3	2	1	SDHX,LPSI-P,HPSI-P,LDHX,Misc, RCPm & sc CRDM Clg	
Cal.Cliff-2	3	2	1	Same as Unit 1	
Ft. Calhoun	3	4	1	SDHX,LDHX,SFPHX,RCPm & sc,Chg-P,LPSI-P HPSI-P,CS-P,CEDM,Misc,Cont.Air Clg, CRAC	Raw Water Back up
Maine Yankee	4	4	2	RHR-HX,SFPHX,LDHX,SWHX,LPSI-P, Chg-P, RCP-m,Misc,CEA clr, Cont.Air.Clr	2 sep. subsystems
Millstone-2	3	3	1	SDHX,LDHX,SFPHX,CS-P,HSPI-P,LPSI-P,Cont. Air.Clg, Misc, ESF Rm Clg, RCPm & t.b., CEDM Clg	
Palisades	3	2	1	SDHX,SFPHX,LDHX,Misc.,CRD seals,RCP oil clr,HPSI-P,LSPI-P,CS-P,Chg-P	Service Water backup to ESF-Pumps
Palo Verde 1 ECW	2	2	2	SDHX, Ess. Chiller SFPHX	X-connection to NCW
Palo Verde 1 NCW	2	2	1	RCPm & sc,Misc,Norm Chiller, LDHX, CEDM clr, SFPHX	
Palo Verde 2 ECW	2	2	2	Same as Unit 1	X-connection to NCW
Palo Verde 2 NCW	2	2	1	Same as Unit 1	
Palo Verde 3 ECW	2	2	2	Same as Unit 1	X-connection to NCW
Palo Verde 3 NCW	2	2	1	Same as Unit 1	
San Onofre-2	3	2	2	SDHX LDHX SFPHX HPSI-P LPSI-P CS-P CCW-P RCPm & sc Misc Cont.Air clr CR Chiller CEDM clr	

Table A.2 (Cont'd.)

PLANT	PUMPS	HXs	SURGE TANKS	LOADS	COMMENTS
San Onofre-3	3	2	2	Same as Unit 2	
St. Lucie-1	3	2	1	SDHX Cont.Fan Clr LPSI-P HPSI-P CS-P SFPHX LDHX Misc CEA Air Clr RCPm & sc	
St. Lucie-2	3	2	1	SDHX Cont.Fan Clr HPSI-P CRAC SFP-HX LDHX Misc. CEDM Clr RCPm & sc	
Waterford-3	3	4	1	SDHX LDHX SFPHX EDGs HPSI-P LPSI-P CS-P RCPm & S.C. Cont. Fan Clr Chillers Misc CEDM Clr	Cooling by CT or ACCW

Table A.3
BABCOCK & WILCOX (B&W) PLANTS - CCW SYSTEMS SUMMARIES

PLANT	PUMPS	HXs	SURGE TANKS	LOADS	COMMENTS
ANO-1	3	3	2	RCPm & s.c, LDHX, SWHX, SFPHX, Misc. CRD clg	Non-SR. System 4 booster pumps
Crystal River-3	3	4	1	RCPm & sc, SFPHX,SWHX,LDHX,MU-P,NSCCS-P NSSWS-P,RxB. Fan Clg Vent Fan-m, Cont. Chill CRDM clg Misc	2 booster pumps
Davis-Besse	3	3	1	DHR-HX, EDG-HX, LDHX, SWHX, SFP-HX, DHR-P HPI-P, MU-P, RCPm & sc, Misc, CRD Clg	
Oconee-1	2	1 & 1sh	1	RCPm & t.b., LDHX, CRD Clg	Non S.R.
Oconee-2	2	1 & 1sh	1	RCPm & t.b., LDHX, CRD Clg	Non S.R.
Oconee-3	2	2	1	RCPm & t.b., LDHX, CRD Clg	Non S.R.
Rancho Seco NSCW	2	2	2	DHR-HX, RX Bldg Cooling Units (emerg.)	2 sep. trains
Rancho Seco CCW	2	2	1	LDHX,Rx Bldg Cooling Units (Norm), Misc RCPm & t.b., CRD clg Turb. Plant, SWHX SFPHX, HPI-P, MU-P	
TMI-1 (ICS)	2	2	1	LDHX, RCDT, CRD, RCPm	
TMI-1 (NSCCWS)	3	4	1	RCPm & t.b. SFPHX, RB Fan clg, MU-P,CRAC Area Rm Clg, Misc	
TMI-1 (DHRCCCWS)	2	2	2	DHR-HX, DHR-P, DHRCCW-P, MU-P, RB Spray-P	2 sep. trains

Table A-4 Abbreviations

Key		Definition
ACCW	-	Auxiliary Component Cooling Water
AFW	-	Auxiliary Feedwater
CEDM	-	Control Element Drive Mechanism
Chg	-	Charging
CP	-	Charging Pump
CRAC	-	Control Room Air Conditioning
CR	-	Control Room
CS	-	Containment Spray
CW	-	Chilled Water
DHR	-	Decay Heat Removal
ESF	-	Engineered Safety Feature
HPI	-	High Pressure Injection (B&W)
HPSI	-	High Pressure Safety Injection (CE)
HX	-	Heat Exchanger
LD	-	Letdown
LPSI	-	Low Pressure Safety Injection
MCP	-	Main Coolant Pump
Misc	-	Miscellaneous Loads
M/U	-	Make Up
NCW	-	Nuclear Cooling Water
N.Sh.TK	-	Neutron Shield Tank
Non-Reg./ Regen.	-	Non Regenerative
P	-	Pump
PDP	-	Positive Displacement Pump
RCPm & t.b.	-	Reactor Coolant Pump Motor and Thermal Barrier
Recirc	-	Recirculation
RHR	-	Residual Heat Removal
RHX	-	Residual Heat Exchanger
sc	-	Seal Cooler
SE	-	Seal Water
SFP	-	Spent Fuel Pool
sh	-	Shared
SI	-	Safety Injection
SNUPPS	-	Standard Nuclear Power Plant System
SR	-	Safety Related
SWHX	-	Service Water Heat Exchanger
UPS	-	Uninterruptable Power Supply
XLD	-	Excess Letdown

To illustrate some of the variations between plants' CCW systems the following two tables were prepared. Table A-5 shows that the name of the system serving the function of CCW varies somewhat. Additionally, the name of the open cooling water system (e.g. Service Water) providing cooling to the CCW heat exchangers also varies. Component Cooling Water and Service Water are clearly the two preferred names and are generally used in this report, however, many other exist.

Table A-5 CCW System Summary - All PWR Plants

This review included: 50 PWR Sites 79 PWR Units 85 CCW Type Systems

The names of the systems performing the component cooling function varied considerably from plant to plant as shown here.

Names of CCW Systems:

Component Cooling Water (CCW)	60
Essential Cooling Water	3
Nuclear Cooling Water	3
Intermediate Cooling System	2
Primary CCW	2
Auxiliary CCW	2
Charging Pump Cooling Water	2
Component Cooling System	2
Charging Pump Cooling System	1
Safety Injection Pump Cooling System	1
Reactor Plant CCW	1
Reactor Building Closed Cooling Water	1
Secondary CCW	1
Nuclear Services Cooling Water	1
Nuclear Sercices Closed Cooling Water	1
Nuclear Services Closed Cycle Cooling Water	1
Decay Heat Removal Closed Cycle Cooling Water	1
Total	85

The names of the open cooling water system providing cooling to the CCW system heat exchangers also varied from plant to plant as shown here:

Names of Cooling Water Systems:

Service Water	31
Essential Service Water	7
Nuclear Service Cooling Water	4
Nuclear Service Water	4
Nuclear Services River Water	3
Nuclear Services Sea Water	1

Names of CCW Systems: (Cont'd)

Nuclear Services Raw Water	1
Low Pressure Service Water	3
Plant Cooling Water	4
Intake Cooling Water	4
Cooling Water	3
Essential Spray Pond System	3
Essential Cooling Water	2
Station Service Water	2
Salt Water	3
Salt Water Cooling	2
Auxiliary Salt Water	2
River Water	2
Raw Water	1
Plant Service Water	1
Essential Raw Cooling Water	1
Dry Cooling Tower (Auxiliary CCW)	1
Total	85

Table A-6 illustrates the wide variation in CCW system design in terms of the number and size of the main components. Some of the variation in size is because in plants with multiple CCW systems, often one system is quite small and may serve only one load, such as the charging pumps cooling. Some of the CCW systems in recent plants (e.g. Comanche Peak and Palo Verde) are very large and serve many loads. Table A-6 also lists the typical loads served by CCW systems. The safety-related loads are broken down by NSSS vendor, since the three vendors have slightly different names for similar components.

Table A-6 CCW Systems - Per Unit Data Variations (All PWRS)

# of Pumps:	2 to 8
Pump Flow:	25 gpm to 17,500 gpm per pump
Pump Head:	54' to 275' (TDH)
# HXs:	1 1/2 to 8 (1/2 means 1 shared HX between 2 units)
#Surge Tanks:	1/2 to 4 (1/2 means 1 shared tank between 2 units)

Typical Safety Related Loads:

Westinghouse:	Residual Heat Removal (RHR) Heat Exchangers (HX), RHR pumps, Safety Injection Pumps, Containment Spray Pumps, Containment Coolers.
Combustion Eng:	Shutdown HXs, LPSI Pumps, HPSI Pumps, Containment Spray Pumps, Various Chillers, Containment Air Coolers.
B & W:	Decay Heat Removal HXs, DHR Pumps, HPI Pumps;, Reactor Building Fan Coolers.

Table A-6 (Cont'd)

Typical Non Safety Related Loads:

Reactor Coolant Pump Motor and Thermal Barrier (Seal Cooler), Letdown HX, Excess Letdown HX, Seal Water HX, Spent Fuel Pool HX, Charging or Makeup Pumps, Contr ol Rod Drive or Control Element Drive Mechanism Cooling, Miscellaneous Loads.

One of the reasons for the wide variation in CCW system design is that the system is typically designed by the plants' Architect-Engineer (AE) and not the NSSS vendor. Thus, due to the much larger number of AEs involved, there are a larger number of CCW system designs. Table A-7 below shows the distribution of CCW systems by the AE that designed them.

Table A-7 CCW System Designers

Designer Name	# of CCW Systems Designed
Bechtel	26
Stone & Webster	12
Duke Power	7
Gilbert Assoc.	7
Pioneer Services and Engineers	5
Ebasco	4
Sargent & Lundy	3
Gibbs & Hill	3
United Engineers & Constructors	3
American Elec. Power Service Corp.	2
Pacific Gas & Electric	2
TVA	2
Brown & Root	2
Unknown	7
Total	85

A.3 CCW SYSTEM DESCRIPTIONS

A.3.1 Overall System Arrangement

As mentioned in Section A.2, there are plants with multiple CCW systems (either 2 or 3 systems per unit), plants with fully shared systems, and even one site that has both multiple and shared systems. Table A-8 lists those plants with multiple and shared systems. A plant with a shared system between units uses one common set of pumps, surge tanks, and heat exchangers for both units serving each unit's loads through separate piping headers. In the summary statistics presented herein, the number of components for a unit using a shared system is obtained by dividing the total by two. This results in a half of a component, if the total number of components is odd.

Table A-8 Units With Multiple or Shared Systems

<u>Two Units With 3 Systems</u>

Millstone 3	Reactor Plant Component Cooling Water System
	Charging Pump Cooling System
	Safety Injection Pump Cooling System
Three Mile Island 1	Decay Heat Removal Closed Cycle Cooling Water System
	Nuclear Services Closed Cooling Water System
	Intermediate Cooling System

<u>Nine Units With 2 Systems</u>

Maine Yankee
Palo Verde 1,2,3
Rancho Seco
Surry 1,2
Vogtle 1,2

<u>Seven Sites (14 Units) with Shared Systems</u>

Braidwood 1 & 2
Byron 1 & 2
North Anna 1 & 2
Sequoyah 1 & 2
Surry 1 & 2
Watts Bar 1 & 2
Zion 1 & 2

Note: Surry 1 & 2 share a CCW system but each unit also has its own Charging Pump Cooling Water System.

Cross Connections

A number of two-unit sites utilize separate, but cross-connected systems. These are noted in the comments of Tables A-1, A-2, and A-3. These cross-connections are at various places; the pump suction and discharge, the surge tanks, the CCW heat exchangers, and at various loads. Some plants have several cross-connection points, others onlytwo. Some of the plants with multiple systems have cross-connections between the systems.

Shared Systems

Figure A-3, shows that there are seven sites or 14 units with a fully shared system. Five of these sites use a 5 pump, 3 heat exchanger (HX), 2 surge tank (5,3,2) design. This is equivalent to 2 1/2 pumps, 1 1/2 HXs, and 1 surge tank per unit, which is less than the most common one-unit design of 3 pumps, 2 HXs, and 1 surge tank. However, the shared design has additional redundancy by having the extra components available to either unit should only one component fail. Thus for single (or perhaps even two) component failures the shared design appears superior. For multiple component failures (e.g., due to common cause) which fail the entire CCW system the shared design is inferior, since system failure would affect two units instead of only one. While five sites have the (5,3,2) design, the other two sites have a 4 pump, 4 HX, 1 surge tank (4,4,1) design. This design is not as reliable in the surge tank area, where there is only one surge tank for two units.

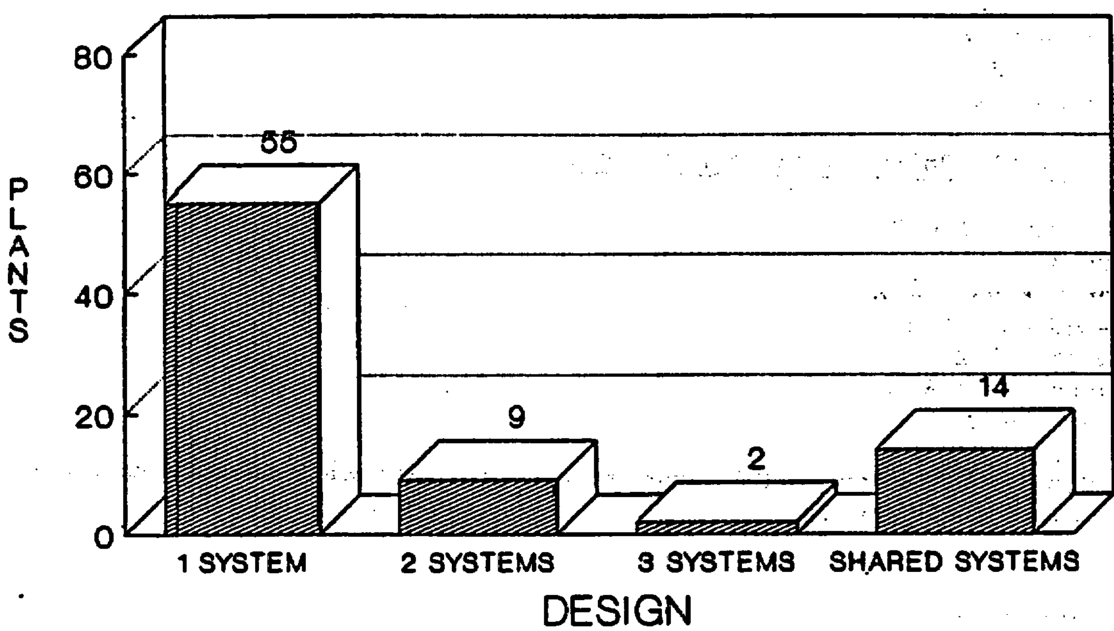


Figure A-3 CCW System Designs

Multiple Systems

The multiple system design is generally more reliable in that more components and greater redundancy is provided. Failure of one entire system does not affect all loads. One good feature of some multiple system designs is that Reactor Coolant Pumps (RCP) seals and high pressure injection pump cooling are provided by different systems. This limits the likelihood that a single system failure will cause an RCP seal LOCA and fail the injection system needed to mitigate it (Generic Issue 65). In some of the plants with multiple CCW systems, one system supplies safety-related loads (e.g. Decay Heat Removal) and the other system supplies non-safety related loads. Rancho Seco and Vogtle are examples of this design, where the RCP seals are cooled by the non-safety systems, which have less redundancy than the typical CCW system.

Since loss of CCW to the RCP seals is significant (Generic Issue 65), this is one calse when the multiple system design may be less desirable than the typical one system design. Additionally, the safety-related CCW system, atseveral of the plants with two CCW type systems, has only two pumps, which again is less than that provided in the typical one CCW system design.

Unique Arrangements

Three plants have unique arrangements. Seabrook has one CCW system with two independent loops, plus a separate RCP thermal barrier cooling loop that has two pumps and two series HXs. The two series HXs are cooled as a load from each CCW loop.

Surry has a typical CCW system shared between two units. However, tapping off the CCW pump suctions of this system is a separate chilled component cooling subsystem with 3 pumps and 3 HXs. These HXs are cooled by Chilled Water. This subsystem provides added cooling for some loads when normal CCW temperatures are not low enough. Surry also has two separate charging pump cooling systems, one for each unit.

Waterford uses two types of cooling in series for its CCW System. First, the CCW water is sent to a dry cooling tower and then to a standard, shell and tube CCW HX. This CCW HX is cooled (when needed) by an Auxiliary CCW system rejecting heat to a wet cooling tower. Either of these HXs can be bypassed depending on which is providing cooling.

Header Arrangements

There are many different designs for the pipe header arrangements of the CCW systems. Two common arrangements between pumps and heat exchangers (HX) and a few common arrangements for loads will be presented here. Figure A-4 shows the two common arrangements of the pump-to-HX header.

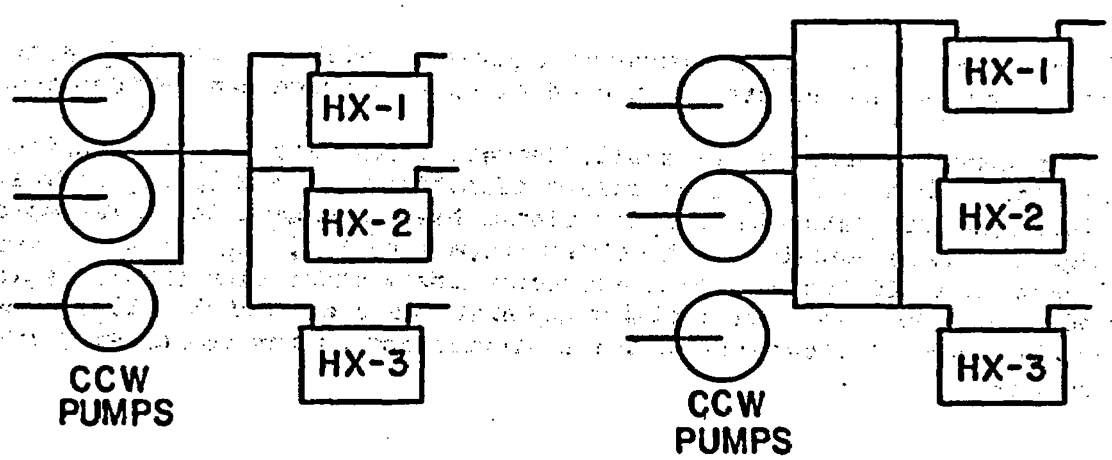


Figure A-4 Typical CCW Pump-HX Header Arrangement

The header arrangements to the loads are much more varied. Figure A-5 shows the simplest design commonly used in early plants.

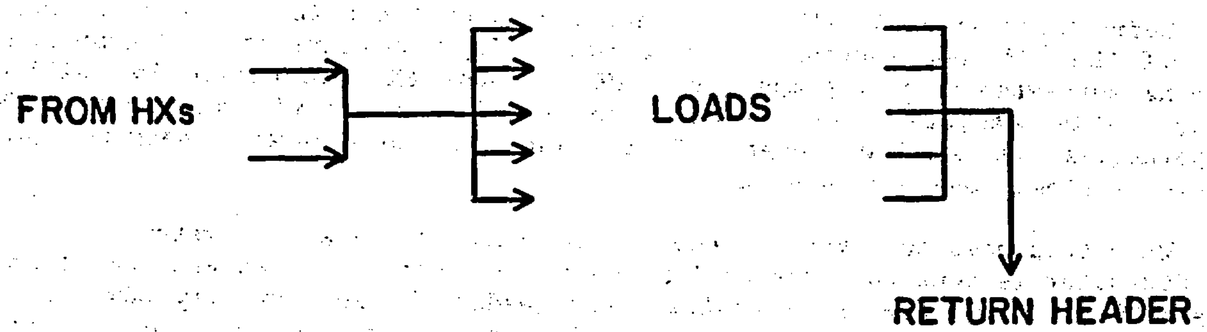


Figure A-5 Typical CCW HX-Loads Header Arrangement

Figure A-6 shows a very common arrangement, which has two safety related (SR) loops and one non-safety related (Non-SR) loop which can be isolated.

Other common header features utilized are a separate loop for inside containment loads such as RCPs. This allows isolation of these loads, if there is an accident inside containment. Many plants have automatic isolation-on-accident signals and on sensing of high flow from the RCPs, signifying a possible tube leak in the RCP thermal barrier cooler. Many other complexities exist in the header arrangements of individual plants. These generally will not affect the overall system performance, but are significant as far as individual loads are concerned.

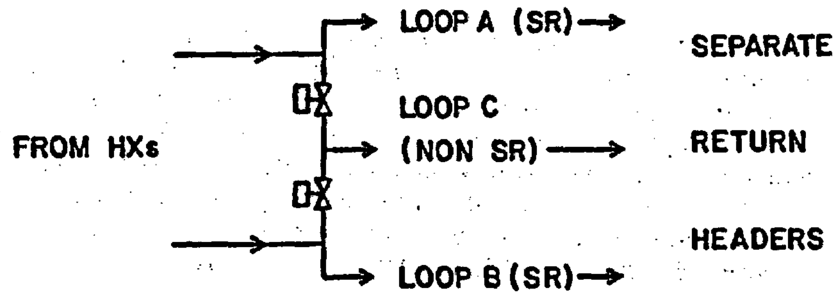


Figure A-6 Typical CCW Safety Related Load Header Arrangement

Emergency Water Supply

Several plants have backup emergency water supplies to their CCW system. These supplies generally are not as pure as the normal supply, but come from a safety-grade system such as Service Water. Such backups are mostly found in Combustion Engineering plants, but a few Westinghouse and B&W plants also have them. These emergency supplies provide additional CCW water should surge tank level drop due to leakage in the system. The emergency supply injects either to the entire system at the CCW pump suction or simply provides cooling to certain selected CCW loads via the normal CCW piping.

BWR Systems

Boiling Water Reactor (BWR) Plants use a system similar to CCW for cooling called Reactor Building Closed Cooling Water (RBCCW). Two BWR-RBCCW Systems were reviewed briefly to determine their similarity to the PWR systems. These systems were found to be somewhat similar in both function and design to the PWR CCW systems. This report, however, does not address the BWR systems.

A.3.2 CCW System Major Components

The number of individual CCW components at each NPP unit varies just as the overall system varies. The most common arrangement is three pumps and two heat exchangers; however, some units have as many as eight of each. Figure A-7 shows the distribution of pumps in CCW systems and Figure A-8 illustrates the distribution for heat exchangers. Those plants with 7 or 8 components per unit are the ones with multiple systems. Plants with 1.5 or 2.5 components per unit are the ones sharing 3 or 5 components between two units. Those with more pumps and more HXs provide greater reliability, albeit with the reservations about common cause failure noted in the previous section. The CCW pumps are all motor-driven centrifugal types. For some small systems serving only a few components, flows are as low as 25 gpm. For the large systems at the newer plants that cool many components, flows are as high as 17,500 gpm per pump. Pump heads (total developed head or TDH) likewise vary from 25 feet to 275 feet . The majority of pumps fall in the flow range of 3000 gpm-10,000 gpm, in the TDH range of 150' - 225', and from 400-700 horsepower.

The pumps are driven by either 480 volt or 4160 volt ac electrical power. In most cases they are powered from safety-related, Class 1E buses, which can receive power either normally from onsite or offsite sources or in an emergency from the onsite diesel generators. A few systems have non-safety related pumps that are powered from non-safety buses. Also, a few plants have small booster CCW pumps that provide added flow at higher pressure to particular components such as RCP seals or control rod drive mechanisms. These booster pumps are on the order of 200 gpm, 200' TDH, and 15 horsepower.

The majority of all HXs are of the shell and tube type, with CCW on the shell side and Service Water on the tube side. The HXs are typically downstream of the CCW pumps before the loads, but several plants, primarily the B&W designs, have the HXs after the loads and before the CCW pumps. This placement affects the differential pressure (DP) across the tubes between SW and CCW; the temperature of water that the CCW pump seals see; the CCW pump net positive suction head available; and the temperature of water supplied to the loads. Regarding the DP between CCW and SW, plants have chosen two different designs. Some plants operate with CCW pressure higher than SW and state in their FSAR that this prevents the impure SW from leaking into the CCW system and causing corrosion. Other plants operate with SW pressure higher than CCW and state that this ensures that the potentially contaminated CCW system will not leak out to the environment. Whatever the design, a tube leak during normal operation in the CCW-HX is relatively easily detected by changes in CCW surge tank level, and the consequences are not severe.

Figures A-9 and A-10 illustrate the number of surge tanks used for plants with shared CCW systems and plants without shared systems. The most common design for the non-shared system is one surge tank. Several single surge tanks have an internal baffle, dividing the tank into two halves, so that a single leak does not incapacitate the entire system. All of those units having 3 or 4 surge tanks are units with multiple CCW systems. Regarding those plants with shared systems, one plant has only a single surge tank for both units. Another plant that has 1 shared system and one individual system per unit, has 3 surge tanks. However, here the shared CCW system has only 1 surge tank for both units. While this design is not as reliable as the others, it should be noted that the surge tank does not show as a dominant cause of failure in the CCW-PRA study. The great majority of plants have their surge tank located on the suction side of the main CCW pumps to provide net positive suction head for the pumps, in addition to serving as a surge volume for the system. A few designs place the tanks on the pump discharge. Most plants also use the surge tanks for normal make-up to the CCW system from a demineralized water system and for addition of corrosion inhibitors. Generally there are provisions for recirculation to ensure good mixing of added chemicals. The majority of surge tanks are vented to the atmosphere through a vent line with

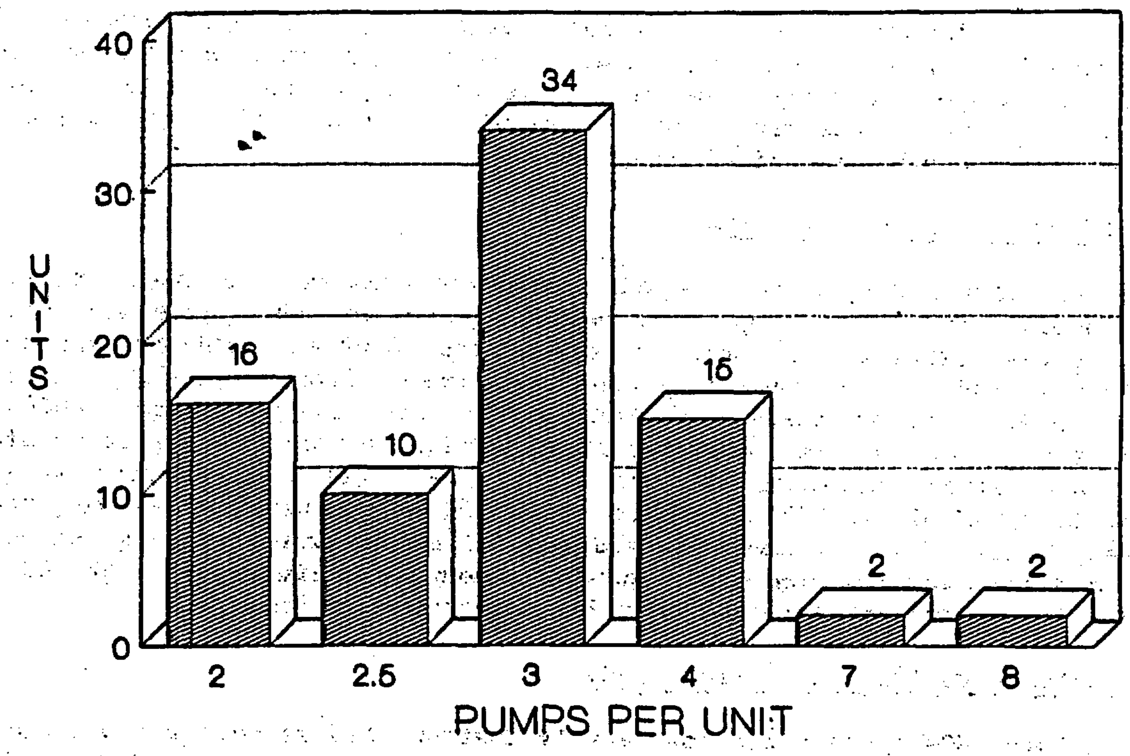


Figure A-7 CCW Pumps Per Unit

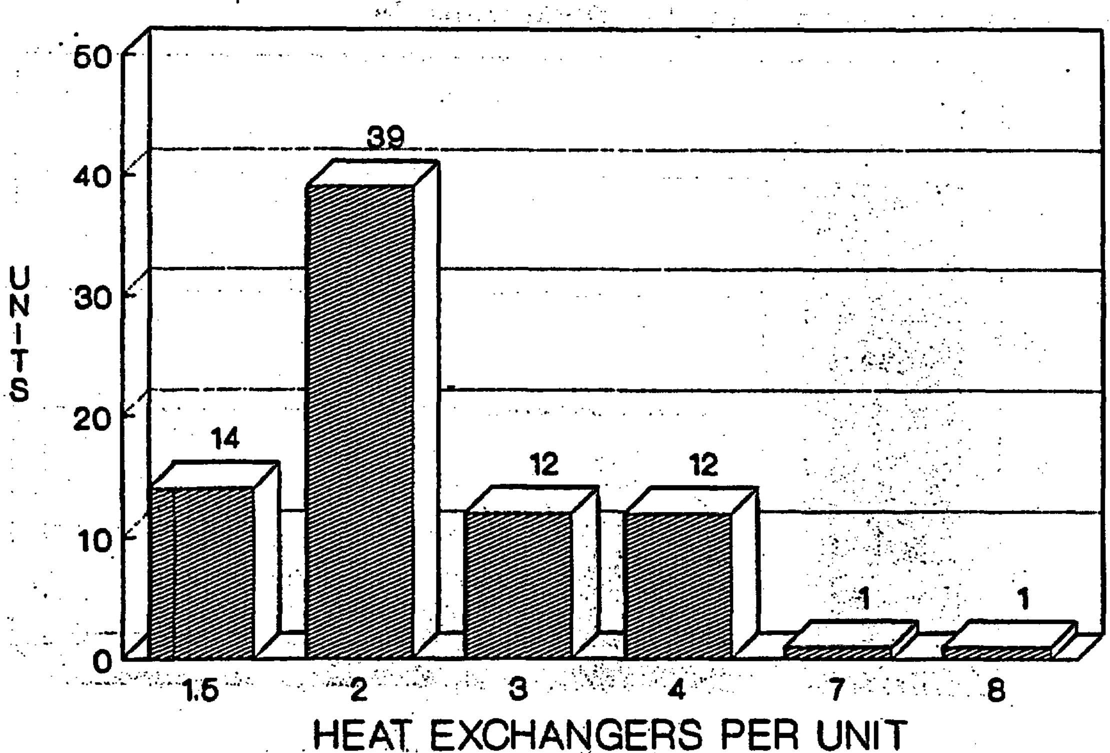


Figure A-8 CCW Heat Exchangers Per Unit

an automatic isolation valve. If primary coolant were to leak into the CCW system, say through an RCP-HX leak, then a radiation monitor would alarm and automatically close the surge tank vent valve to prevent the overflow of contaminated water out of the vent line. A few plants (e.g. Trojan) operate with normally isolated and pressurized surge tanks.

A.3.3 CCW System Electrical Power

As with most NPP systems, the CCW system receives various types of electrical power. The major components needing power are the pumps and motor operated valves (MOVs). The pumps receive either 480V or 4160V AC depending on their size and design, generally (but not always) from the safety related buses. The MOVs are typically powered from 480V motor control centers (MCCs). Some of the MOVs, such as containment isolation valves, train separation valves, and valves isolating the non-safety loads are powered from safety related, Class 1E MCCs. The remainder of the MOVs are usually powered from non-safety MCCs. 125V DC power is used for solenoid-operated valves, circuit breaker control, and various logic and instrumentation circuits. 120V AC is also used for instrumentation and sometimes control circuits.

A.3.4 Instrumentation and Control (I&C)

As a rule, CCW systems are not heavily instrumented as compared with some of the other systems in a NPP, but the instrumentation installed varies between plants. Generally, the newer plants have significantly more instrumentation than plants completed in the early to mid-70's. This section will discuss the types of I&C equipment installed in CCW systems in three categories: indications, alarms, and interlocks/controls.

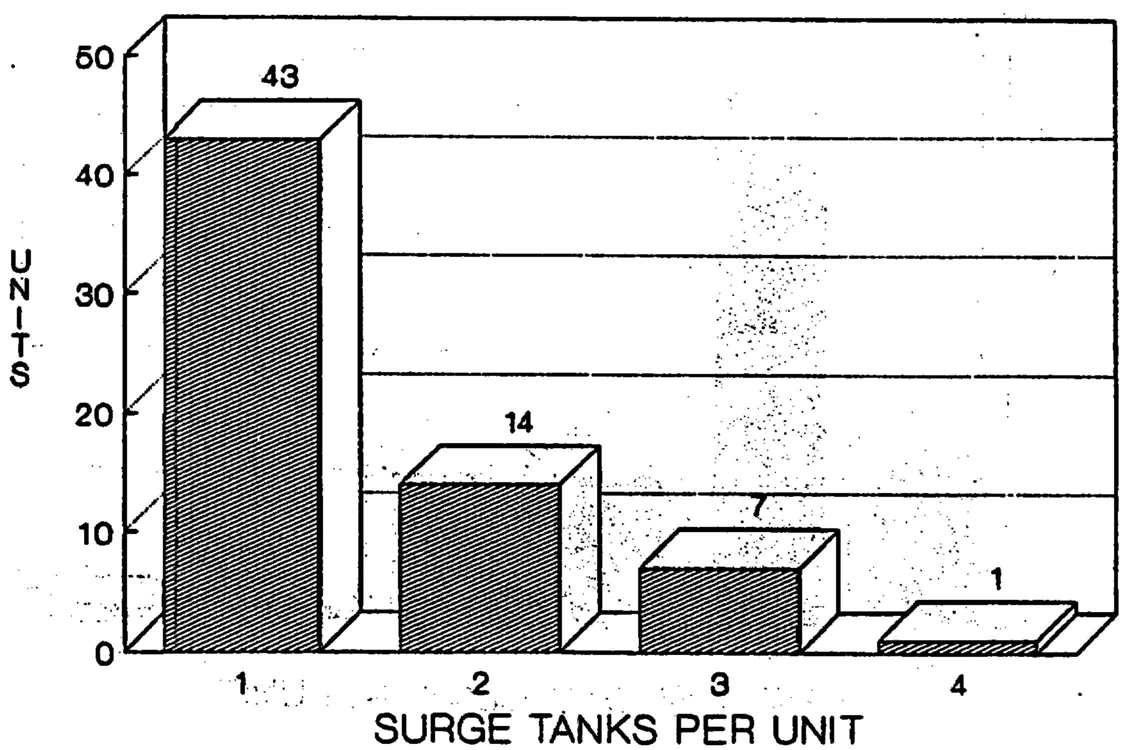


Figure A-9 CCW Surge Tanks Per Unit for Non-Shared Systems

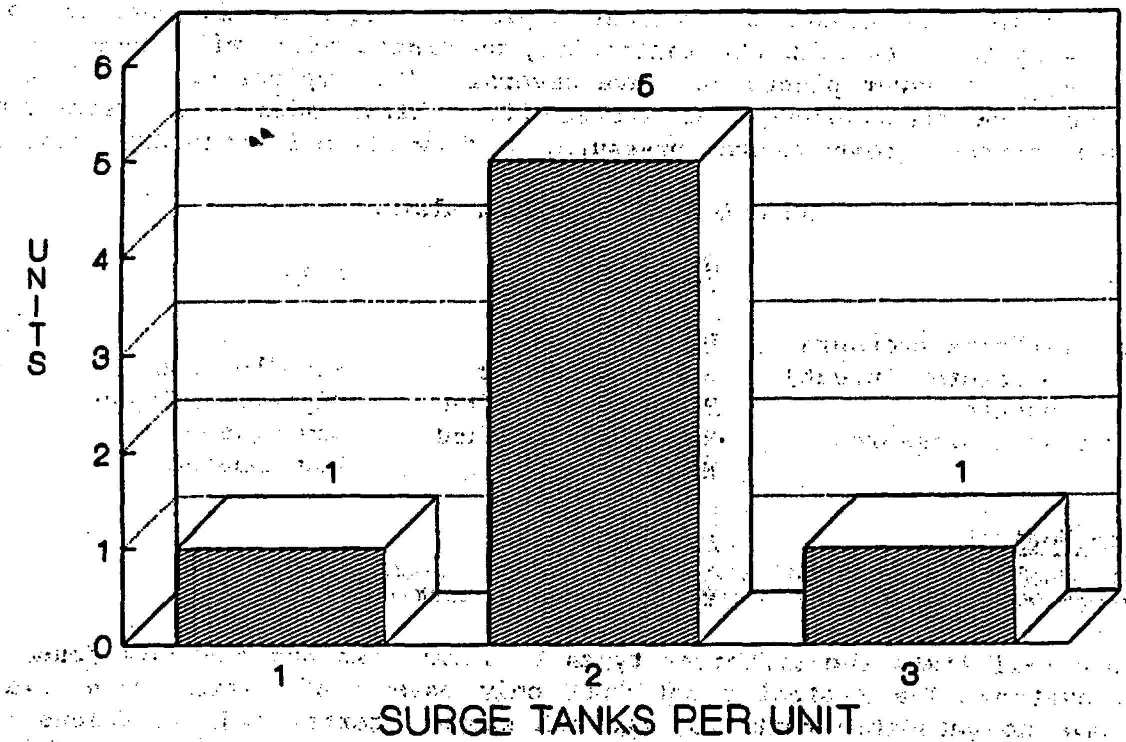


Figure A-10 CCW Surge Tanks Per Unit for Shared Systems

Table A-9 summarizes the various types of indication in CCW systems. No single plant would have all of these indicators, but some new plants have most of them. A typical plant has one system pressure indicator, a temperature indicator, a flow indicator, a radiation monitor, a surge tank level indicator, and some indicator on the RCP header. Some, but not all, of the indicators provide for a readout in the control room.

Table A-9 CCW System Indication

Pressure	Temperature	Flow
Pump discharge	CCW-HX in/out	Total
RCP header	RCP out	Pump suction
Main loops or headers	Pump suction	CCW-HX
		Individual Components
		RCP out

Radiation Monitor	Level	Miscellaneous
System	Surge Tank	Pump Vibration
		Pump Differential Pressure
		Valve Position

Table A-10 illustrates the various types of alarms in the control room for the CCW system. As with the indication, no single plant will have all of these alarms, but newer plants may have several. The typical plant will have alarms for high temperature, low system flow, high radiation, surge tank level, and perhaps a temperature, pressure, or flow alarm for the RCP header.

Table A-10 CCW System Alarms

Pressure	Temperature	Flow
Low pump discharge pressure	High HX out	
Low CCW HX pressure (in/out)	High RCP header	High/low CCW HX
Low loop pressure	High pump suction	High/low RCP out
High/low system pressure	High component out	Low system
	High loop	Low component
Radiation Monitor	**Level**	
High system radiation	High/low surge tank	

Table A-11 lists the different types of interlocks and controls found in NPP CCW systems. The typical plant will only have 2 of them. Some older plants have no automatic controls, and all stops, starts, and isolations are done manually by the operators. Many newer plants have 3 or 4 of the automatic interlocks, the most common being closure of surge tank vent in high radiation, auto-start of CCW pump, isolation of non-safety loads on ESF signal, and closure of the containment isolation valve.

Table A-11 CCW System Automatic Interlocks/Controls

Common Items

1. Closure of containment isolation valves on Engineered Safety Features (ESF) or Safety Injection (SI) signal.

2. CCW pump start on low pump discharge pressure, on ESF/SI signal, or on trip of running pump.

3. Isolation of non-safety equipment and loads from safety-related equipment on containment isolation signal, ESF signal, SI signal, low surge tank level, or low pump suction pressure.

4. Split of redundant safety related trains on containment isolation signal, low surge tank level, ESF signal, or SI signal.

5. Surge tank vent valve automatic closure on high CCW system radiation.

6. Isolation of reactor coolant pump header on leak as measured by high pressure, high temperature, or high flow.

Table A-11 (Cont'd)

Rare/Unique Items

1. CCW pump trip on high temperature.

2. CCW pump trip on low/low surge tank level.

3. Automatic start of Service Water on CCW auto start.

4. Automatic opening of valves to required CCW loads on SI.

5. Modulation of temperature control valves at components or CCW HXs.

6. Start of DC powered RCP thermal barrier emergency cooling pump on low CCW pressure or on loss of AC power.

7. Automatic cross connection of safety related CCW system to non-safety CCW system on Loss of Offsite Power (LOOP) in order to cool RCP seals. Also automatic isolation of two systems on SI signal.

A.3.5 CCW System Miscellaneous Equipment

Pipes and Pipe Supports

CCW system piping is generally seamless carbon steel pipe. Therefore, a corrosion inhibitor is used in almost all systems. Common inhibitors are chromate or nitrate compounds. Sodium hydroxide also is added sometimes to raise the pH and reduce corrosion. Some portions of the CCW system in contact with primary coolant, such as at the RCPs or letdown heat exchangers, are made of stainless steel.

Pipe supports for the CCW system are of standard power plant design with pipe hangers and restraints. Safety-related portions of the systems are also supported by seismic restraints and often snubbers. Sometimes portions of the non-safety sections are also similarly supported to prevent damage to nearby safety-related systems.

Valves

Since the CCW system is a large system serving many loads, there are a large number of valves in the system. Figures A-11 and A-12 illustrate the total number of valves found in the CCW systems on a per system basis and a per unit basis. The main reasons for differences in these figures are the plants with multiple systems and those with shared systems. Within a given plant's CCW system (and even more so between plants), there is a variety of types of valves in use including motor-operated valves (MOVs), pneumatic-operated valves, manual valves, solenoid valves, relief valves, automatic control valves, and check valves.

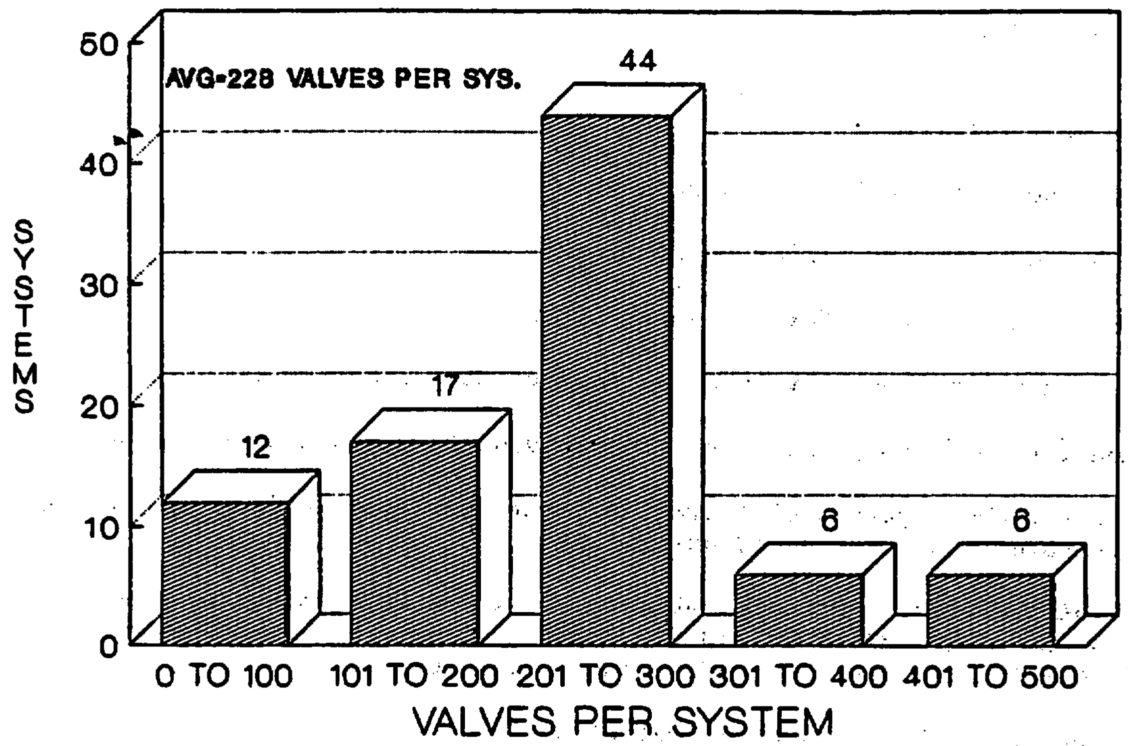


Figure A-11 CCW Valves Per System

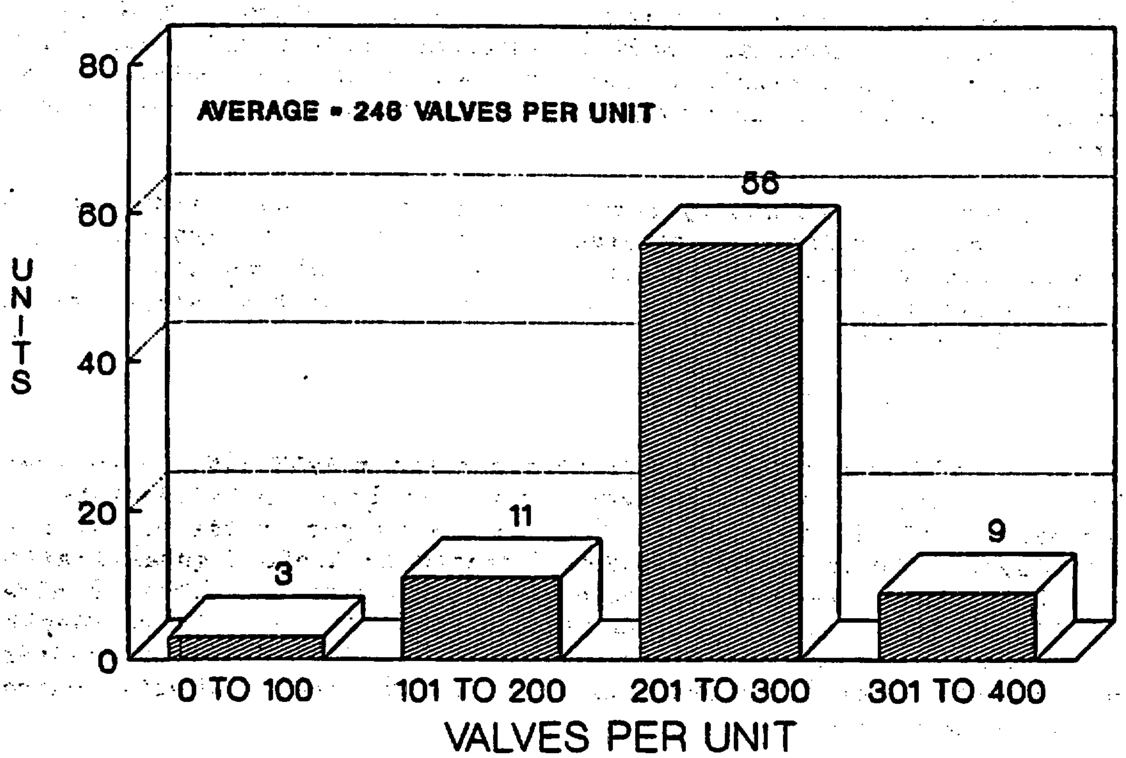


Figure A-12 CCW Valves Per Unit

A few of the MOVs will typically fall into the plant's ASME, Section XI valve surveillance testing program and be routinely tested for operability and stroke time. Some MOVs receive automatic signals for realignment on containment isolation or engineered safeguards actuation. Others are manually operated from the control room. Some of the loop or header isolation valves and some of the control valves are pneumatic. Air to the pneumatic valves is generally controlled by DC-operated solenoid valves. Manual valves are used for component and header isolation, for system venting and draining, for throttling and setting of flow rates, and for instrument root valves. Check valves are included in the discharge of all system pumps and in the discharge of a number of component headers, such as the RCP headers. Relief valves are located on the surge tank and often on each cooled, component header to provide relief in case of component isolation and subsequent heatup.

Cabling

The electric cabling to the safety-related portions of the CCW system will generally be routed in redundant cable trays that are fire-protected, and seismically supported. Large portions of CCW systems are non-safety and the power supplies and cabling will be standard power plant design. Wiring to indication and alarms is also typically non-safety related.

Structures

The CCW systems at PWRs traverse a large portion of the plant due to the amount of equipment to be cooled. Typical locations are: the auxiliary building for major components such as pumps, heat exchangers, surge tank and several loads; the containment for many other loads such as RCPs, and excess letdown heat exchangers. Sometimes they are located in other buildings such as the radwaste building or turbine building for loads.

APPENDIX B

INDIAN POINT-2 CCW SYSTEM DESCRIPTION

CONTENTS

FIGURES

TABLES

B.1 INTRODUCTION

Indian Point Nuclear Generating Unit No. 2 (IP-2) is a Westinghouse designed PWR, owned and operated by Consolidated Edison Co. and located in Buchanan, N.Y. The Architect-Engineer for the plant was United Engineers and Constructors. The plant is a four-loop design with a net output of 873 MW(e). This plant was selected as the baseline plant for our CCW system because:

1. It was representative of United States PWR CCW systems, as determined by the review of CCW designs described in Appendix A.

2. The plant has been in operation more than 14 years, which qualifies it as an aged system.

3. The utility is willing to share information with BNL.

This section describes the IP-2 CCW system in detail. A brief overview of the system is given in the summary sheet in Appendix A, Figure A-2.

B.2 OVERALL SYSTEM LAYOUT

The main CCW system components are located in the Primary Auxiliary Building (PAB) with cooled loads throughout the plant, including inside the Primary Containment. Figure B-1 shows that the system consists of three pumps in parallel pumping water to two parallel heat exchangers, which are cooled by Service Water. All water passes through a single 20-inch pipe, then splits into various sub-headers for eventual distribution to all loads in a parallel arrangement. Any individual load can be isolated, some remotely from the central control room with motor-operated valves and others with manual valves. The recirculation pumps in containment have separate small booster pumps for their header called Auxiliary CCW pumps. After passing through the loads, the return flow is joined into a single 20-inch suction header for the CCW pumps. A single surge tank is also attached to this suction header. Pure makeup water for the system is supplied to the surge tank from the primary water system or the flash evaporator. The surge tank is normally vented to the PAB through a pneumatic valve, which automatically closes if there is high radiation in the CCW pumps suction header. Potassium chromate (175-225 gpm) is added to the surge tank to inhibit corrosion since most of the CCW components and the piping are made of carbon steel.

B.3 MAJOR COMPONENT DESCRIPTION

B.3.1 Pumps

The three main CCW pumps are horizontal, centrifugal pumps with a capacity of 3600 gpm and a design head of 220 feet TDH. They are driven by a 250 HP electric motor powered from 480 volt AC safety-related buses. Figure B-2 shows the piping and instrumentation arrangement for the pumps.

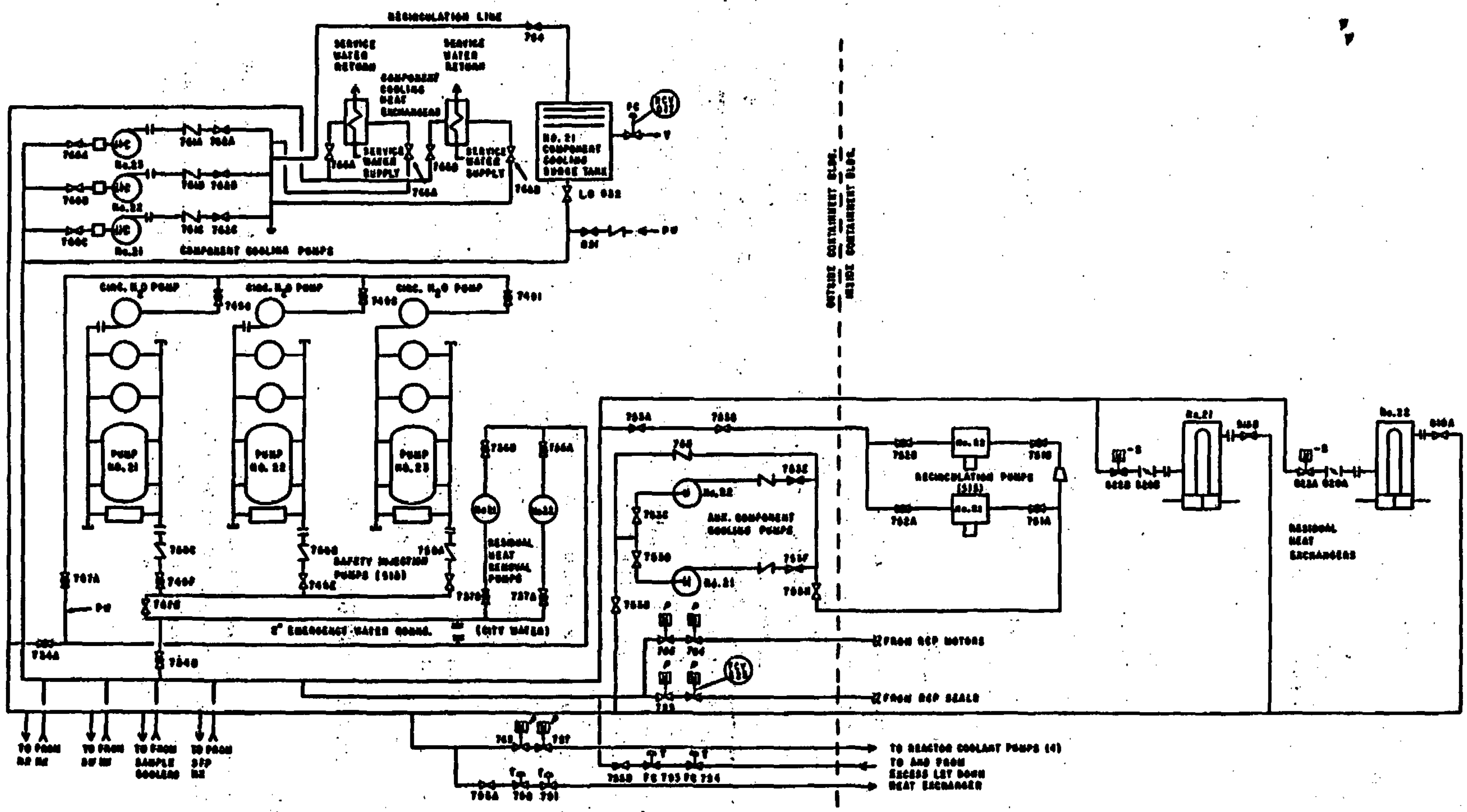


Figure B-1 Indian Point-2 CCW System Layout

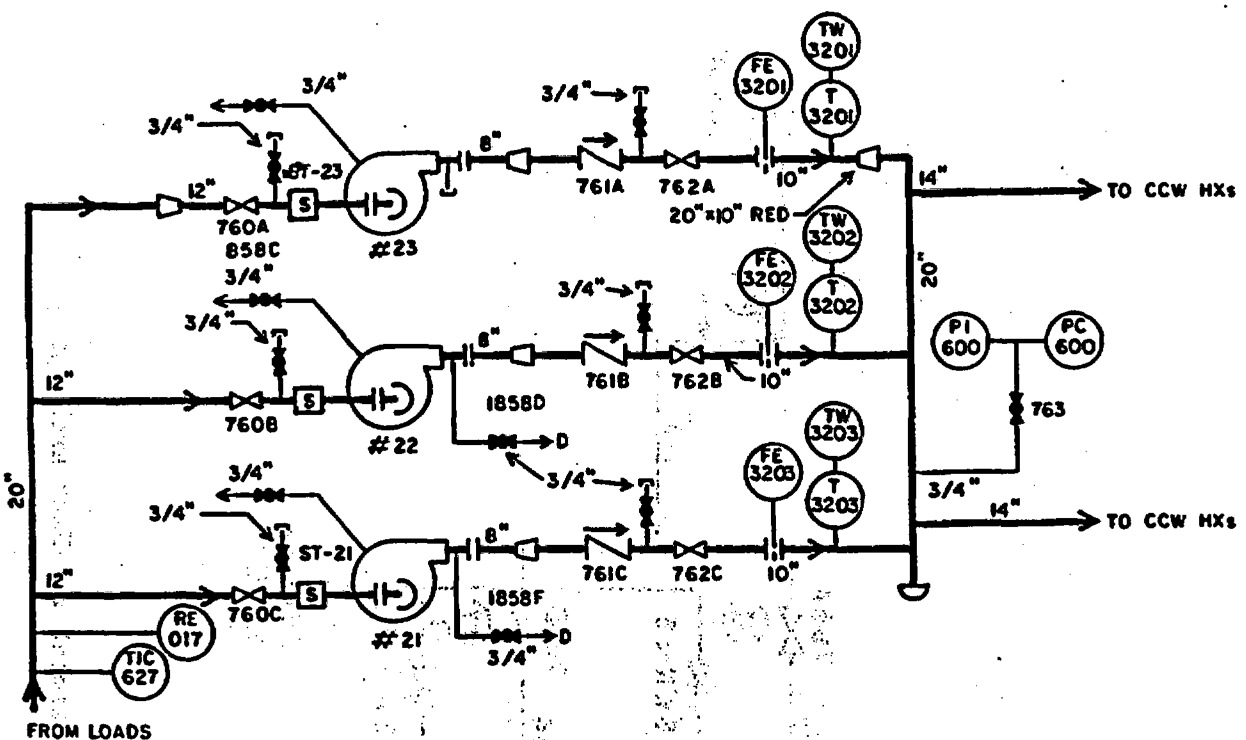


Figure B-2 IP-2 CCW Piping and Instrumentation Arrangement for Pumps

Each pump has suction and discharge maintenance isolation valves, a discharge check valve, a suction strainer, a casing vent, piping vent and drains, flow meters, and discharge temperature indicators. The common pump discharge header has a pressure indicator (PI), which reads locally, and a pressure controller (PC) that actuates a control room alarm and automatically starts the standby pump at 20 psi below the normal pressure of 100 psig. During normal operation, two of the three pumps are in operation. TIC 627 on the pump suction alarms on a high temperature of 155°F.

B.3.2 Heat Exchangers (HX)

The CCW heat exchangers are shell and tube type with CCW on the shell side and Service Water on the tube side. Figure B-3 illustrates the arrangement of the two CCW HXs.

During normal power operation both HXs are in service. Local temperature indicators are on the CCW inlet to each HX. The common CCW HX outlet header has flow and temperature indication and alarms for the control room. Normal CCW flow (with the residual heat removal loop normally isolated) is 6600 gpm and the low flow alarm is set at 1500 gpm. Normal CCW inlet temperature is 100°F and the outlet is 90°to 95°F, with a high outlet temperature alarm at 120°F.

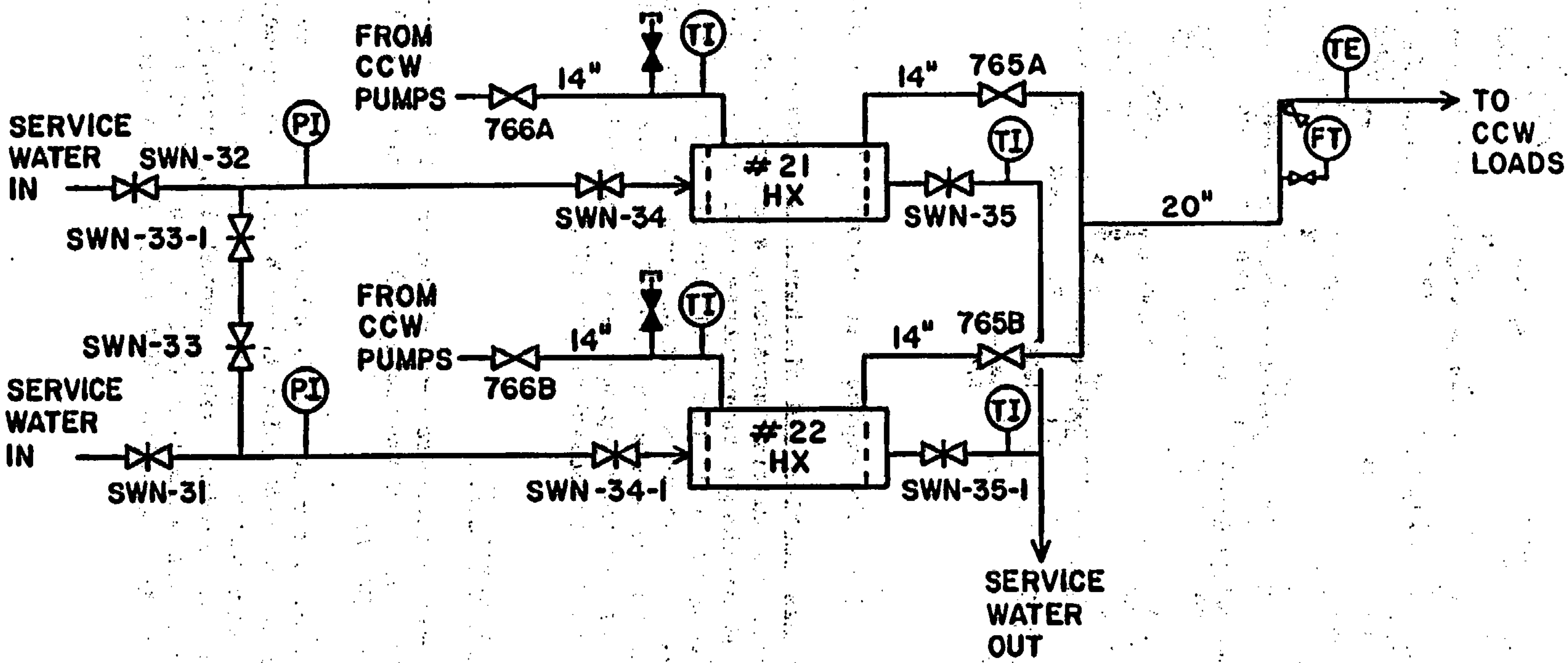


Figure B-3 IP-2 Piping and Instrumentation Arrangement for Heat Exchangers

Service water (SW) to the heat exchangers is normally cross-connected between the two loops, but with all flow coming through SWN-32. The HXs have inlet SW pressure indicators and outlet temperature indicators. Temperature of the CCW system water is controlled by manually throttling the two SW outlet valves SWN-35 and SWN-35-1. If it becomes necessary to adjust the temperature at individual CCW loads, this is done with the individual CCW throttle valves at each load.

B.3.3 Surge Tank

The surge tank is located in the PAB high above the CCW pumps to provide NPSH. Its total volume is 2000 gallons and normally operates at about 50% of capacity. Figure B-4 illustrates the IP-2 surge tank arrangement. A level transmitter provides high and low level alarms at 4 inches above or below the normal level. The surge tank has volume to accommodate in-leakage to CCW (e.g., from the RCP Cooling loop), out-leakage from CCW, or thermal expansion-/contraction. Normal makeup and chemical addition is via the surge tank. The tank is normally vented to atmosphere through pneumatic valve RCV-017, which closes automatically on high radiation sensed at the CCW pump suction. There is also a relief valve on top of the tank which relieves to the waste hold up tank.

B.3.4 Auxiliary CCW pumps

The two auxiliary CCW pumps are vertical centrifugal pumps arranged in parallel and are rated at 80 gpm and 100 feet TDH. They are automatically started on an ESF signal and supply component cooling water at a higher pressure and flow to the two recirculation pump motors inside containment. The recirculation pump motors are totally enclosed fan-cooled motors with cooling provided by CCW. Return flow from the recirculation pump motors goes through a single line with a flow meter and low flow alarm set at 60 gpm.

B.4 LOADS

The CCW system provides cooling to both safety and non-safety related loads. Table B-1 below lists the loads, their normal operating flow, and their maximum design flow (where applicable).

Table B-1 CCW Loads

Safety Loads	Normal Operating Flow	Design Flow
High Head Safety Injection Pumps (per pump)	15 gpm	--
Residual Heat Removal Pumps (per pump)	15 gpm	--
Spent Fuel Pit Heat Exchanger	2000 gpm	2850 gpm
Residual Heat Exchanger (per exch.)	~4000 gpm	10000 gpm
Charging Pump Oil Cooler (per pump):	90 gpm	--
Speed Controller	85 gpm	--
Bearing Cooler	5 gpm	--
Recirculation Pumps (per pump)	40 gpm	--

Table B-1 (Cont'd)

Safety Loads	Normal Operating Flow	Design Flow
Non-Safety Loads		
Flash Evaporated Product Cooler	400 gpm	--
Letdown Heat Exchanger (Non-Regenerative)	1000 gpm	--
Seal Water Heat Exchanger	200 gpm	220 gpm
Reactor Coolant Sample Heat Exchanger (per exch.) (2 in series)	14 gpm	40 gpm
Pressurizer Steam Sample Heat Exchanger (per exch.) (1 in series)	14 gpm	40 gpm
Pressurizer Liquid Sample Heat Exchanger (per exch.) (2 in series)	14 gpm	40 gpm
Steam Generator Blowdown Sample Heat Exchanger (per exchanger)	14 gpm	40 gpm
Boric Acid Evaporator Condenser (per evap.)	750 gpm	760 gpm
Boric Acid Evaporator Condensate Cooler (per evap.)	65 gpm	96 gpm
Boric Acid Evaporator Air Ejector Condenser (per evap.)	—	—
Gross Failed Fuel Detector Sample Cooler	14 gpm	--
Waste Gas Compressor Cooler	25 gpm	--
Reactor Coolant Pump Upper Motor Bearing Heat Exchanger (per pump)	150 gpm	--
Reactor Coolant Pump Lower Motor Bearing Heat Exchanger (per pump)	5 gpm	--
Reactor Coolant Pump Thermal Barrier Heat Exchanger (per pump)	25 gpm	--
Reactor Vessel Support Blocks Cooling Coils (combined)	50 gpm	--
Excess Letdown Heat Exchanger	230 gpm	245 gpm

Some features of important loads are discussed below.

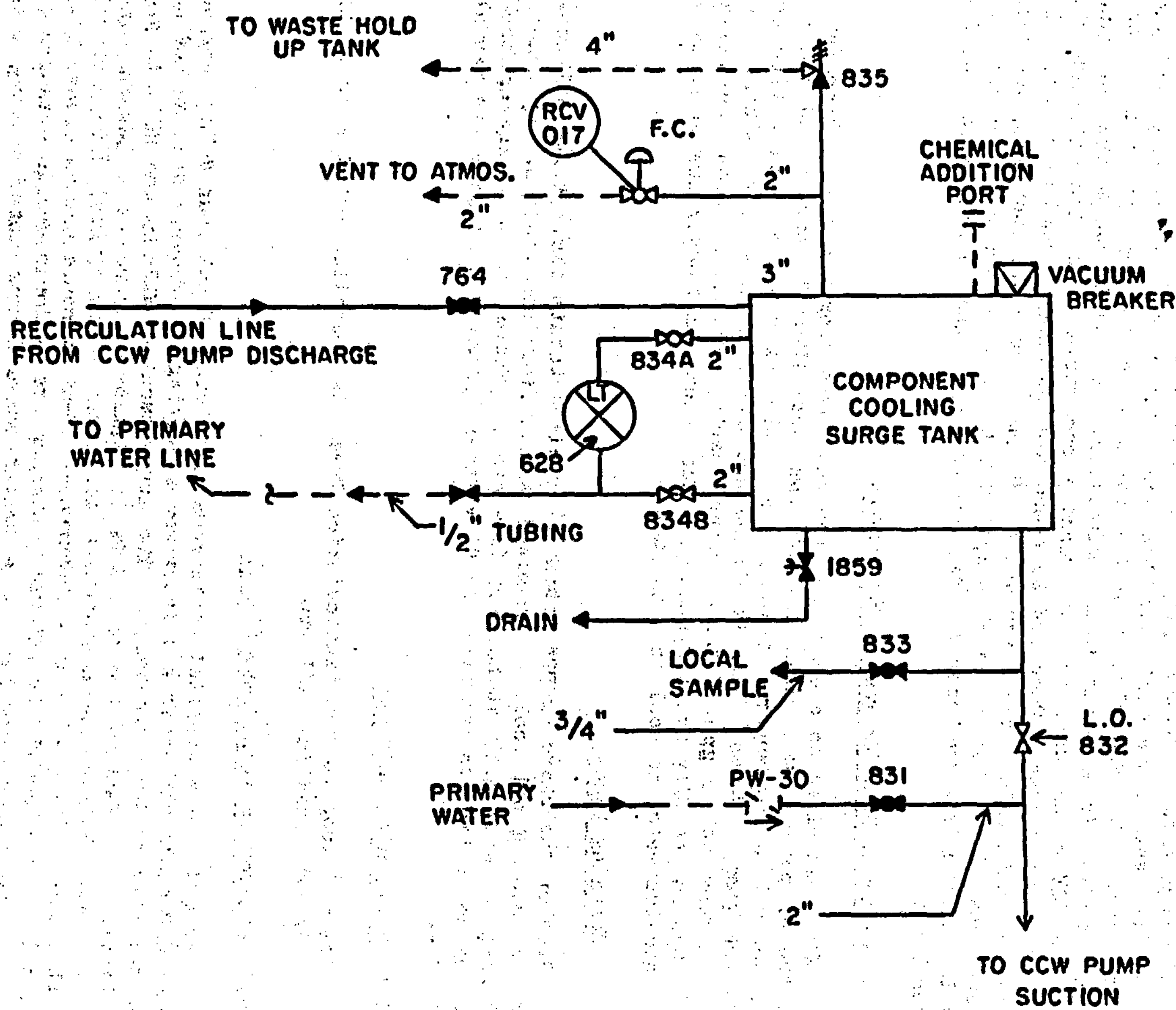


Figure B-4 IP-2 CCW Surge Tank Arrangement

B.4.1 Safety Injection Pumps

CCW flow to the High Head SI pumps is boosted by a small circulating pump directly connected to each SI pump motor shaft (Figure B-1). For each SI pump CCW is then sent to two seal water HXs, one oil cooler, and two pump seal jacket coolers. The CCW flow from the three SI pumps then joins and passes through a flow element with local indication. If the SI pumps are not running, the CCW circulators are not running and there is no flow through this portion of the system. When the SI pumps are running, normal circulator flow is 45 gpm, with a low flow alarm set at 30 gpm. There are two emergency backup supplies of cooling to the SI pumps' CCW header: (1) a manual valved connection to the Primary Water System, and (2) a flanged connection to the City Water System.

B.4.2 Residual Heat Removal (RHR) Pumps

CCW is supplied to each of the two RHR pumps from the same sub-header as for the SI pumps. CCW cools the RHR pump seal thermal barrier and the pump seal water heat exchanger. A local flow indicator measures outlet flow from each RHR pump. Normal flow is 15 gpm and there is a low flow alarm in the control room at 12 gpm. The emergency backup connection from the Primary Water System and City Water System previously mentioned under the SI pumps can also provide water to cool the RHR pumps.

B.4.3 Charging Pumps

Each of the charging pumps has CCW cooling to its fluid drive oil cooler and its bearing oil cooler. Combined outlet flow from the charging pumps is measured by a locally indicating flowmeter. There is an emergency backup cooling connection to and from the City Water System. All valves are manually operated.

B.4.4 Recirculation Pumps

The CCW loop to the Recirculation Pumps is described earlier in Section B.3.4 with the Auxiliary CCW pumps.

B.4.5 Reactor Coolant Pumps (RCPs)

The CCW system supplies cooling water to each of the four RCPs. For each pump, CCW cools the upper and lower motor bearing oil coolers and the RCP seal thermal barrier heat exchanger. The arrangement of CCW to the RCPs is shown in Figure B-5. Each of the four RCPs is similar so only one is shown. Inlet water comes through two containment isolation valves (CIVs) #769 and #797, which close automatically on a containment hi-hi pressure signal (Phase B isolation). The lines to the RCP motor bearing are of lower design pressure than that to the thermal barrier. The relief valve in the motor bearing return line is set at 150 psi and in the thermal barrier line at 2485 psi due to the potential for a thermal barrier cooler failure allowing reactor coolant pressure into that line. Normal flow to the RCP upper bearing is 150 gpm and to the lower bearing 5 gpm. Combined RCP bearing return flow has a low flow alarm at 125 gpm. The bearing water then exits containment and passes through

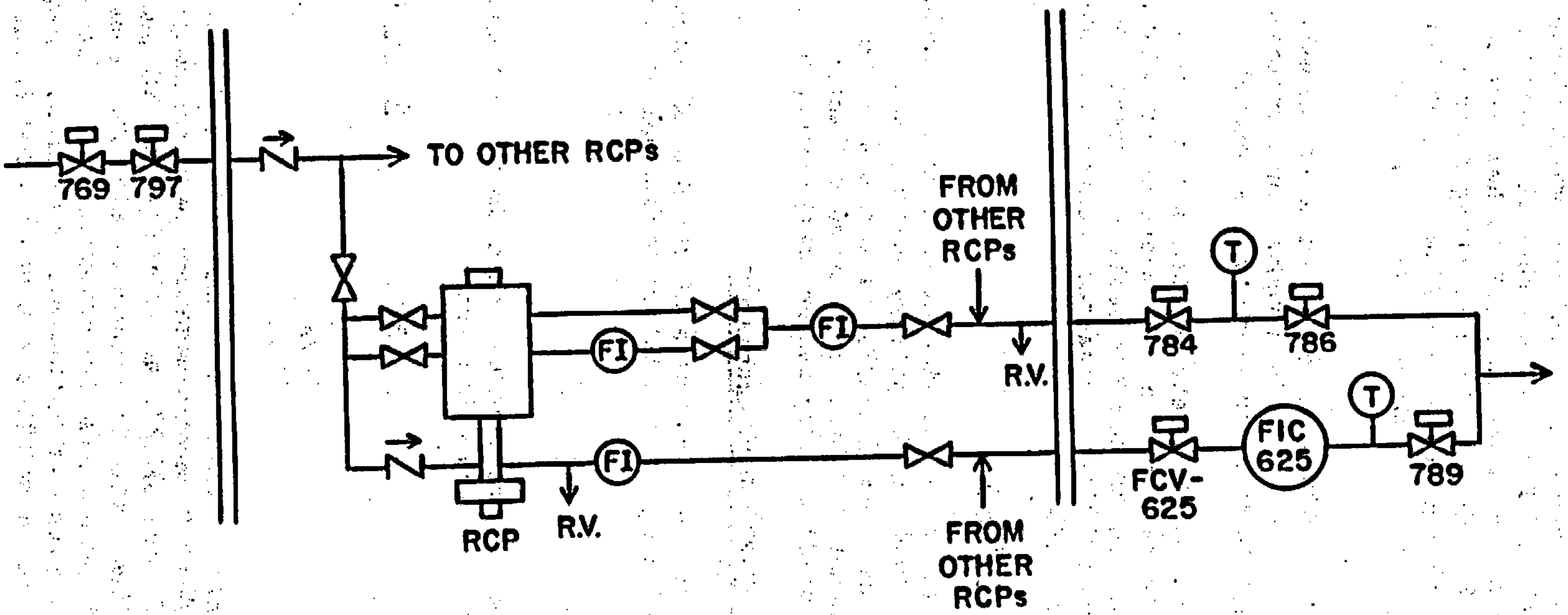


Figure B-5 CCW Supply Arrangement to RCP's two other CIVs (#784 and 786), which also close on a Phase B isolation signal. Between these CIVs is a temperature indicator which alarms in the control room at 120°F. The normal CCW flow to each thermal barrier is 25 gpm. After the thermal barrier return flow from each RCP is headered together it exits containment and passes through the flow indicating controller, FIC-625. This has a low flow alarm at 80 gpm and a high flow alarm at 120 gpm. At 120 gpm a thermal barrier rupture is indicated and valve FCV-625 is automatically closed.

Valves FCV-625 and -789 also close on a containment Phase B isolation signal. Between these two valves is a temperature indicator. Normal temperature here is 120°F and there is an alarm in the control room at 140°F.

Although not considered safety-related and not necessary post-accident, CCW flow to the RCP thermal barrier is important in maintaining the integrity of the seal. Loss of CCW to the RCP seals can result in seal failure and a resultant primary system LOCA. Procedures require that RCPs be stopped within two minutes of loss of CCW.

B.4.6 Residual Heat Exchanger

CCW is provided to the shell side of each of two Residual Heat Exchangers (RHXs) which are located inside primary containment. Normally there is no flow to the RHXs since MOVs (822A/B) on the CCW outlet lines are normally shut. These two MOVs automatically open on an ESF actuation. Since this is a closed, seismically qualified loop inside containment, no CIVs are provided.

B.4.7 Spent Fuel Pit (SFP) Heat Exchanger

CCW is supplied to the shell side of the SFP HX which is located in the Spent Fuel Storage Building. The CCW return line has both a flow and temperature indicator.

B.5 ELECTRICAL AND I&C

The CCW pump motors are squirrel-cage induction motors, supplied with safety-related 480 Volt AC power from buses:

Pump #	Bus #	Diesel Generator
21	5A	21
22	2A	22
23	3A	22

An important design item is that pumps 22 and 23 are supplied from 480 volt buses (2A and 3A) that receive emergency power from the same diesel generator (DG). Thus failure of DG #22 will fail two CCW pumps. Also the IP-2 Technical Specifications call for two operable CCW pumps from a different power supply. If two pumps from different buses are not operable, then only 24 hours of continued plant operation are permitted. Hence, the plant may operate indefinitely with pump 22 or 23 out of service, but only 24 hours with pump 21 out of service.

Each pump control switch in the control room has four positions: Pull-out, Stop, Auto, and Start. The ACCW pumps are supplied with 480 Volt AC power:

Pump #	MCC #
21	26A
22	26B

These pumps are operated locally at the MCC or automatically started on an ESF actuation. Motor-operated valves for the CCW system are powered by 480 Volt AC from MCCs, MCC-26A and -26B.

The CCW system has a considerable amount of local and remote instrumentation in the control room. The indication and alarms are summarized on Figure A-2 of the previous Appendix. The details of the location of each instrument are given on Con Edison drawings A227781 and 9321-F2720. The automatic controls for the CCW systems are also summarized on Figure A-2 and are described in more detail in the sections above that discuss the components affected.

B.6 VALVES

Figure B-6 illustrates the breakdown between the various types of valves and shows the total number of each type in the CCW system at Indian Point-2. The majority of valves are manually operated. The motor-operated valves (MOVs) are used for containment isolation and startup/shutdown of the Residual Heat Exchanger loop. The air-operated valves are used for containment isolation and temperature control of the non-regenerative HX loop. Each load capable of being isolated and generating substantial heat, has a relief valve. Each of the CCW pumps and some loads have check valves. Each component cooled by CCW has an individual throttle valve, set in accordance with IP-2 procedures. The position of these valves is generally constant. If a component (load) is not in use, CCW flow still is usually supplied to the component. If it is necessary to isolate a component (load), it is done with the manual stop valves and not the throttle valves.

B.7 SYSTEM OPERATION

The entire CCW system is seismic Class 1 and is housed and supported in seismic structures. The piping and components generally are all carbon steel with welded connections, except where flanged connections are used to facilitate maintenance.

During normal plant operation the CCW system is in continuous operation. The normal configuration is two CCW pumps and one CCW HX operating. Two HXs may be used at any time, but the third pump may only be used if there are sufficient loads to allow adequate flow (that is, both RHXs and both CCW HXs must be on line). The CCW temperature out of the CCW HXs to the loads should be between 70°-100°F, except that during the first three hours of RHR operation temperature is permitted to vary from 70°-120°F. The maximum temperature allowable after the loads at the CCW pump suction is 160°F. Flow for each CCW pump must be kept between 300-3600 gpm.

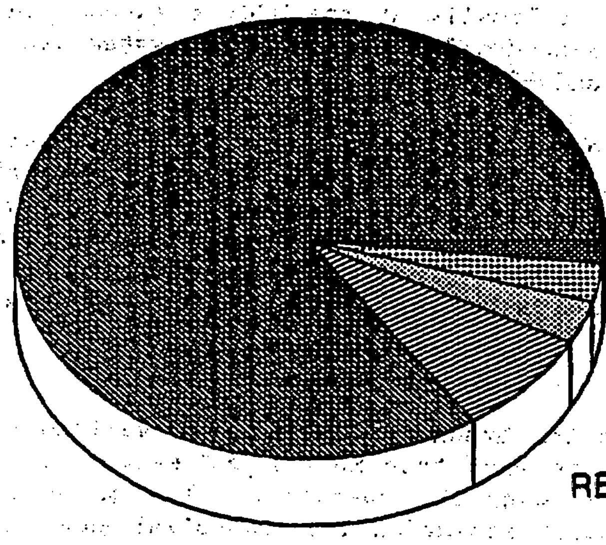


Figure B-6 IP-2 CCW Valve Types

Procedures provide for an emergency cooling-water supply to the Safety Injection Pumps and RHR Pumps in the event of loss of CCW. This can be provided manually by the operators from either the Primary Water System or the city water header. Additionally, in an emergency, the city water header can provide cooling to the Charging Pumps in place of CCW; this is also accomplished manually by the operators.

B.8 MAINTENANCE PROGRAM

Corrective maintenance (CM) was performed on various components of the CCW system over the years to repair failures that occurred. This maintenance was documented on Work Orders and Maintenance Work Requests. Recent maintenance (1984 to 1987) is also included in a computerized tracking system. The plant is currently involved in a backfit effort to enter past data into the same computerized system.

Preventive maintenance (PM) performance varied considerably over the years. Before 1981 there are no records on what PM was performed: since 1981, the PM is summarized by component. The CCW pumps were overhauled for PM each four years per procedure M.P.-4.1, which resulted in an extensive refurbishment of the pump. If extensive CM was performed on a pump then credit was taken for that in the PM program. The CCW pump motors also were subjected to PM once in four years. The pumps/motor bearings were greased every 3 months and had their oil and grease changed yearly. At the oil change, a sample of the oil was analyzed. Oil levels were checked daily. Motor-operated valves were inspected each refueling outage (approximately 18-24 month intervals). The circuit breakers of the CCW pump motor were also maintained each refueling outage. No PM was performed on manual valves, heat exchangers, or piping.

B.9 INSTRUMENTATION AND CONTROL (I&C) CALIBRATION

As with most plant systems, the CCW system contains a considerable amount of instrumentation and control circuitry, most of which is monitoring equipment. The automatic control circuits, such as initiation and containment isolation, were checked and calibrated if necessary, as part of the technical specification testing conducted generally at refueling intervals, in accordance with detailed procedures. The other instrumentation was calibrated at 2-3 year intervals as follows:

Flow Instruments - I&C Group
Temperature Indicators - Test Performance Group
Temperature Control Valves - I&C Group
Level Indicators - I&C Group
Pressure Indicators - Test Performance Group
Pressure Switches - I&C Group
Valve Limit Switches - By Maintenance

The instrumentation calibrated by the I&C group was generally calibrated using data sheets without a procedure specific to the instrument. The equipment calibrated by the Test Performance Group and Maintenance required specific procedures.

B.10 TEST PROGRAM

The CCW system components were tested over the years as described below.

<u>Auxiliary CCW Pumps:</u>

From 1973 to 1984, the auxiliary CCW pumps were tested per procedure PTM-20, in accordance with Technical Specification 4.II.A.1. This test started the pump, measured pump head, and ran the pump for 15 minutes each month. From 1984 to the present, the testing was increased to meet the ASME Code, Section XI, Subsection IWP test requirements, by procedure PT-Q31. This testing recorded and documented any trends in pump vibration, pump head and pump bearing temperature. The test was performed quarterly unless problems required increasing the frequency to monthly. The bearing temperatures were only measured annually.

Main CCW Pumps

From 1981 through 1984 these pumps were tested monthly for operability by procedure PTM-43. From 1984 to the present they were tested quarterly by procedure PT-Q30 to the requirements of the ASME Code, Section XI.

Pump Motor Circuit Breakers

From 1979 to the present they were tested each refueling outage by procedure PT-R46 to verify operability and the setting of the overcurrent trips.

Radiation Monitors

The radiation monitor that initiates closure of the surge tank vent valve was tested semi-annually since 1973 per procedure PT-M10A.

Piping

Piping must be tested once every 10 years per the ASME Section XI. This was done preoperationally and then inservice for the first time in 1984, with an Inservice Pressure Test, by procedure PT-104-L.

Valves

The air-operated valves used for containment isolation were tested quarterly for operability and stroke time quarterly per procedure PT-Q13 and PT-R35. The AOVs and MOVs used for containment isolation were leak-rate tested every refueling cycle since 1973.

APPENDIX C

DETAILED DESCRIPTION OF THE PRAAGE CODE

CONTENTS

FIGURES

C.1 OVERVIEW

Initial ground rules for the PRAAGE code were:

* Simplicity for use by any engineer familiar with the plant but without computer training.

* Provide appropriate preassigned values to minimize the amount of data input to new information.

* Provide measures of component importance familiar to the engineer such as unavailability and contribution to unavailability.

* Provide automatic rank ordering of components by their importances.

* Print and Plot the results.

* Prepare the code within a limited budget in 2 months or less.

* Operable on low-performance IBM-PC such as might be expected at plants or regional offices of the NRC.

These considerations indicated a computer language with graphics. A compiled program was desirable for ease of running and code protection which lead to the selection of Turbo Pascal (Borland International) with the Graphix Toolbox. This is a good implementation but it is limit to compiling programs < 60k bytes. This resulted in the necessity of chaining programs. (Actually they are not chain but execute files to allowing breaking into the chain).

Figure C-1 presents a block diagram of PRAAGE's chained structure. The first program serves as an introduction with the title and a synopsis. There are three types of graphics adapters on the IMP-PC: Color Graphics, Enhanced Graphics and Hercules. Programs prepared for one either will not work on the other or the graphics will be distorted. Therefore, PRAAGE is a available for each type of adapter.

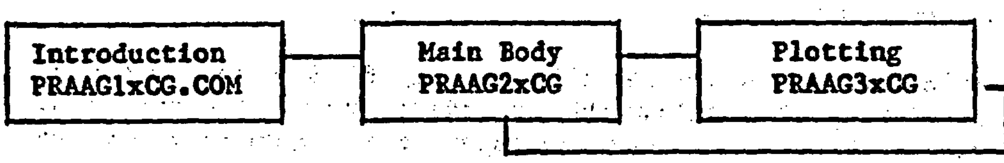


Figure C-1 Block Diagram of PRAAGE Chaining

PRAAGE is started by getting the drive prompt, inserting the disk and typing PRAAG1xx.COM where xx is CG, EG or HR depending on the respective type of adapter on the machine. The program chain then starts at the Introduction Program. When the operator is familiar with this material, it may be bypassed by beginning at the Main Body Program by typing PRAAG2xx.COM.

This step performs the computations and provides tabular output to the screen or to paper copy but it requires the third program to graph the data. (Note: it is not possible to start at the Plotting Program because data must be transferred from the second program). After the graphing is completed, it returns to the Main Program for recalculations or for introducing new data into the calculation. The time for calculations is about 4 seconds.

C.2 PRAAG1xx.COM - THE INTRODUCTORY CODE

This program provides the program title (for 3 seconds) and then presents a synopsis which is left by pressing any key.

The graphics have been changed several times. They were begun in BASIC at high resolution, converted to low resolution in color using PASCAL and finally implemented using the Graphix Toolbox to achieve compatibility with the various types of color adapters.

The program begins with the main program calling the procedure (PASCAL terminology for a subroutine) BIGNAME. This prints "Brookhaven National Laboratories presents:" in standard-sized characters and then begins printing "PRAAGE" in large characters using the graphics. Transportability is achieved by defining world coordinates as 0, 2, 640, 200 to convert from the original absolute coordinates used originally for the CGA. In plotting the characters, the straight lines are drawn as lines between endpoints but the circles cause a problem.

Turbo PASCAL does not draw segments of circles at arbitrary angles. While this may be done with the Graphix Toolbox, the previously written procedure ARCS was used because of its easy adaptability to the final coding. ARCS divides the specified arc into 20 segments and draws as segmented circle using an aspect ratio of 2.33 to correct for the rectangular IBM screen. The letters are not filled because of a word conflict between PASCAL and GRAPHIX. A timer is used to hold the name for 3 seconds and then the procedure SYNOP is called to provide a brief description of the code. Pressing a key results in a call to load the main program PRAAG2xx.COM.

C.3 PRAAG2xx.COM - THE MAIN PROGRAM

This is the most complex although not the largest of the three programs. Figure C-2, the logic flows, shows that the program performs 3 operations: data input (modification of default data), calculating the system model and presentation of the results.

The main program begins according to PASCAL convention by the declaration of variables in the order: label, constant, type, variable, procedures and functions. This places the main program at the end, in which is contained the default data for the aging rates, aging start times, component identifications, generic identifications and generic failure rates. Following this, the procedure MAIN is called.

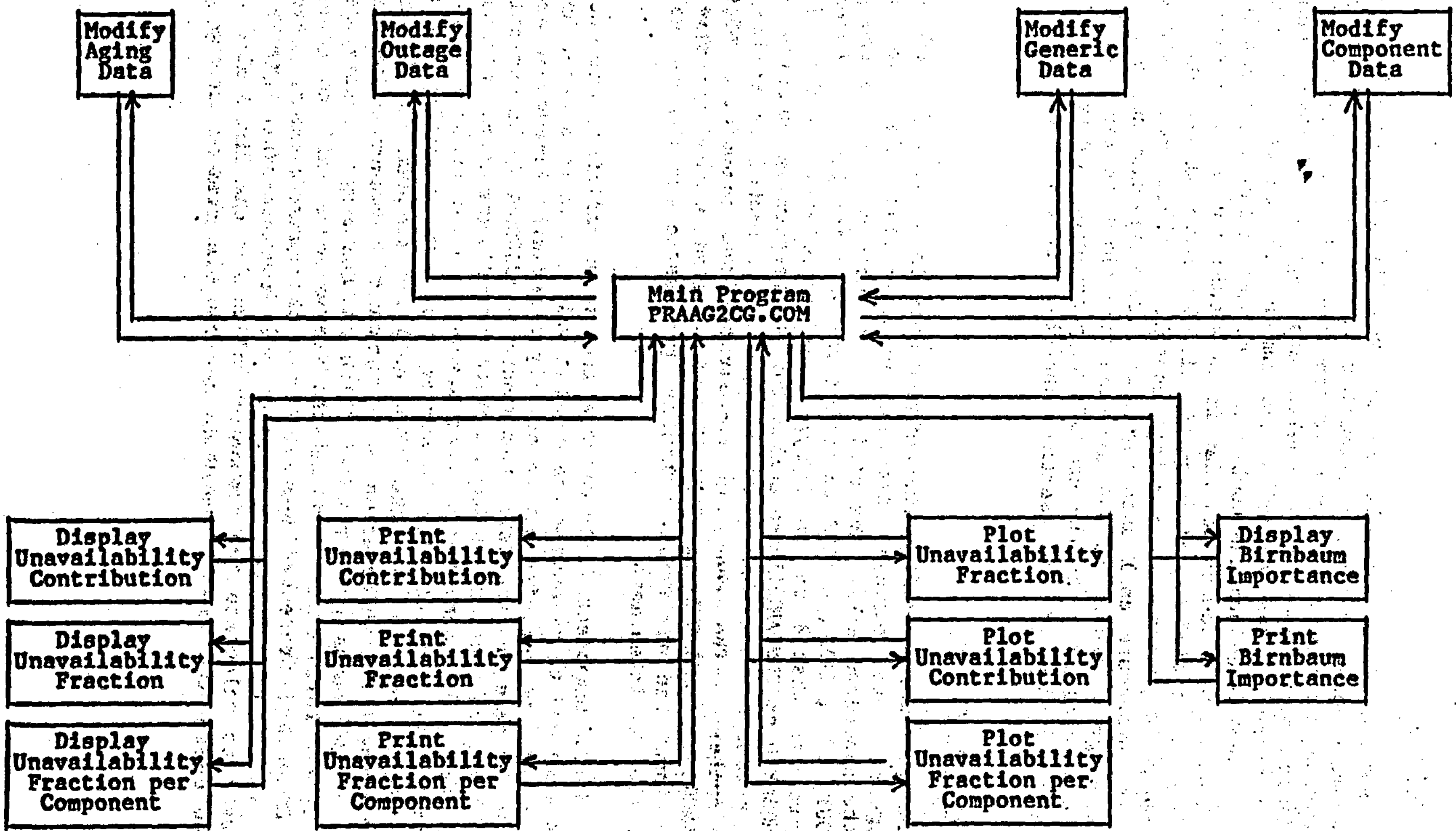


Figure C-2 Logic Flows in PRAAG2CG.COM

C.3.1 Procedure Main

This defines the window and color and presents the selections that can be made from the main menu. Only one selection may be made at a time by the operator inputting the number of the selection followed by a carriage return. (This is necessary because some of the selections require 2 digits. Complete selection is indicated by the "enter"). The number selected is read in by the procedure INTCK, which accepts the number as a string variable and examines it to determine that it is a number and that it is within a prescribed range. If neither of these are satisfied, it responds with an error statement. If the entry is accepted it passes to a parsing sequence of if-then-else statements to determine the procedure call associated with the selected number. After the call, the program returns to the location from which it was called and a GOTO statement is used to go to the end of the procedure. (This use of GOTO could be avoided by the use of reentry, whereby procedures may call themselves).

C.3.2 Data Preparation

The technique used to modify the default data is similar for the aging, restoration,generic and component data and is essentially the same as described for the procedure MAIN. The call from the main menu is to the procedures AGE, OUT, LMDA, LMDB, or GEN according to the selection. As before the selected number is tested to avoid typos and procedures AGEMENU, EXMENU, GENRATE, ALMDAMENU or BLMDAMENU are called for the respective function. These present the parameters for modification which are selected by typing their identifying number. The input is typo-tested and procedures AGECHANG, EXCHANG, GENCHANG, ALMDACNG or BLMDACNG are called and the operator is requested to input the new value. This is typo-tested and the new value is presented in the rewritten menu.

If individual component failure rates are to be modified, the mechanics are much the same and these change the default values. In this case, however, the default values are not directly entered by a default library but are constructed from the generic data using the procedure CALCULA. The variables are the failure rate data that are shown in the Component Parameters Menu but they are not necessarily the actual data that are used in the calculations. Most of these numbers are in units of frequency and must be multiplied by the restoration time. Because some the IPPSS data are frequencies, it would be confusing to present the information as the product of frequency and restoration time. An additional problem exists with the CCW pumps, in that the data are of two types: failure to start and the frequency of failure to continue running. These are presented separately in the generic data. Procedure CALCULA is called from the LMDA procedure and from procedure IMPORT. The former is used to calculate the component values for the first time and the latter is to update the values according to possible changes in the generic data.

C.3.3 Calculating Aging Importances and Unavailability

After the data are suitable, one of two importances is selected resulting in calls to procedure IMPORT which calls CALCULA, MAKECOEF, MAKEAA, MAT and IMPRT. MAKECOEF uses the default values of the modified values to construct generic time dependent variables such as pmpcoef meaning pump coefficient. The general form is:

```
If (TAGE[n] <= xxfr then xxcoef = H1
 xxcoef = H1*(1+offon*TAGE[n]-xxfr)*xxDUB)
```

The upper line consists of a test such that aging is turned off for time (TAGE[n]) less than the aging threshold (xxfr) for generic component xx. If time is greater than the threshold then the lower equation is used in which Hxx is the restoration time and the terms in the outer parenthesis provide the linear aging at a fractional increase per year of xxDUB. The term offon is used as a switch to turn off aging for certain reliability studies.

The IPPSS-2 CCW model results in first, second and third order cutsets. It can be shown that the importances may be calculated by treating these as matrices. These were identified in Section 6.2 as the A through F matrices. The inspection importance of a component in the first order cutsets consists of the probability of the corresponding element of the A matrix. The importance of elements in the second order cutsets are the probability elements of the B matrix multiplied by the sum of the probability elements in the C matrix and vice versa. The importance of elements in the third order cutsets are the probability elements of the D matrix multiplied by the sum of the probability elements in the E matrix multiplied by the sum of the probability elements in the F matrix and for all combinations. The summation operations just described are performed in the procedure MAT and these permutations in the procedure IMPRT. IMPRT ends by adding up all the importances and dividing the individual values by the total. The result of this is that if all of the importances are added together, the total is "1". It should be noted that this is not the same a Vesely- Fussel importance since that would be normalized by the unavailability that would be calculated by probabilistic addition. Normalized Inspection Importance has the very practical property of representing the fractional contribution of each component to the unavailability and was devised for this purpose as part of this study.

Procedure IMPRT continues and calculates the generic unavailability contribution, the fractional generic unavailability contribution and the fractional generic unavailability contribution per component. This last measure was added to address the concern that a generic component may be important if there are a very large number of components that individually are not very important. For example the CCW system has 3 pumps and 20 valves. The importance of the two generic classes are about the same hence the individual valve importance is about 15% of a pump's importance. These numbers are expressed in percent to aid in perception.

The Birnbaum Importance (same as Risk Reduction Worth Increment) is calculated by the same procedures as Inspection Importance except that when it is selected from the main menu, an identifying integer, aa[n] is transferred to procedure MAT causing it to set the component failure probabilities of the component for which the importance is being calculated to "1" thus exhibiting the failure probabilities of component trains remaining after the component of interest has failed. It may be noticed that the Birnbaum Importances of elements in the first order cutsets are "1".

C.3.4 Displaying Results: Unreliability Contribution, Unreliability Percent, Unreliability Percent per Component and Component Importances.

After the importance calculations have been performed, the results must be displayed. To do this control returns to the Main Menu from which four different ways of presenting the results are available. From the Main Menu, procedures BUGDSPLY, GENIMPDSPLY, AVGGENDSP or COMPIMPDSPLY, respectively, may be called. If importances have not been calculated previously, an error message will result. To improve the utility of the results, they are ordered by descending importance by the procedure SORT. Procedure SORT is a bubblesort routine that does not actually sort the data but orders the data-identifying indices according to the sorting of the data. Procedure SORT calls procedure SORTYEAR to ask the operator the year to use in sorting the data. This is because some components age faster than others consequently the ordering may change with time. With this information, SORT calls procedure SWAP which does the actual sorting of indices.

C.3.5 Printing Results: Unreliability Contribution,Unreliability Percent, Unreliability Percent per Component and Component Importances.

Printing is performed by an adaptation of the display menus. To avoid adding extra sections to the main menu and assuming an operator will want to see the output before printing it, an identifier of the previous display menu is transferred to the print procedures: BUGPRT, GENIMPPERT, AVGGEMPRT and COMPIMPPRT and the tables are printed in a format adapted to the printed page.

C.3.6 PRAAG3xx.COM - Plotting Results

Selection 12 on the Main Menu plots 6 or less parameters from the generic menus. If this is selected, the program performs an execute command to call in PRAAG3xx.COM, the graphing program. For the graphs to be drawn the information that was calculated in PRAAG3xx.COM must be transferred.This information is: the display menu identifier, identifiers of the components and the table of the ordered importances.

The program sets up for log-log plotting of the data by taking logarithms of the plotting times. In order to perform cubic-spline fitting of the data, it is necessary to extrapolated end points. The ones chosen are 0.1 and 50 years.

This is followed by calling procedure BGNPLOTSEL to present a menu appearing like the previously displayed results with the importances omitted. Six or less components are selected for plotting their importances. The scaling of the graph is done automatically, so the greater the dispersion in importances, the less resolution in the graph. The selections are collected by the procedure GETNOS which makes use of a sorting table to make the proper association between identifiers and identified numbers.

The program enters a "while" loop to set up for each selected component. A "while" loop is used rather than a "do" loop because of problems encountered if only one component importance is selected for plotting. This loop calls procedure PREPA which also has to perform the index order decoding. It uses the display identification to associate the proper importances and calls procedure LOG to obtain base 10 logarithms of the importances. Next procedure MINMAX is called to determine the maximum and minimum values of the data and procedure INTERP is called to perform the linearly extrapolated data for the years 0.1 and 50.

Before calling the plotting procedure, function YSCALE is called to determine the maximum and minimum powers of 10 providing margins beyond the data. Then procedure SETUPWNDO is called.

This procedure begins by defining the plotting symbols to be used, the window, the world coordinates and the window title. The identification of the type of data being plotted, the generic components, line style and identification symbols are provided in the upper left corner of the graph. It also includes the instruction to type "P" to get a hardcopy of the screen. It then draws an outline of the window, provides "tick marks" at the decade and 2 and 5 values on the ordinate as well as labeling the abscissa. It goes into double "do" loops to plot the data points and calls SPLINE to obtain 50 interpolated values for the smooth curves in the different line styles. Upon completion of the plotting, the display is held for viewing by a read(kbd,x) command awaiting a "P" if hardcopy is required. If any other key is pressed, it returns to the main program from which PRAAG2xx.COM is called.

Normal exiting from PRAAGE is performed by selecting "13" from the main menu.

C.4 Using PRAAGE

This discussion is an overview of how to use the PRAAGE code. Using the input parameters, PRAAGE:

- Calculates the system unavailability,
- Orders the components according to their importance to the system for various ages,
- Provides numerical measures of their contribution to the system unavailability as they age,
- Presents the results of the analyses as tables and graphs.

It performs this work in three major tasks:

a) Data Presentation and Setup
b) System Model Calculation
c) Presentation of Results

These groups are selected from the Main Menu to which the program returns after it completes a task. The Main Menu has 13 selections (Figure C-3). Selections 1 to 4 pertain to data presentation and setup, system calculations are performed by steps 5 or 6, results are presented by selecting one of steps 9 through 12 and normal termination of the program takes place through selection 13.

a) Data Presentation

Data may be reviewed and modified by selections 1 through 4 from the main menu. Each of these selections will bring up the appropriate menu for generic component aging, generic component restoration times and generic and component specific failure rates and probabilities. This is the information needed to calculate the system model.

Menu #1 (Figure C-4) presents for review and modification the annual fractional increase in the failure rate for each generic component, i.e. the linear increase from the constant failure rate. PRAAGE does this on a generic basis because it is assumed that aging information specific to a single component is not available - only for classes of components. This menu also presents the time at which the aging starts for this generic component type.

Menu #2 (Figure C-5) presents times for operation of generic components to calculate the probability that a failed component will be out of service for a critical period. This is part of the success criteria of the PRA for the CCW system, that is, after an accident condition, the CCW system must operate for 24 hours to be successful. IPPSS and PRAAGE assumed this to be 24 hours.

Menu #3 (Figure C-6) presents the age-independent generic failure rates and probabilities. If information on individual components is not available, it is convenient to set the values by generic groups. These are the λ_0 values for the generic components.

Menu #4 (Figures C-7a & b) presents the individual default failure rates and probabilities. It also identifies the type and location of the component, its failure mode and provides the identification used in the IPPSS. Using this menu, individual component failure rates can be modified or components can be failed.

b) System Calculation

Once the data are specified, the calculation of the system model may be made. The system model is introduced into the PRAAGE model as "cutsets", i.e. those components whose coincident failure will cause the system to fail. These cutsets provide a mathematical formulation of the CCW system and cannot be changed from the menus.

MAIN PRAAGE MENU FOR AGING ANALYSIS

No.	Task
1	Modify Aging Parameters
2	Modify Restoration Times
3	Modify Generic Failure Rates
4	Modify Individual Component Failure Rates
5	Calculate Normalized Inspection Aging Importance
6	Calculate Birnbaum Aging Importance
7	Display Generic Aging Budget
8	Display Generic Aging Importance
9	Display Average Generic Aging Importance
10	Display Individual Component Aging Importance
11	Print Last Display
12	Present Graph of Generic Importances and Unavailability
13	Quit

Select Task Number and Return

Figure C-3 Main PRAAGE Menu - Aging Analysis Tasks

#1 MENU OF AGING PARAMETERS

No.	Parameter	Value
	Analysis Time of this Aging Study	40.0 yrs.
1	Manual Valve Aging Fract. Increase/Yr	0.21
2	Manual Valve Aging Start Time in Yr	4.70
3	Check Valve Aging Fract. Increase/Yr	0.02
4	Check Valve Aging Start Time in Yr.	2.00
5	Pipe and Tank Aging Fract. Increase/Yr	0.00
6	Pipe and Surge Tank Aging Start Time in Yr.	2.50
7	Heat Exchanger Aging Fract. Increase/Yr	0.02
8	Heat Exchanger Aging Start Time in Yr	2.00
9	CC Pumps Aging Fract. Increase/Yr	0.28
10	CC Pumps Aging Start Time in Yr	9.20
11	Service Water Aging Fract. Increase/Yr	0.00
12	Service Water Aging Start Time in Yr	10.00
13	Switchgear Aging Fract. Increase/Yr	0.00
14	Switchgear Aging Start Time in Yr	10.00
15	Turn Off (0)/On(1) Aging	1

If you want to change a parameter, type its parameter number and press RETURN. Otherwise, enter something else.

Figure C-4 PRAAGE Menu 1 - Aging Parameters

#2 RESTORATION TIME MENU

No.	Identifier	Restoration Time
1	Valves	24.0 h
2	Pipe and Surge Tank	24.0 h
3	Heat Exchanger	24.0 h
4	CC Pump	24.0 h

If you accept these, type the parameter number and return. Otherwise, enter something else.

Figure C-5 PRAAGE Menu 2 - Restoration Time

#3 MENU OF GENERIC FAILURE RATES

No.	Generic Component	Value
1	Pipe and Surge Tank	8.6E-10/hr
2	Service Water	2.5E-04
3	Electrical Bus	7.1E-05
4	Valve: Manual CCW	2.2E-07/hr
5	Valve: Manual SW	2.2E-07/hr
6	Valve: Check	4.3E-06/hr
7	CC Pump: Failure to Run	9.1E-05/hr
8	CC Pump: Failure to Start	6.4E-03
9	Heat Exchanger: Leak or Rupture	3.8E-05/hr
10	Pump Maintenance Unavail. Contribution	8.5E-02
11	Common Cause Unavail. Contribution	0.0E+00

If you want to change these, type the parameter number and return. Otherwise, enter something else.

Figure C-6 PRAAGE Menu 3 - Generic Failure Rates

#4 MENU 1 OF FAILURE RATES

No.	Component	Failure Rate
1	UXV734AC vlv 734A fc	2.2E-07/h
2	UTK0021L CC Surge Tank 1/r	8.6E-10/h
3	UXV734BC vlv 734B fc	2.2E-07/h
4	UPPFAILS CC Pipe Failure	1.5E-08/h
5	TSWINOFL SW Failure	2.5E-04
6	TXV32--C SW iso vlv 32 fc	2.2E-07/h
7	TXV35-1C HE 22 SW out vlv fc	2.2E-07/h
8	UXV766BC HE 22 CCW in vlv fc	2.2E-07/h
9	UHE0022L CC HE 22 1/r	3.8E-05/h
10	UXV765BC HE 22 CCW out vlv fc	2.2E-07/h
11	TXV34-1C SW HE 22 out vlv fc	2.2E-07/h
12	TXV33--C SW iso vlv 33 fc	2.2E-07/h
13	TXV31-1C SW iso vlv 31-1 fc	2.2E-07/h
14	TXV35--C HE 21 SW out vlv fc	2.2E-07/h
15	UXV766AC HE 21 CCW in vlv fc	2.2E-07/h
16	UHE0021L CC HE 21, 1/r	3.8E-05/h
17	UXV765AC HE 21 CCW out vlv fc	2.2E-07/h
18	TXV34--C HE 21 SW out vlv dc	2.2E-07/h

(First of two screens)
*Where fc - fail to close; fo - fail to open; 1/r - leak or rupture. Select # and enter for change; select other to skip.

Figure C-7a PRAAGE Menu 4 - Failure Rates

#4 MENU 2 OF FAILURE RATES

No.	Component	Failure Rate
1	JBS-25AD Bus 5A failure	3.0E-06
2	UCV760AC Pump 23 in vlv fc	2.2E-07/h
3	UPM0021S CC pump 21 failure	9.1E-05/h
4	UXV761CQ CC pump 21 out ck fo	4.3E-06/h
5	UXV762CC CC pump 23 out vlv fc	2.2E-07/h
6	JBS-22AD Bus 2A failure	7.1E-05
7	UXV760BC Pump 22 in vlv fc	2.2E-07/h
8	UPM0022S CC pump 22 failure	9.1E-05/h
9	UCV761BQ CC pump 22 out ck fo	4.3E-06/h
10	UXV762BC CC pump 22 out vlv fc	2.2E-07/h
11	JBS-23AD Bus 3A failure	7.1E-05
12	UXV760CC pump 21 in vlv fc	2.2E-07/h
13	UPM0023S CC pump 23 failure	9.1E-05/h
14	UCV761AQ CC pump 23 out ck fo	4.3E-06/h
15	UXV762AC CC pump 21 out vlv fc	2.2E-07/h
16	UPM0021S CC pump 21 fail start	6.4E-03
17	UPM0022S CC pump 22 fail start	6.4E-03
16	UPM0023S CC pump 23 fail start	6.4E-03

(Second of two screens)
(Where fc = fail to close; fo = fail to open; 1/r = leak or rupture. Select # and enter for change; select other to skip.

Figure C-7b PRAAGE Manu 4 - Failure Rates

Selection 5 from the main menu computes the "Normalized Inspection Importance" (NII) for a given component but a menu is not presented - only a busy statement until the return to the main menu. NII is the calculation of the probability of the coincident component failures that result in system failure for the given component. This is normalized by dividing by the sum of all NIIs, the result of which is interpreted as the fractional contribution that the component makes to the system unavailability.

Selection 6 from the main menu calculates the Birnbaum Importance which is similar to NII except that it assumes the given component has failed, so its probability of failure is set to "1".

c) Presentation of Results

After PRAAGE has calculated an importance measure, the manner of presenting the results must be selected. It is assumed that the primary method of presentation is video and that this would normally be used before printing or graphing. This assumption reduces in computer coding but the data must be viewed to indicate to the printer or plotter the data to be used.

Selection 7 from the main menu presents the results as a "budget" of how much of the system unavailability is due to generic component classes. Before this is accomplished, a menu (Figure C-8) is presented for selecting the time at which ordering is to take place. This is necessary because the order of importance changes with age, since some components age faster than others. Results similar to those shown in Figure C-9 are presented.

Selection 8 from the main menu gives the results as a percent of the system unavailability that is due to the various generic classes of components. An example of this type of output is shown in Figure C-10.

Selection 9 presents results similar to those selected by 8, except they are divided by the number of components in the generic class. The reason for this is because a generic class collectively may be very important because it contains a large number of components each having a small importance, while some other class may have a large importance resulting from a few important components. An example of this data presentation is provided in Figure C-11.

Selection 10 presents results of the importance measures for the individual components (Figure C-12). Selection 11 prints the results of selections 7 through 10.

Selection 12 plots on the screen output selected by 7 through 9 on a log-log scale. Up to six items at a time may be plotted. A typical selection menu is shown as Figure C-13. If a print of the graph is required, typing "P" will do this. Figure 6-3 is a sample plot of some of the generic components presented in Selection 7.

What year do you want to use for sorting?

1	Year 1
2	Year 2
3	Year 5
4	Year 10
5	Year 20
6	Year 40

Figure C-8 PRAAGE Menu for Sort Time

#7 Unavail. Budget Contributions by Generic Components
All Numbers in Percent (%)

Class	Year1	Year2	Year5	Year10	Year20	Year40
UNAVAIL	0.02678	0.02678	0.02689	0.02867	0.03619	0.09544
SVC WTR	0.02479	0.02479	0.02479	0.02468	0.02022	0.01142
TOT VLV	0.00158	0.00158	0.00168	0.00336	0.00557	0.00650
CC.M.VLV	0.00105	0.00105	0.00112	0.00224	0.00369	0.00424
SW.M.VLV	0.00053	0.00053	0.00056	0.00112	0.00185	0.00209
PUMPS	0.00020	0.00020	0.00020	0.00036	0.01010	0.07721
HEAT EX	0.00017	0.00017	0.00019	0.00022	0.00025	0.00024
PIPE	0.00004	0.00004	0.00004	0.00004	0.00004	0.00002
CK VLV	0.00000	0.00000	0.00000	0.00000	0.00004	0.00016
ELECT	0.00000	0.00000	0.00000	0.00000	0.00001	0.00005

Figure C-9 Sample PRAAGE Unavailability Budget Results

#8 System Unavail and Generic Inspection Aging Importance
All Numbers in Percent (%)

Class	Year1	Year2	Year5	Year10	Year20	Year40
SVC WTR	92.5888	92.5888	91.1753	86.1031	55.8725	11.9693
TOT VLV	5.8945	5.8945	6.2476	11.7310	15.3979	6.8109
CC.M.VLV	3.9192	3.9192	4.1537	7.8010	10.1864	4.4467
SW.M.VLV	1.9665	1.9665	2.0845	3.9161	5.1010	2.1936
PUMPS	0.7340	0.7340	0.7318	1.2463	27.9093	80.9056
HEAT EX	0.6333	0.6333	0.6967	0.7788	0.6935	0.2480
PIPE	0.1452	0.1452	0.1446	0.1351	0.0876	0.0188
UNAVAIL	0.0268	0.0268	0.0269	0.0287	0.0362	0.0954
CK.VLV	0.0088	0.0088	0.0093	0.0140	0.1105	0.1706
ELECT	0.0041	0.0041	0.0041	0.0057	0.0391	0.0474

Figure C-10 Sample PRAAGE System Unavailability and
Generic Inspection Aging Importance

#9 System Unavail and Inspection Importance per Component
All Numbers in Percent (%)

Class	Year1	Year2	Year5	Year10	Year20	Year40
SVC WTR	92.5888	92.5888	92.1753	86.1031	55.8725	11.9693
CC.M.VLV	0.3266	0.3266	0.3461	0.6501	0.8489	0.3706
HEAT EX	0.3167	0.3167	0.3483	0.3894	0.3468	0.1240
SW.M.VLV	0.2809	0.2809	0.2978	0.5594	0.7287	0.3134
TOT VLV	0.2679	0.2679	0.2840	0.5532	0.6999	0.3096
PUMPS	0.2447	0.2447	0.2439	0.4154	9.3031	26.9685
UNAVAIL	0.0268	0.0268	0.0269	0.0287	0.0362	0.0954
PIPE	0.0076	0.0076	0.0076	0.0071	0.0046	0.0010
CK VLV	0.0029	0.0029	0.0031	0.0047	0.0368	0.0569
ELECT	0.0014	0.0014	0.0014	0.0019	0.0130	0.0158

Note: These are Display #8 values/number of components in class

Figure C-11 Sample PRAAGE System Unavailability and Inspection Importance per Component

INSPECTION AGING IMPORTANCE OF CCW COMPONENTS

Component	Year 1	Year 2	Year 5
TSWINOFL SW failure	9.3E-01	9.3E-01	9.2E-01
UXV734AC vlv 734A fc	2.0E-02	2.0E-02	2.1E-02
UXV734B vlv 734B fc	2.0E-02	2.0E-02	2.1E-02
TXV32--C SW iso vlv 32 fc	2.0E-02	2.0E-02	2.1E-02
UHE0021L CC HE 21 l/r	3.2E-03	3.2E-03	3.5E-03
UHE0022L CC HE 22 l/r	3.1E-03	3.1E-03	3.5E-03
UPM0021S CC pump 21 failure	2.5E-03	2.5E-03	2.5E-03
UPM0022S CC pump 22 failure	2.4E-03	2.4E-03	2.4E-03
UPM0023S CC pump 23 failure	2.4E-03	2.4E-03	2.4E-03
UPPFAILS CC pipe failure	1.4E-03	1.4E-03	1.4E-03
UTK0021L CC surge tank l/r	7.6E-05	7.6E-05	7.6E-05
UCV761CQ CC pump 21 out ck fo	3.0E-05	3.0E-05	3.1E-05
UCV761BQ CC pump 22 out ck fo	2.9E-05	2.9E-05	3.1E-05
UCV761AQ CC pump 23 out ck fo	2.9E-05	2.9E-05	3.1E-05
JBS-23AD Bus 2A failure	2.0E-05	2.0E-05	2.0E-05
JBS-23AD Bus 3A failure	2.0E-05	2.0E-05	2.0E-05
TXV35--C HE 21 SW out vlv fc	1.8E-05	1.8E-05	2.1E-05
UXV766AC HE 21 CCW in vlv fc	1.8E-05	1.8E-05	2.1E-05
UXV765AC HE 21 CCW out vlv fc	1.8E-05	1.8E-05	2.1E-05
TXV34--C HE 21 SW out vlv fc	1.8E-05	1.8E-05	2.1E-05
TXV35-1C HE 22 SW out vlv fc	1.8E-05	1.8E-05	2.0E-05
UXV766BC HE 22 CCW in vlv fc	1.8E-05	1.8E-05	2.0E-05
UXV765BC HE 22 CCW out vlv fc	1.8E-05	1.8E-05	2.0E-05
TXV34-1C SW HE 22 out vlv fc	1.8E-05	1.8E-05	2.0E-05
TXV33--C SW iso vlv 33 fc	1.8E-05	1.8E-05	2.0E-05
TXV31-1C SW iso vlv 31-1 fc	1.8E-05	1.8E-05	2.0E-05
UXV760AC Pump 23 in vlv fc	1.5E-06	1.5E-06	1.6E-06
UXV762CC CC pump 23 out vlv fc	1.5E-06	1.5E-06	1.6E-06
UXV760BC Pump 22 in vlv fc	1.5E-06	1.5E-06	1.6E-06
UXV760BC CC pump 22 out vlv fc	1.5E-06	1.5E-06	1.6E-06
UXV762CC Pump 21 in vlv fc	1.5E-06	1.5E-06	1.6E-06
UXV762AC CC pump 21 out vlv fc	1.5E-06	1.5E-06	1.6E-06
JBS-25AD Bus 5A faiaure	8.7E-07	8.7E-07	8.6E-07

(First of Two Forms)

Figure C-12 Sample PRAAGE Inspection Aging Importance of CCW Components

You Can Only Plot Generic Displays

To control clutter, only 6 generic components can be selected at one time, so select from the following menu:

1	UNAVAIL
2	SVC WTR
3	TOT VLV
4	CC.M.VLV
5	SW.M.VLV
6	PUMPS
7	HEAT EX
8	PIPE
9	CK. VLV
10	ELECT.

Select then enter. To end select, spacebar then enter.

Figure C-13 PRAAGE Plot Selection Menu

APPENDIX D

COMPONENT POPULATION ESTIMATES

As part of the data analysis performed for this study, failure rates were calculated for various components. To do this, population information was required for the components. This appendix describes the methods used to estimate the populations.

The population of pumps and heat exchangers was determined from a review of each plant's CCW system design. Since these populations are relatively small, the components were counted directly.

Valve population was more difficult to determine due to the large number of valves and the various types used in each plant. Therefore, the piping and instrumentation drawings of the CCW system for Indian Point-2, Braidwood 1 and 2, and Prairie Island were reviewed, from which an average number of manual valves per component or load serviced by the CCW system was determined. The averages are shown in Table D-1. Together with results of the design reviews performed for each plant which identified the major components and loads, these numbers were used to estimate a total population of manual valves.

The population of motor-operated valves (MOV's) was determined from the design reviews which identified the key MOV's in the system which performed automatic functions. The number then was adjusted to account for other automatic system functions not identified, such as train splitting, nonsafety load isolation, containment isolation and surge tank isolation. An additional factor of three MOV's per system was added to account for non-automatically controlled MOV's which are typically found in system designs.

The air-operated valve (AOV) population also was determined from the design reviews, which identified the key AOV's used to perform automatic system functions. An adjustment of one AOV per surge tank was then added to account for surge tank vents.

Check valve population was estimated by including one check valve for each CCW pump, each auxiliary cooling water pump, each reactor coolant pump cooled by CCW and each surge tank. An adjustment factor of 8 valves was then added for each system containing a large number of loads to account for miscellaneous check valves. An adjustment factor of 2 was used for systems with a small number of loads.

Results of the population estimates are shown in Table D-2.

Table D-1 Average Number of Manual Valves Per Component/Load

	Component/Load	Number of Valves
1.	Auxiliary Component Cooling Pump	14
2.	Auxiliary Feedwater Pump	10
3.	Charging Pump	12
4.	Chilled Water Condenser	7
5.	Control Room Air Conditioner/Essential Chiller	7
6.	Core Spray Pump	10
7.	Core Spray Hx	15
8.	Component Cooling Water Pump	8
9.	Component Cooling Water Hx	6
10.	CRD, CRDM, CEDM, CEA	11
11.	CR Chiller	7
12.	Emergency Diesel Generator	12
13.	ESF	12
14.	Excess Letdown	6
15.	H_2 - Recombiner	4
16.	Letdown	5
17.	Make Up Miscellaneous	2/Item
18.	Make Up Pump	10
19.	Miscellaneous	70
20.	Non-Regenerative Hx	11
21.	Normal Chiller	7
22.	Reactor Coolant Pump	14
23.	Recirculation Pumps	19
24.	RHR Pumps	10
25.	RHX	15
26.	Rx Building Cooling Unit/Fan Air CLC	15
27.	REDT	5
28.	Safety Injection Pump	13
29.	Seal Water HX	8
30.	Spent Fuel Hx	11
31.	Surge Tank	9
32.	SDHX	15
33.	Ups - AC/Uninteruptable Power	7
34.	Positive Displacement Pump	12

Table D-2 CCW Component Population Estimates

	Plant	Pumps	Hx's	MOV's	Check Valves	Manual Valves
1.	Beaver Valley 1	3	3	12	16	208
2.	Byron 1 & 2	5	3	14	19	465
3.	Callaway 1	4	2	6	18	257
4.	Catawba 1	4	2	11	18	291
5.	Cook 1 & 2	5	4	22	31	465
6.	San Onofre 3	3	2	4	17	259
7.	St Luci 1 & 2	6	4	12	32	510
8.	Waterford 3	3	4	6	16	297
9.	Diablo Canyon 1 & 2	6	4	6	32	520
10.	Farley 1 & 2	6	6	10	32	470
11.	Ginna	2	2	0	13	195
12.	Haddam Neck	2	2	0	11	118
13.	Indian Point 2 & 3	6	4	20	31	494
14.	Kewaunee	2	2	11	13	178
15.	McGuire 1 & 2	8	4	32	34	448
16.	Millstone 3	7	7	6	20	302
17.	Ano 1 & 2	6	6	17	32	338
18.	Calvert Cliffs 1 & 2	6	4	0	28	366
19.	Fort Calhoun	3	4	4	14	240
20.	Maine Yankee	4	4	1	18	213
21.	Millstone 2	3	3	2	16	294
22.	Palisades	3	2	7	16	189
23.	San Onofre 2	3	2	4	17	259
24.	Crystal River 3	3	4	2	16	175
25.	Davis Besse	3	3	2	16	258
26.	Oconee 1 & 2 & 3	6	5	24	21	165
27.	Rancho Seco	4	4	0	27	244
28.	Three Mile Island 1	7	8	12	33	327
29.	North Anna 1 & 2	4	4	8	17	456
30.	Point Beach 1 & 2	4	3	16	28	324
31.	Robinson 2	3	2	0	14	233
32.	Salem 1 & 2	6	4	0	32	458
33.	San Onofre	3	2	0	14	218
34.	Sequoah 1 & 2	5	3	6	19	308
35.	Summer	3	2	16	16	208
36.	Surry 1 & 2	8	8	9	25	507
37.	Trojan	3	2	15	17	256
38.	Turkey Point 3 & 4	6	6	16	28	430
39.	Wolf Creek	4	2	4	18	270
40.	Zion 1 & 2	5	3	10	17	420
	TOTALS	177	145	347	852	12633

APPENDIX E

DISCUSSION OF NATIONAL DATA BASES

CONTENTS

E.1 Nuclear Plant Reliability Data System (NPRDS)

The NPRDS data base was established in 1974 in response to the growing need for data on failure of nuclear plant components. It was developed by the Southwest Research Institute (SWRI) which processed all data from 1974 to 1983. Starting in 1974 data was reported to the NPRDS on a voluntary basis. Since participation was voluntary, data during the earlier years is incomplete.

In January 1982, the Institute of Nuclear Power Operations (INPO) assumed responsibility for the management of the data base, and in July 1983 took over operation of the NPRDS. INPO currently maintains the data base in their computers and performs all data processing. Participation is still voluntary, but reporting has improved significantly, although it still has deficiencies.

The data reported to the NPRDS include engineering and failure information for safety-related systems and components. The information is reported on standardized forms using codes developed specifically for the NPRDS. The engineering information includes the component in-service date, safety class, operating mode, operating environment, manufacturer, model number, design specifications, and frequency of testing. Engineering information is submitted for every reportable component.

When a component fails, a report is submitted by the plant to the NPRDS which includes the date the failure occurred, the date it was discovered, the date corrected, the system status when the failure occurred, and the method by which it was detected. The report also includes the cause of failure, its effect on the system, and narrative descriptions of the cause and corrective actions taken.

As for each of the data bases used, the NPRDS data base has certain limitations which must be recognized. One major limitation is that participation in the NPRDS is voluntary. Since not all plants participated at first, the failures reported during the initial years cover only a small portion of the total. This limitation is further complicated by the fact that the boundaries of the components for which data was to be reported were not well defined. Both these situations have improved over the years, but it must be recognized that the number of failures reported to NPRDS will be optimistic (lower than actual).

An additional limitation of the NPRDS is the inconsistencies between plants in interpreting the numerous codes and definitions which must be used in filing reports. The contents of the reports submitted by the various plants are influenced by the experience and training of the individual reporting the data. This was evident in the reports examined for data identifying the failed component, the cause of failure, and whether it was related to age. For example, if a valve failed to open because of a problem with the actuator, the component failed was sometimes reported as the valve instead of the actuator. This was corrected during the review.

A further aspect of the NPRDS data which should be noted concerns the date of occurrence of the failure. If it is unknown, the date reported is midway between the time the component was last known to be functional and the time the failure was discovered. This "best estimate" could introduce errors in determining the age of the component at failure.

An additional limitation is that the process of obtaining NPRDS data is unnecessarily difficult and time-consuming since access to the data base is currently restricted. This is felt to be a major deficiency in this data base.

E.2 In Plant Reliability Data System (IPRDS)

The second source used was the IPRDS data base, which was established in 1978 to compile a more complete history of component operating experience at nulcear power plants. Data from in-plant maintenance and repair records were collected for the data base. Initial collection was made by the IEEE Reliability subcommittee which relied on volunteers to perform the work.

The IPRDS data base is unique in that it contains data on component populations, failures and repairs obtained directly from raw, in-plant records. Various plants were visited, and the records for specific types of components were collected.

Data was obtained for safety-related and non-safety related components; the information included a population record for each type of component in the plant. These records describe the design, operating environment, and operating mode of the component and can be used to determine the total component population for the plant.

For each component, records of its failures and repairs were obtained. The failure record includes the date the failure occurred, its mode, cause, and severity, and a description of the failure. The repair record includes the time to complete repairs, the crew size, and a description of the repairs performed.

The major disadvantage of the IPRDS data base is the limited amount of data in it. Information on only pumps and valves was collected before the program was terminated in 1983. These data were obtained from visits to three PWR plants and four BWR plants, but not all records from the plants were incorporated into the data base. Data from the IPRDS are, therefore, expected to be fairly complete for only a few plant units and are not expected to be representative of a cross-section of the industry.

E.3 Licensee Event Reports (LERs)

The third source of data used is the Licensee Event Reports (LERs). As part of the licensing requirements for nuclear power plants, the licensee must report to the NRC any event (including the failure of safety-related components) which results in a deviation from the plant's technical specifications. In 1973 a data base was established containing information extracted from these reports.

The information presented in the LER includes the date of the event, along with a description. The LERs were not originally intended to be used as a source of reliability data and have less detailed information than the other data bases.

As with the other data bases, the LER data base has various limitations. Since the system was not intended as a source of reliability information, the records do not contain information such as component population or failure modes and mechanisms. In addition, failures are only reported for safety-related components. However, not all failures are reported, even if the component is safety-related. Some may not result in deviations from technical specifications and are not reportable. After a change in LER reporting requirements in 1984 failure data is even more limited, due to the reduction in the scope of reportable events. For these reasons, data from the LER data base are considered conservative (lower than actual). Results from LER data such as dominant failure modes and causes, however, are expected to be consistent with those based on NPRDS data since both data bases represent a cross-section of all operating plants.

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An aging assessment of Component Cooling Water (CCW) systems in Pressurized Water Reactors (PWRs) 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 generated will be used to assess the impact of aging on plant safety and to develop effective mitigating actions.

Aging in the CCW system was 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 in CCW systems. Time-dependent failure rates for major components were calculated to identify aging trends. Plant specific data were obtained and evaluated to supplement data base results.
A computer program (PRAAGE) was developed and implemented to model a typical CCW system design and perform Probabilistic Risk Assessment (PRA) calculations. Time-dependent failure rates were input to the program to evaluate the effects of aging on component importance and system unavailability. Changes in component importance and system unavailability with age were observed and are discussed.

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