NUREG/CR-5379
PNL-7916
Vol. 2

Nuclear Plant Service Water System Aging Degradation Assessment

Phase II

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Pacific Northwest Laboratory
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Prepared for
U.S. Nuclear Regulatory Commission

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NUREG/CR-5379
PNL-7916
Vol. 2
RM, R9

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Manuscript Completed: July 1992
Date Published: October 1992

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NRC FIN B2911

Abstract

The second phase of the aging assessment of nuclear plant service water systems (SWSs) was performed by the Pacific Northwest Laboratory (PNL) to support the U.S. Nuclear Regulatory Commission's (NRC's) Nuclear Plant Aging Research (NPAR) program. The SWS was selected for study because of its essential role in the mitigation of and recovery from accident scenarios involving the potential for core-melt, and because it is subject to a variety of aging mechanisms. The objectives of the SWS task under the NPAR program are to identify and characterize the principal age-related degradation mechanisms relevant to this system, to assess the impact of aging degradation on operational readiness, and to provide a methodology for the management of aging on the service water aspect of nuclear plant safety.

The primary degradation mechanism in the SWSs, as stated in the Phase I assessment and confirmed by the analysis in Phase II, is corrosion compounded by biologic and inorganic accumulation. It then follows that the most effective means for mitigating degradation in these systems is to pursue appropriate programs to effectively control the water chemistry properties when possible and to use biocidal agents where necessary.

A methodology for producing a complete root-cause analysis was developed as a result of needs identified in the Phase I assessment for a more formal procedure that would lend itself to a generic, standardized approach. It is recommended that this, or a similar methodology, be required as a part of the documentation for corrective maintenance performed on the safety-related portions of SWSs to provide an accurate focus for effective management of aging.

Summary

The goal of the Service Water System (SWS) Aging Degradation Assessment task was to advance the understanding and management of the technical safety issues relating to the aging of SWSs in operating commercial nuclear power plants.

A Phase I interim assessment report, with the primary goal of understanding the aging process in SWSs, was completed in June 1989 (NUREG/CR-5379). The report concluded that the principal mechanism leading to SWS degradation and failure is corrosion compounded by biologic and inorganic accumulation. This conclusion was based on in-depth, single plant, open system (once-through) plant information and was verified using opinions of subject experts. The report also pointed out that existing performance measures performed in compliance with the plant technical specifications were not capable of effectively detecting aging degradation prior to failure; nor were the existing diagnostics capable of providing adequate identification of the stressor(s) responsible for the root cause of degradation or failure.

The Phase II comprehensive aging assessment was performed to substantiate the Phase I conclusions, as well as to explore suitable methods of monitoring and managing the aging of the nuclear plant SWS. The Phase I in-depth database was extended with similar data from a closed SWS type (intermediate heat exchanger), and a recirculating (spray pond) plant to provide reasonable assurance that all common degradation mechanisms and stressors had been reviewed. The analysis of the extended Phase II database and a review of the comprehensive NRC-AEOD assessment of SWS degradations and failures (NUREG-1275, Vol. 3) show the Phase I conclusions, relative to corrosion being the degradation mechanism primarily responsible for SWS failures, to be valid. This conclusion suggests that the key to SWS aging mitigation lies in the control of service water quality and chemistry. In systems with limited chemistry control (such as in the open system), the most effective efforts to prevent aging failures are targeted toward:

- understanding the active degradation mechanisms at the specific site
- selecting system materials to minimize the impact of the known degradation mechanisms
- using system designs that allow thorough periodic cleaning
- developing a comprehensive condition monitoring program to detect component degradation early enough to allow planned maintenance actions.

The issues regarding management of SWS aging have received considerable attention from the U.S. Nuclear Regulatory Commission (NRC). With the implementation of the Generic Letter 89-13 on SWS problems affecting safety-related equipment, utilities are required to conduct component performance tests and inspections and provide chemical biofouling control. The work summarized in this and other Pacific Northwest Laboratory (PNL) reports provided the basis for, and input to, the formulation of Generic Letter 89-13.

The final product of the Phase II effort is the development of a generic root-cause analysis methodology. This report recommends that this method, or a similar root-cause analysis process having the elements described herein, be performed for all corrective maintenance required on safety-related portions of the nuclear plant SWS. Integrating the documentation of these analyses will then provide an accurate focus for an effective, plant-specific, SWS aging mitigation program.

Acknowledgments

The authors wish to thank all of the cooperating power plants for providing knowledge, as well as data records on operations and maintenance of their service water systems. We would also like to thank our sponsor, Dr. J. J. Burns of the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, for his encouragement and drive to spread the knowledge gained through this project to other NRC components. Dr. A. B. (Burt) Johnson, Jr., project manager, provided technical direction and expertise and review of the material. The efforts of our editor, Nora Buel, are also gratefully acknowledged.

Acronym List

ACWS	auxiliary cooling water system
AEOD	Office for Analysis and Evaluation of Operational Data
ARCs	annunciator response cards
BWR	boiling-water reactor
DBA	design-basis accident
DBE	design-basis earthquake
ECCS	emergency core cooling system
EDG	emergency diesel generator
EPRI	Electric Power Research Institute
ESF	emergency safety feature
ESW	essential service water
FDT	forced draft cooling term
FSAR	final safety analysis report
GDC	general design criteria
HX	heat exchanger
HXFR	heat transfer
IPA	Integrated Plant Assessment
LERs	License Event Reports
LCO	limiting conditions for operation
LOCA	loss-of-coolant accident
MIC	microbiologically-influenced corrosion
MOVs	motor-operated valves
MPC	maximum permissible concentration

MSIV	mainstream isolation valve
M/U	make-up (water source)
NPAR	Nuclear Plant Aging Research (program)
NRC	U.S. Nuclear Regulatory Commission
PNL	Pacific Northwest Laboratory
PWR	pressurized-water reactor
RCA	root-cause analysis
RHR	residual heat removal
SAR	safety analysis report
SER	safety evaluation report
SRP	Standard Review Plan
SSW	standby service water
SWS	service water system
SWWG	Service Water Working Group
TS	technical specifications
UDC	under deposit corrosion
UHS	ultimate heat sink
USAR	updated safety analysis report

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1 Introduction

The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, Division of Engineering, is implementing the Nuclear Plant Aging Research (NPAR) Program Plan (NRC 1991) to resolve technical safety issues related to the aging of commercial nuclear power plants. Aging, in the context of the characteristics of a system, structure, or NPAR Program Plan, means gradual changes in the component with time or use as the result of one or more of the following factors:

- natural processes during operation
- external stressors caused by storage or operation
- service wear caused by operational cycling
- excessive testing
- improper installation, application, operation or maintenance.

This report was prepared by Pacific Northwest Laboratory (PNL);[a] it documents the second phase of the Nuclear Plant Service Water System (SWS) Aging Degradation Assessment task of the NPAR program.

The report summarizes investigations of the safety-related portions of the nuclear plant SWS [often designated as essential service water (ESW) or standby service water (SSW)]. During a loss-of-coolant accident (LOCA) or similar core-threatening, postulated accident scenario, this system is relied upon to transfer heat from vital plant equipment, such as the residual heat removal (RHR) heat exchangers (HXs) and the emergency diesel generators (EDGs), to the ultimate heat sink (UHS). This investigation emphasizes identifying and characterizing the mechanisms of material and component degradation during service, and evaluating methods of inspection, surveillance, condition monitoring and maintenance to mitigate these effects.

(a) PNL is operated by Battelle Memorial Institute for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

1.1 Nuclear Plant Aging Research Program Goals

The specific goals of the NPAR program are:

1. Identify and characterize the primary aging mechanism(s) that could cause safety-related component degradation.
2. Identify methods of inspection, surveillance, and monitoring, which will ensure timely detection of significant aging effects prior to loss of safety function.
3. Evaluate the effectiveness of storage, maintenance, and replacement practices in mitigating the rate and extent of aging degradation.

This report documents the second phase of the NPAR phased approach to system research, as stated in the NPAR Program Plan (NRC 1991). It describes the results of Phase II for the SWS task and includes information from the following:

- available subject literature
- available generic databases
- utility machinery history
- utility system expert inputs
- commercial expertise
- expert SWS observations.

This information is analyzed and evaluated to identify the principal SWS aging mechanisms to provide recommendations for monitoring of aging management programs.

1.2 System Risk Importance

The impetus for this study originates from the significant number of documented SWS degradation-related events that have seriously impaired the ability of this system to carry out its intended safety function. As previously stated, the system safety function is vital to the successful termination of many potential core-melt scenarios. The NRC report on SWS failure and degradation, issued by the Office for Analysis and Evaluation of Operational Data (AEOD) (Lam and Leeds 1988), indicated significant SWS system degradation performance with a frequency of 0.4 per reactor year of operation, and a complete system failure (loss of all system functions) of 1.5×10^{-2} per reactor year. These figures are among the 10 highest risk factors found in an NRC study on severe accident risks (NRC 1989). This provides clear corroboration of the risk frequency inferred by the Phase I study.

Considering the multiple safety functions that are provided by the SWS and the confirmed frequency of partial or complete functional degradation, the importance of understanding, managing and monitoring the stressors and condition of the SWS is considered essential to minimizing core damage probability.

1.3 SWS Task Development Summary

The complete SWS task is depicted in Figure 1.1. The task consists of understanding the aging phenomena (Phase I), managing the aging process and monitoring for aging degradation (Phase II).

1.3.1 Understanding Aging

The Phase I effort was initiated by a review of the regulatory requirements for nuclear plant SWSs, which resulted in an understanding of the Generic Issues [GSI 51 (Hayes 1983)] and related regulatory concerns for the subject system. The design bases for system operability were defined from the General Design Requirements 44, 45 and 46 of 10 CFR 50 (1983).

A team of engineers representing component design, operations and maintenance, corrosion metallurgy, and statistical analysis was assembled at PNL to perform the assessment. This team, in conjunction with utility experts and water chemistry consultants, provided the core of expertise necessary to produce a real-world assessment of the aging stressors that exist in the SWS environments.

A 1987 review of available literature and database information indicated that motor-operated valve torque switches were the source of failures in the majority of SWS failure events. This conclusion was in conflict with the operating experiences of the PNL team, suggesting that corrosion was the principal degradation mechanism, and that the torque switch "malfunction" was only a manifestation of the more basic stressor which produced valve disk to seat corrosion. To resolve the issue, the assistance of a cooperating utility was enlisted to provide actual machinery records and personnel recollection for analysis. These accounts contradicted the dominance of electrical component failures, indicating that biologic and sedimentary accumulation were contributory mechanisms to the primary degradation mechanism, corrosion. This fundamental conclusion was confirmed by the analysis performed in Phase II, and forms the premise for the remainder of the study.

The remaining Phase I conclusions were actually the identification of specific investigative foci on 1) the obvious need for a systematic approach to performing and documenting the root cause of a component failure event, and 2) the instrumentation schemes that would be necessary to detect the progress of degradation phenomena to prevent or mitigate any subsequent failure events.

A Phase I report was prepared to document the methodology and findings of the initial phase of the SWS task (Jarrell et al. 1989). The report was extensively peer reviewed by participating NPAR laboratories and by the Electric Power Research Institute (EPRI) Service Water Working Group (SWWG), a task force from 36 nuclear utilities. It was focused on resolving the source of degradation problems encountered during SWS operation. This completed the Phase I basic understanding of the SWS aging mechanisms.

1.3.2 Managing Aging

Following the focus established by Phase I, the second phase of the SWS task (refer again to Figure 1.1) focused on understanding the management of aging processes from a regulatory perspective.

1.3.2.1 Regulatory Implementation Scheme

The nuclear power industry is regulated by the NRC to comply with a set of general design criteria (GDC) intended to ensure the protection of the public from the potential dangers of radioactive release in the event of a postulated plant accident. The GDCs are the underlying basis of all nuclear plant operations and maintenance. GDCs 44, 45, and 46 (10 CFR 50 Appendix A) define the functions and delineate the required component redundancy, inspection, and testing for the required levels of components important to safety for nuclear plant SWSs.

These design criteria are subsequently carried (see Figure 1.2) into the updated safety analysis report (USAR), which is produced for each nuclear power plant to demonstrate the manner in which the GDCs will be implemented for a specific power plant. With regard to the SWS, the design safety specifications for each SWS cooler are generally stated in terms of the required heat removal rate that must be met under both LOCA and plant emergency shutdown conditions. The adequacy of these specifications is then evaluated by the NRC in a safety evaluation report (SER), which provides an adequacy judgment for each condition delineated in the utility's USAR. These two documents, the USAR and the SER, form the basis for an agreed-upon envelope of licensed operating conditions.

The actual implementation of the contractual license agreement is embodied in the technical specifications for each plant. These documents translate the requirements for safe operation as defined by the USAR and approved by the SER into parametric limits for plant operation and component (and system) surveillance testing. Since the operational and surveillance limits are descendants of the GDCs, operation within established normal operating bands and test results that are within acceptance criteria should indicate that the plant operations and maintenance are inside the safety envelope established by the GDCs.

As indicated by Figure 1.2, the failure of a safety-related component to fulfill its intended function either during normal operation or during a surveillance test indicates a reduced ability to cope with potential accident conditions. Consequently, a timed limit on the duration of operation under a specific failure condition is established. For component failure conditions which do not require an immediate cessation of operation, limiting conditions for operation (LCO) are said to exist. These LCO time intervals are based on the increased risk of not being able to successfully mitigate a hypothetical core-melt scenario because of the degraded functionality of a component or system.

As Figure 1.2 indicates, all plant operations and maintenance are guided by the GDC through application of the regulatory structure.

1.3.2.2 Source of Information

The primary inputs to the Phase II effort consisted of two additional utility visits, the AEOD report on SWS failures (Lam and Leeds 1988), and a case study at an operating boiling-water reactor (BWR) site. These inputs provided the necessary information from which to build and partially substantiate the validity of the Phase I and II products.

The two additional cooperating utility visits provided validated degradation and failure information to the SWS database and allowed a single-point examination of all types of SWS configurations for stressors which might have otherwise gone unrecognized.

The AEOD failure and degradation compilation provided substantiating data on a more global basis. The use of Licensee Event Reports (LERs) as input data for root cause of failure investigations was discarded as potentially misleading during Phase I of this task (Murphy et al. 1984). More demanding root-cause evaluation requirements have been added since that publication, and the event compilation presented by the AEOD correlates well with the independent conclusions of this assessment.

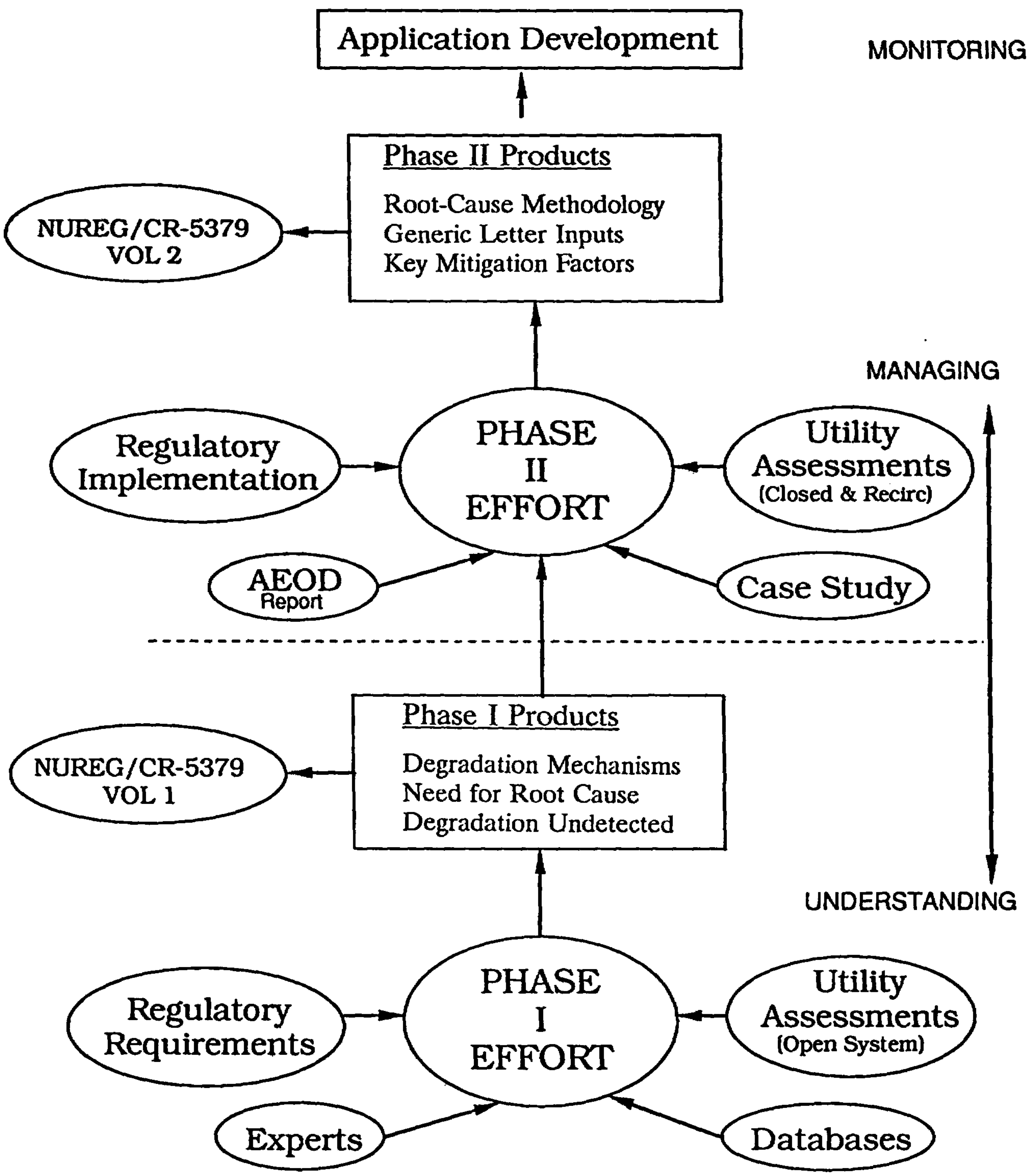

Figure 1.1 Service Water System Task Development

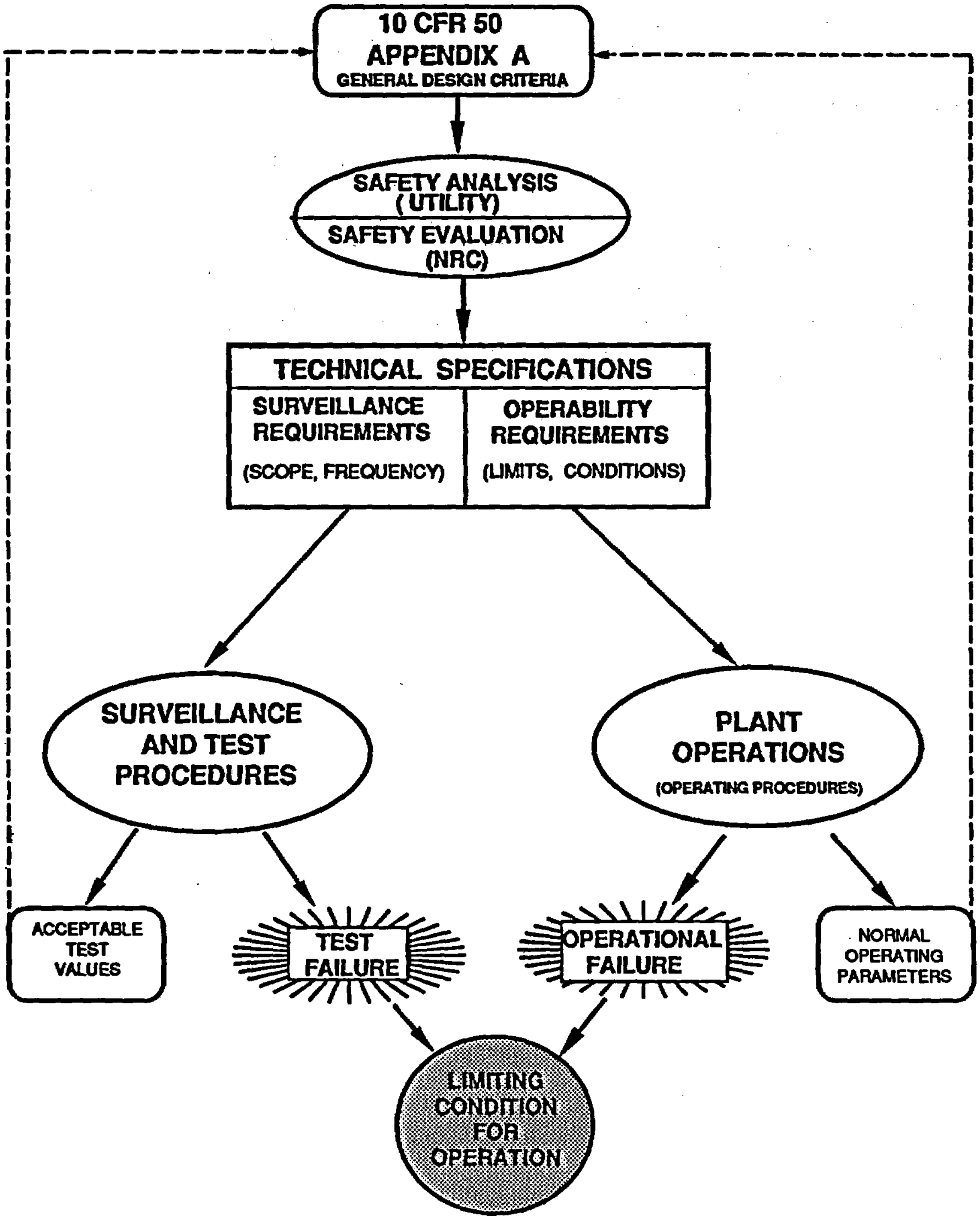

Figure 1.2 Design Bases Regulated Operation

The final information source was provided through PNL assistance to regional NRC offices in investigating SWS events at operating BWR plants. These opportunities not only provided additional in-depth validated failure data, but also allowed a "field test" of the root-cause methodology developed by this task.

1.3.2.3 Aging Management Products

Two of the three principal aging management products (refer to Figure 1.1) are incorporated into this report: SWS aging control mitigation factors and the failure root-cause analysis methodology. The SWS Generic Letter (NRC 1989) on SWS problems affecting safety-related equipment has been supported by this task, as well as other PNL efforts (Daling et al. 1989; Neitzel and Johnson 1988). Generic Letter 89-13 was published in the Federal Register (54 FR 43209-10).

1.3.3 Monitoring Aging

The NRC's Generic Letter on SWS problems provides an effective agent to monitor the degradation process in SWSs. Individual plant records of component conditions and failure histories are the true indicators of the effect of environment on the SWS. A PNL goal is to provide continued support of individual plant monitoring efforts and to assist in collective analysis by the NRC AEOD.

2 System Definition and Description

The three safety-related heat sources served by the SWS in transferring heat loads from various sources in the plant to the ultimate heat sink are core decay heat, decay heat removal components, and emergency power sources. The functional definition and design description of the SWS pertaining to heat load transfer are explained in the following sections.

2.1 SWS Definition

Because of the wide variation in each plant's ultimate heat sink and the application of multiple system design approaches, the SWS is defined from a functional standpoint. This includes all components, their associated instrumentation, controls, electrical power, cooling and sealing water, lubrication, and other auxiliary equipment comprising the final heat transfer loop between the safety-related heat sources and the ultimate heat sink.

The functional definition of an SWS is shown in Figure 2.1. The dashed boundary in the figure shows the following range of components considered by this study:

- the intake structure, including canals or other diversion structures from the ultimate heat sink to the intake debris removal mechanism
- the pump galley and structures with all associated water-level control devices (weirs, gates, valving, etc.) and instrumentation
- the service water pump, shafting and motive source, including controls, cabling and electrical distribution system
- the piping distribution network from the pumps to the heat exchangers, including all valving, manifolds, instrumentation, and logic networks
- the service water side of the actual heat exchange devices themselves
- all discharge piping, valves, and manifolds from the heat exchangers to the outlet or discharge structure
- the discharge structure, gates, and associated effluent channeling devices
- where applicable, the ultimate heat source structures and components.

Only those components specified as essential to reactor safety [Nuclear Safety Class 3 (NRC 1978b)] and designated Seismic Category I are examined in this investigation. Seismic Category I is that category requiring that plant structures, systems, and components be designed to withstand a design-basis earthquake (DBE).

In addition to the other requirements, safety-related service water cooling loops in commercial reactors are also designed to meet the single failure criterion. That is, redundant components are provided such that the failure of any single active component in the SWS will not prohibit the adequate removal of heat from any of the safety-related loads.

2.2 SWS Descriptions

SWSs can be categorized into a number of groupings; the grouping method used for this investigation is based on the nature of the plant's ultimate heat sink. This method provides a practical division of SWSs into two broad classifications, open and closed, although the open SWS classification is often subdivided into two configurations, open once-through and open recirculating. The following description of materials, environments, and configurations provides a broad knowledge of the design and function of SWS components and does not represent any specific BWR or pressurized-water reactor (PWR) design.

2.2.1 Open Once-Through Systems

A simplified diagram of a typical open SWS is shown in Figure 2.2a. This type of arrangement, often referred to as a "straight through" system, is generally characterized by the availability of a large volume of water as an ultimate heat sink. The advantage of this configuration is its relative simplicity and resultant lower initial cost to

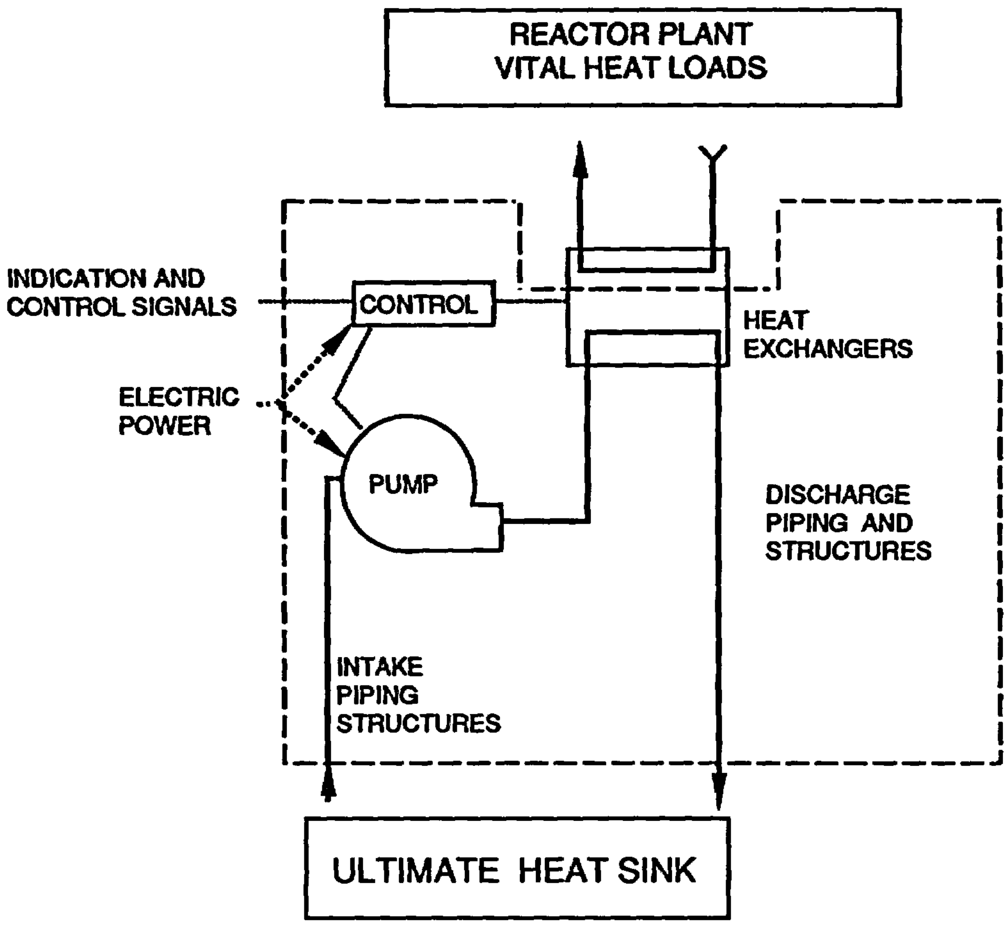

Figure 2.1 Functional Service Water System Boundary Definition

the utility. The lack of a large-capacity intermediate heat exchanger and no requirement for a secondary set of component cooling water pumps, as found in the closed-cycle configuration, make this an attractive layout because it can be installed economically and because it has fewer components to maintain. The major disadvantage is the potential for problems associated with exposing a large number of components to a potentially aggressive raw water environment.

2.2.2 Open Recirculating Systems

The recirculating version (Figure 2.2b) has a self-contained ultimate heat sink, which is frequently achieved through using a spray cooling pond or a dedicated cooling tower. The advantage of this arrangement is twofold: 1) through settling and make-up water filtration, the water purity (turbidity) is vastly improved, leading to significantly reduced siltation in low-flow velocity areas; and 2) chemical control of the circulated water is achievable, allowing a reduction in corrosion and biofouling without the limitations imposed by environmental discharge restrictions.

2.2.3 Closed Systems

The basic distinction between open and closed systems is the plant personnel's ability to control the coolant chemistry that comes in contact with the system load heat exchangers. A simplified closed system, illustrated in Figure 2.3, resembles the open configuration of Figures 2.2a and b, but adds an intermediate heat

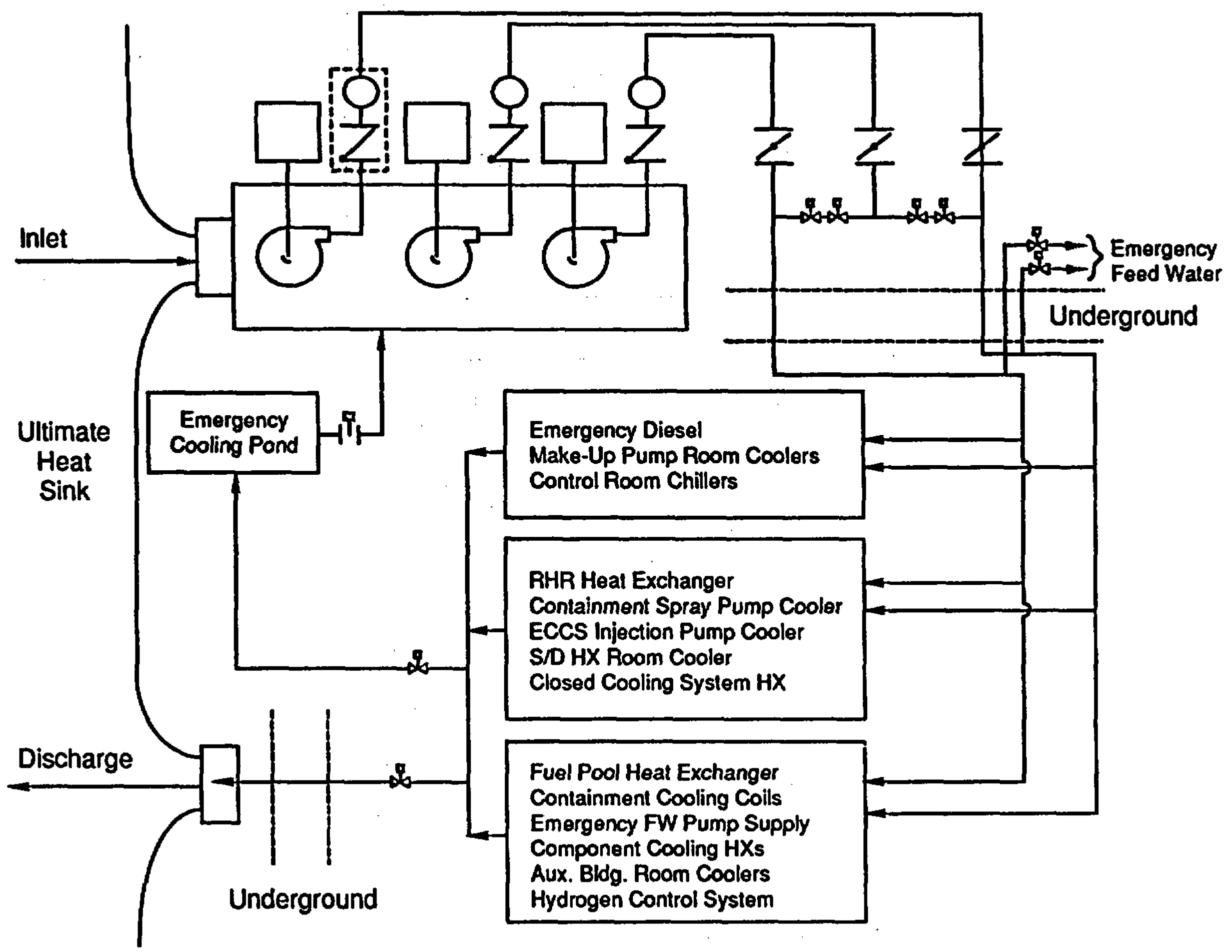

Figure 2.2a Open Service Water System - Once-Through System

exchanger to prevent exposure of associated component load heat exchangers to a raw cooling water environment.

The closed system has been typically sited where corrosive or other hostile environments are known to exist (such as an ocean salt water source). This system's open loop is designed to be simple and redundant for on-line maintenance, and is often provided with the means for mitigating expected degradation stressors (e.g., thermal back flush for biofouling control). The actual closed loop cooling water is chemically pure and pH neutral, and component degradation can be controlled using corrosion inhibitors.

The closed system is thus more expensive initially and contains a larger total number of components, but is not as susceptible to premature aging through extensive corrosion attack within the secondary loop.

2.3 SWS Summary

A compilation of all active commercial reactors, their electrical power ratings, source of water used for the ultimate heat sink, and specific SWS configuration is given in Table 2.1. This listing is primarily sorted by SWS cycle type, with plant names listed alphabetically for each of the types. A distinction is made between a closed pond system, which could be controlled via additive chemicals, and an open pond, which is considered to be too large for effective chemical control.

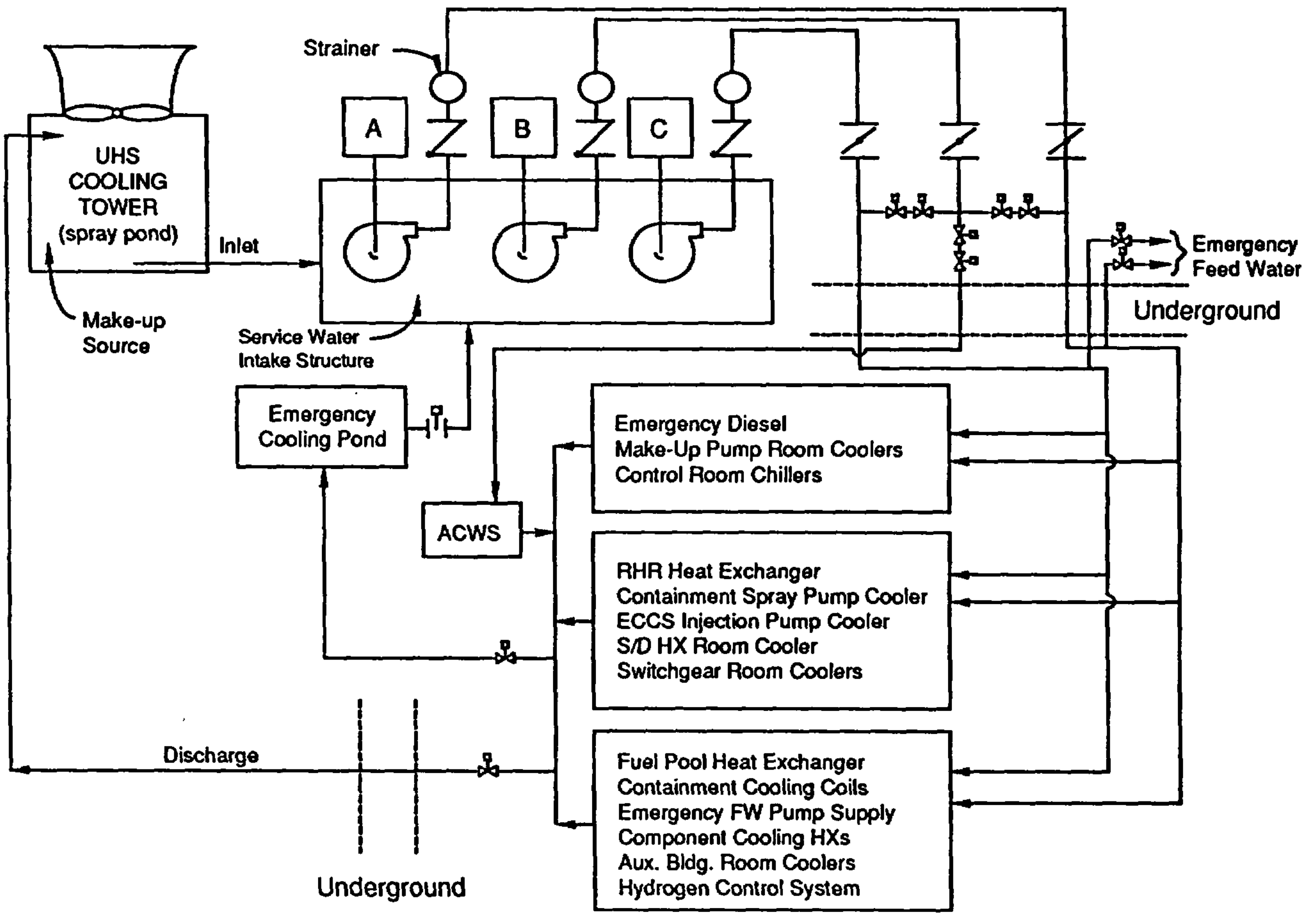

Figure 2.2b Open Service Water System - Recirculating System

2.4 System Characteristics

The conditions and materials, and the component function and process flow path of the SWS are described in the following sections.

2.4.1 Conditions and Materials

Service water system materials are subject to a wide range of environmental conditions depending primarily on the nature of the ultimate heat sink (UHS). The UHS sources of cooling or make-up water include:

- *sea water* - saline, high biological activity, periodic high particulate and detritus, with a temperature range of approximately 30°F to 85°F
- *lake water* - fresh water, biologically active, low to moderate particulate and/or detritus, temperature range 30°F to 95°F
- *river water* - fresh to brackish water, biologically active, low to very high particulate and/or detritus, temperature range 30°F to 100°F
- *cooling pond/tower* - fresh water of low turbidity, low to moderate biological activity, pH-controlled, temperature range 30°F to 95°F

Flow velocities within the system can range from stagnant (for possibly several months at a time) to velocities in excess of 40 ft/s at or near throttling devices.

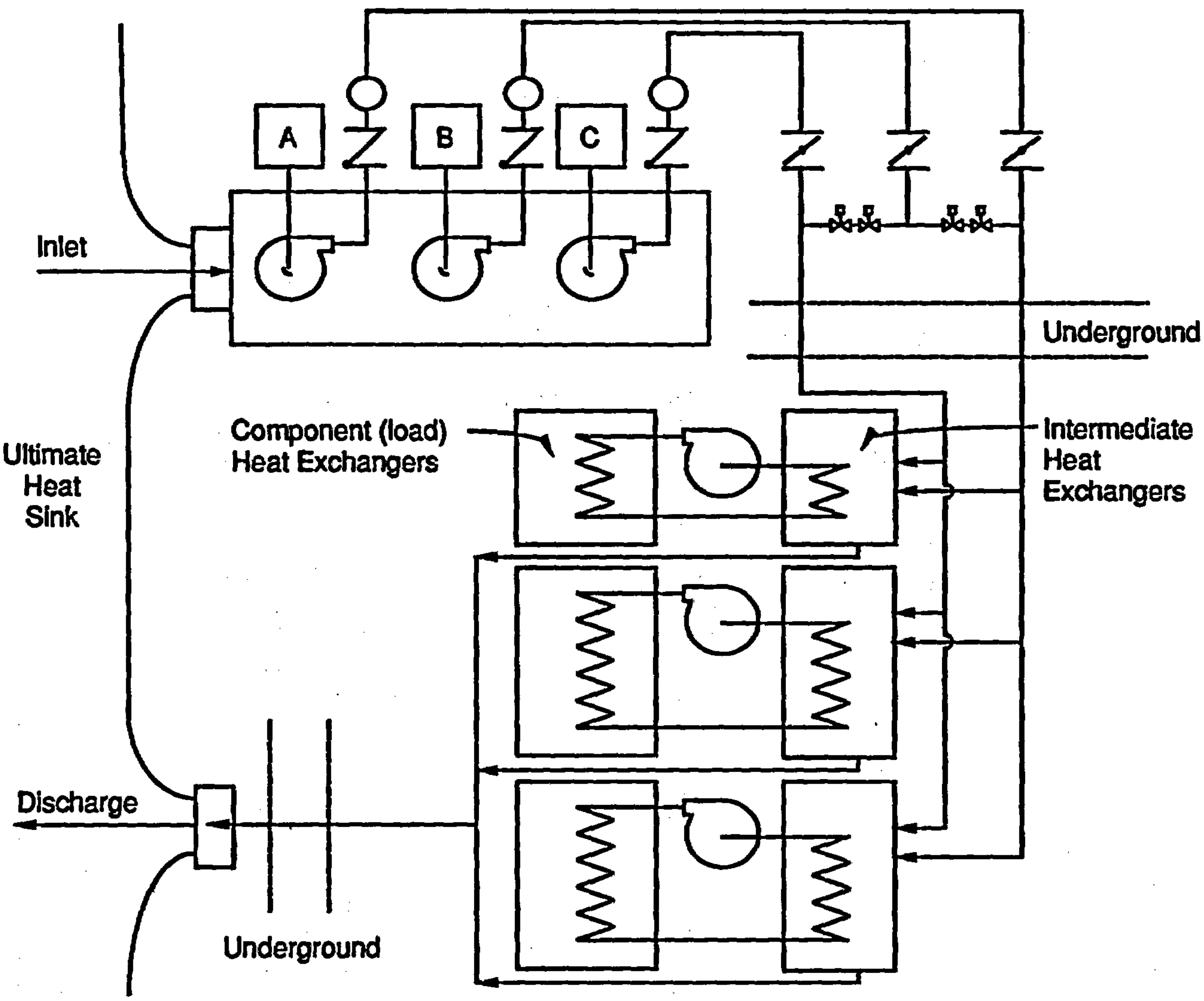

Figure 2.3 Closed Service Water System

The predominant construction material for SWSs is carbon steel. Wide variations in materials exist from plant to plant depending on the degree of agressiveness of the cooling fluid. Other commonly used materials are copper and copper-based alloys, monel (a nickel-copper alloy), and stainless steels (300 to 400 series). Copper-base alloys are widely used in heat exchanger applications because of inherently good corrosion resistance combined with good mechanical properties, excellent thermal conductivity, and ease of soldering or brazing. Stainless steels, particularly types 304 and 316, are used increasingly as nickel-base replacement materials to improve corrosion resistance.

2.4.2 Component Function and Process Flow Path

While construction arrangements, cooling water conditions, and materials used differ widely among plants, the design of each functional area of all systems is basically the same. The following generic system description is typical of an open plant configuration.

The intake structure admits water to the system, provides for the necessary degree of debris removal, and houses the service water pumps and their associated

Table 2.1 U.S. Nuclear Plant Service Water System Summary[a]

Plant Name	Reactor/ Rating	Parent Utility	Ultimate Heat Sink	Service Water Cycle
Clvrt Clfs-1,2	(CE-845)	Baltimore Gas & Elec	Chesapeake Bay	Closed
Cook-1	(W-1030)	Indiana & Mich Elec	Lake Michigan	Closed
Cook-2	(W-1090)	Indiana & Mich Elec	Lake Michigan	Closed
Fermi-2	(GE-1093)	Detroit Edison Co	Hyperbol/Erie	Closed
Fort St Vrain-1	(GA-330)	Colorado Public Serv	FDT/R	Closed
Hope Creek	(GE-1067)	Public Service Elec	Hyperbol/Delawar	Closed
Limerick-1,2	(GE-1055)	Philadelphia Elec Co	Hyperbol/Schukil	Closed
Maine Yankee	(CE-790)	Maine Yankee Atomic	Back R/ATL O	Closed
Pilgrim-1	(GE-655)	Boston Edison Co	Cape Cod Bay	Closed
Rancho Seco-1	(B&W-916)	Sacramento Muni Util	Hyperbol/Canal	Closed
River Bend-1	(GE-940)	Gulf States Util	FDT/Mississippi	Closed
Salem-1	(W-1090)	Public Serv Elec NJ	Delaware R	Closed
Salem-2	(W-1115)	Public Serv Elec NJ	Delaware R	Closed
San Onofre-1	(W-436)	Southern CA Edison	Pacific Ocean	Closed
San Onofre-2	(CE-1070)	Southern CA Edison	Pacific Ocean	Closed
San Onofre-3	(CE-1080)	Southern CA Edison	Pacific Ocean	Closed
Shoreham	(GE-819)	Long Island Lighting	L I Sound	Closed
St Lucie-1,2	(CE-810)	Florida P&L	Atlantic Ocean	Closed
Waterford-3	(CE-1165)	Louisiana P&L	Mississippi R	Closed
Diablo Canyn-1	(W-1084)	Pacific Gas & Elec	Pacific Ocean	Closed
Diablo Canyn-2	(W-1106)	Pacific Gas & Elec	Pacific Ocean	Closed
Palo Vr-1,2,3	(CE-1270)	Arizona Public Serv	Basin Filter-Well	Closed Pond
WNP-2	(GE-1100)	WA Pub PWR Supply	FD/Columbia	Closed Pond
Byron-1,2	(W-1120)	Commonwealth Edison	Hyperbol/Rock	Closed River
Vogtle-1(2)	(W-1125)	Georgia Power Co	FDT/Savannah R	Closed Well
So Tex Proj-1,2	(W-1250)	South Texas Project	Reser./Colorado	Open/Closed
Big Rock Point	(GE-71)	Consumers Power Co	Lake Michigan	Open Lake
Catawba-1,2	(W-1145)	Duke Power Co	FDT/Lake	Open Lake
Clinton-1	(GE-955)	Illinois Power Co	Lake Clinton	Open Lake
Davis-Besse-1	(B&W-880)	Toledo Edison Co	Tower/L Erie	Open Lake
Dresden-2,3	(GE-794)	Commonwealth Edison	Cooling Lake	Open Lake
Fitzpatrick	(GE-821)	PWR Auth State of NY	Lake Ontario	Open Lake
Ginna	(W-470)	Rochester Gas & Elec	Lake Ontario	Open Lake
Kewaunee	(W-535)	Wisconsin Pub Servic	Lake Michigan	Open Lake
Lasalle-1,2	(GE-1078)	Commonwealth Edison	Reservoir	Open Lake
McGuire-1,2	(W-1180)	Duke Power Co	Reservoir	Open Lake
Nine Mile-1	(GE-610)	Niagara Mohawk Power	Lake Ontario	Open Lake
Nine Mile-2	(GE-1080)	Niagara Mohawk Power	Hyperbol/Lake M/U	Open Lake
North Anna-1,2	(W-890)	Virginia Elec & PWR	Reservoir/L Anna	Open Lake
Oconee-1,2,3	(B&W-860)	Duke Power Co	Reservoir	Open Lake
Palisades	(CE-798)	Consumers Power Co	FDT/Michigan	Open Lake

Table 2.1 (Continued)

Plant Name	Reactor/ Rating	Parent Utility	Ultimate Heat Sink	Service Water Cycle
Perry-1,2	(GE-1205)	Cleveland Elec Illum	Hyperbol/L Erie	Open Lake
Point Beach-1,2	(W-497)	Wisconsin Elec PWR	Lake Michigan	Open Lake
Robinson-2	(W-665)	Carolina P&L	Res/Robinson	Open Lake
Summer	(W-900)	S Carolina Electric	Res/Monticello	Open Lake
Wolf Creek	(W-1150)	Kansas Gas and Elec	Cooling Lake	Open Lake
Zion-1,2	(W-1040)	Commonwealth Edison	Lake Michigan	Open Lake
Brunswick-1,2	(GE-790)	Carolina P&L	ATL O Outfall	Open Ocean
Crystl River-3	(B&W-825)	Florida Power Corp	Gulf of Mexico	Open Ocean
Millstone-1	(GE-660)	Northeast Utilities	L I Sound	Open Ocean
Millstone-2	(CE-830)	Northeast Utilities	L I Sound	Open Ocean
Millstone-3	(W-1150)	Northeast Utilities	L I Sound	Open Ocean
Oyster Creek	(GE-620)	Jersey Central PWR	Barnegat Bay	Open Ocean
Seabrook-1	(W-1150)	Public Serv Co of NH	Atlantic Ocean	Open Ocean
Turkey Pnt-3,4	(W-728)	Florida PWR & Light	Canal/Biscayne Bay	Open Ocean
Callaway-1	(GE-1150)	Union Electric Co	Hyperbol/Missouri R	Open Pond
Sharon Harris-1	(W-900)	Carolina P&L	Hyperbol/Reservoir	Open Pond
Yankee-ROWE	(W-175)	Yankee Atomic Electr	Deerfield R	Open Pond
ANO - 1	(B&W-836)	Arkansas PWR & Light	Reservoir, Ark R	Open River
ANO - 2	(CE-858)	Arkansas PWR & Light	Hyprbol/R M/U	Open River
Arnold	(GE-538)	Iowa Elec Light	FDT/R	Open River
Beaver Valy-1,2	(W-833)	Duquesne Light Co	Hyprbol/Ohio R	Open River
Belfont-1(2)	(B&W1213)	TVA	Hyperbol/Tenn R	Open River
Braidwood-1,2	(W-1120)	Commonwealth Edison	Res/Kankakee R	Open River
Brwns F-1,2,3	(GE-1065)	TVA	Combcycle Tenn R	Open River
Coman Peak-1,2	(W-1111)	Texas Util Gen Co	Res/R	Open River
Cooper	(GE-778)	Nebraska Public PWR	Missouri R	Open River
Ct Yankee	(W-582)	CT Yankee Atomic PWR	Connecticut R	Open River
Farley-1,2	(W-860)	Alabama Power Co	FDT/R	Open River
Ft Calhoun-1	(CE-478)	Omaha Pub. PWR Dist	Missouri R	Open River
Grnd Glf-1(2)	(GE-1250)	Mississippi PWR & Li	Hyperbol/Mississippi	Open River
Hatch-1,2	(GE-770)	Georgia Power Co	FDT/R	Open River
Indian Point-2	(W-873)	Con Ed Co of NY	Hudson R	Open River
Indian Point-3	(W-965)	PWR Auth State of NY	Hudson R	Open River
La Crosse	(GE-48)	Dairyland PWR Coop	Mississippi R	Open River
Monticello	(GE-536)	Northern States PWR	FDT/Mississippi	Open River
Peach Bot-2,3	(GE-1065)	Philadelphia Elec Co	FDT/Susquhanna	Open River
Prairie I.-1,2	(W-507)	Northern States PWR	FDT/R	Open River
Quad City-1,2	(GE-789)	Commonwealth Edison	Spray Canal	Open River
Sequoyah-1,2	(W-1148)	TVA	Comb Cycle/Tenn	Open River
Surry-1,2	(W-775)	Virginia Elec & PWR	James R	Open River
Susquehana-1,2	(GE1050)	Pennsylvania P&L	Hyperbol/Susquhan	Open River
TMI-1,2	(B&W-792)	Metropolitan Edison	Hyperbol/Susquhan	Open River
Trojan	(W-1130)	Portland General Ele	Hyperbol/Columbia	Open River

Table 2.1 (Continued)

Plant Name	Reactor/ Rating	Parent Utility	Ultimate Heat Sink	Service Water Cycle
Vermont Yankee	(GE-514)	VT Yankee Nuc PWR Co	Towers/Conn R	Open River
Watts Bar-1,2	(W-1170)	TVA	Hyperbol/Tenn R	Open River

(a) Information is extracted from final safety analysis reports (FSARs) and is subject to change due to plant modifications.
FDT - forced draft cooling tower
hyperbol - hyperbolic natural draft cooling tower
R - river
L - lake
Res - reservoir

switchgear. The open system, shown in Figures 2.2a and 2.2b consists of identical bays housing bar gates, stop logs, traveling screens, screen wash pumps, chlorine injection equipment, and the circulating and service water pumps. An elevation drawing, Figure 2.4, shows a typical arrangement of these components. A separate ventilation system for the service water pump area is designated seismic Class I to ensure that the equipment remains operable during a DBA and is capable of functioning afterward. The service water pumps and their associated bays are also designated Class I seismic equipment.

The traveling screens, in parallel arrangement, function to remove small debris that penetrates the larger (≈3-in. opening) straining provided by the bar gates. The screens are cleaned by an automatic high-pressure spray system that senses differential pressure across the screens.

Three service water pumps are typically used to provide the necessary head for flow requirements of the various heat exchangers in the system (from two to four pumps per plant were observed). Normal operation means that two of the three SWS pumps operating supply all steady-state cooling requirements, with the third pump in a standby condition (one out of two and three out of four is also common). Pump motor power ranges from approximately 200 to 600 horsepower and flow from 4000 to 10,000 gallons per minute per pump, depending on configuration and plant size. Pumps are powered from plant vital ac busses (with diesel backup) to ensure a continuous electrical supply. Pump motor-winding temperature, current, discharge pressure, and bearing temperature instrumentation are generally available at local and remote readout panels.

At the pump discharge, common practice is to pass the water through a discharge strainer to ensure that any remaining particles are small enough so they will not plug the smallest heat exchanger tube in the system. These strainers are monitored for excess differential pressure, which could indicate a plugging condition. The swing check valves located just downstream of the strainers are installed to prevent reverse flow through an idle pump and strainer. Reverse flow short-circuits the intended path through the heat exchanger resulting in a low flow condition, and may cause idle pump damage.

The cross-tie valves located on the discharge header permit full operation of both service water loops with any two SWS pumps running; these are motor-operated valves (MOVs) powered from the plant vital bus. It is common practice to provide for emergency make-up to the reactor vessel or steam generators from the SWS because it is engineered as a highly reliable water source. Redundant headers then provide the necessary flow distribution network to the individual cooling loads. The discharge header then collects the system flows and directs them back through the cooling water flume. Some arrangements allow make-up to the emergency cooling water reservoir from the discharge header.

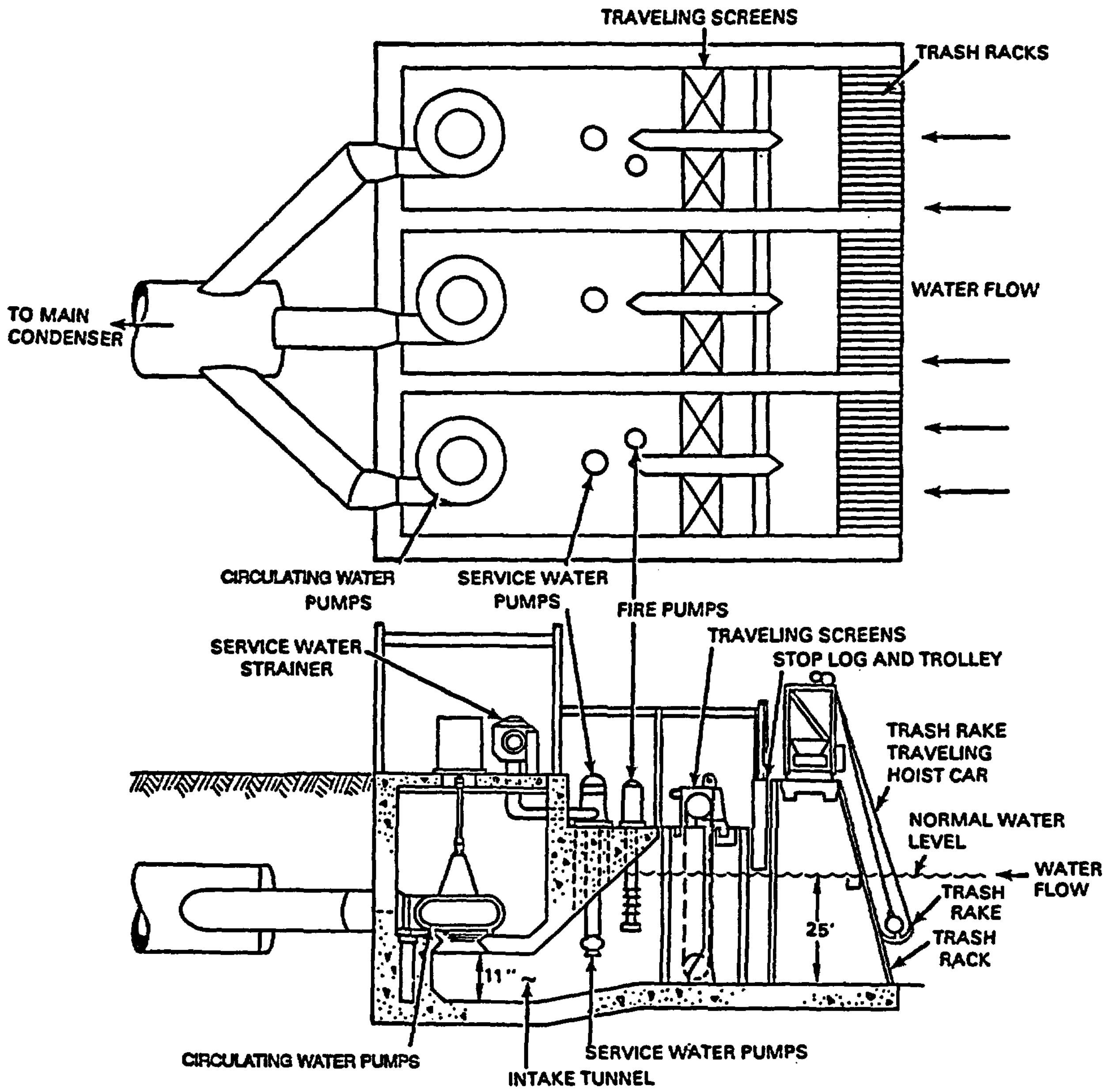

Figure 2.4 Service Water Intake Structure

The emergency cooling water source (river, cooling pond, ocean, etc.) may take various forms depending on local conditions, but it must meet the requirements of Regulatory Guide 1.27 (NRC 1976) with regard to capacity, availability, and accessibility under seismic conditions.

3 Data Acquisition and Compilation

This chapter discusses the various sources of data for the study and the methods used to extract the information required. It then gives an overview of the observations made and draws some qualitative conclusions based on this compilation.

3.1 Data Sources

The sources of component degradation and failure information documented in the Phase I investigation were used and extended to develop the Phase II research plan. These information sources and how they were used are explained in the following subsections.

3.1.1 Database Information

In Phase I an attempt was made to use the available component failure information found in several large databases. Discrepancies between the information found in these sources and the more intuitive knowledge of researchers, who had experience with the operating environment of SWSs, prompted a search for consistent and reliable data. While we understand that a concerted industry effort is ongoing to upgrade the quality and quantity of information in such databases, the usefulness of this data will be subject to question until uniform reporting requirements are fully implemented. Generic database information was not used in the second phase of this research.

3.1.2 Site Visits

The purpose of the site visits was to collect plant SWS descriptive information and site personnel knowledge and insights into system operation and degradation phenomena. Site visits were conducted by a team of PNL engineers at three plants, each plant one of the three different SWS types: open (once-through), recirculating (spray pond), and closed (intermediate heat exchanger).

3.1.3 Nuclear Regulatory Commission Data

This information consisted of 1) the previously mentioned AEOD report *Operating Experience Feedback Report: Service Water System Failures and Degradations* (Lam and Leeds 1988), 2) information available through inspection and incident investigation reports, and 3) direct knowledge from PNL participation in SWS investigations.

3.1.3.1 AEOD Degradation Report

This report is a compendium and analysis of operating events involving SWS failures and degradations in LWRs from 1980 to 1987. The analysis indicates that the failures and degradations are multicausal and can severely impact the safety systems that are required to mitigate reactor accidents. The information is drawn from LERs (documents prepared by the utilities to explain the circumstances surrounding the failure of a safety-related component), as well as from NRC Bulletins, NRC Information Notices, and Reports to Congress on Abnormal Occurrences. Additionally, information from AEOD technical study reports, industry component failure databases, NRC inspection reports, and data-gathering visits to four plant sites was included.

A total of 980 events was evaluated, with 276 of these events being judged safety-significant by the NRC staff.

3.1.3.2 Incident Investigation Reports

These reports evaluate nonroutine plant events or accidents considered to have a potential impact on reactor safety, either through component failure or a more implicit scenario (managerial breakdown). This evaluation may take the form of a fairly detailed root-cause analysis and can therefore shed more light on occurrences than generic data. These reports are not, however, designed to provide information specific to aging research and cover only a small fraction of the plant failure inventory. For these reasons, their use in this task was somewhat limited.

3.1.3.3 Participation in Investigations

As a result of experience gained through this aging assessment task, PNL has developed a system-specific expertise that has been used by regional NRC staff to provide technical assistance in investigating power reactor SWS- related problems. The problem area, component failure, and the root cause of the system fault were found to provide useful information and have been added to the SWS degradation knowledge portfolio. (See Section 4.3 for case study information.)

3.2 Data Acquisition

The success of obtaining reliable and verifiable information on component failure history and the root cause of degradation depends on posing the appropriate questions to the appropriate personnel. The knowledge acquisition methodology developed in Phase I of this project derived considerable validity from accessing the information gained by plant personnel and their first-hand experience at operating and in maintaining the SWS.

3.2.1 Questionnaire Protocol

As in Phase I, the questionnaire protocol, developed for recording previously undocumented plant knowledge, was used at each site to ensure complete and efficient data gathering. Information was collected on all the major components in the SWS, including information relating to the operation, regulation, modification, surveillance, and maintenance of the SWS.

The completed protocol includes all the detailed plant knowledge collected during the visit from key plant personnel, including design engineering, operating supervision, equipment operators, chemists, biologists, maintenance foremen, and maintenance personnel. This information, more than other sources, provides considerably deeper insights into failure occurrences by taking advantage of the inherent root-cause analysis performed by plant personnel.

3.2.2 Site Information Resources

Information pertaining to potential degradation stressors at each of the SWS types was obtained by completing the prepared protocol and by investigating and following up on other prospective sources revealed during the site visit.

To accomplish the knowledge and data-gathering objectives, formal and informal meetings were held with site personnel. Formal meetings were usually confined to entry- and exit-level meetings with the appropriate site manager. Informal meetings were held with small groups of individuals from a specific department or function, or with single individuals having special SWS knowledge and experience.

Site personnel provided tours of selected areas of the site by request or volunteered tours when a special maintenance or surveillance activity was in progress. The volunteered tours afforded an opportunity to inspect areas of the SWS which were normally closed or inaccessible and also in many cases, an opportunity to obtain specimens and chemical samples for further study. Both casual and pointed questions were asked of the tour guide and of personnel working in the tour area, and the candid responses often provided information as well as topics for additional investigation.

The plant databases and historical files were searched, and complete operating and maintenance histories were obtained for individual SWS components. Where system modifications had been performed on some portion or component of the SWS, the data on the original and the modified configuration were reviewed, as well as data pertaining to unique and unusual events affecting the SWS. The events include unanticipated changes in biofouling or water chemistry caused by environmental changes, water hammer events, and "one time only" SWS cleaning activities.

Other appropriate areas were reviewed and, where applicable to a specific utility, system aging information was obtained. Areas investigated at selected utilities included observing the SWS plant operation, witnessing a routine SWS component maintenance activity, and observing scheduled testing.

Information obtained from these various site resources was used to complete the protocol and to supply data for use in the degradation analysis.

3.3 Observation Summary

While information was being collected, a series of observations was made that provide some insights into the problems encountered in operating each type of system. While these statements are more impressions of plant conditions, as formed by the research team members, rather than rigorous data, they nevertheless convey important information that could be valuable in assessing similar situations. These observations are offered as a collective experience of the site visit teams.

3.3.1 Open System Observations

Plant-Specific Environment: river water with high turbidity and organic content.

Corrosion and biofouling are the most common causes of open SWS problems encountered by the site visit team. To maintain required component temperature limits, many components of the open SWS system are throttled or intermittently operated, resulting in low flow velocities, or in many cases, completely stagnant loops. Throttling and intermittent operation accelerates solids deposition and various forms of corrosion in these areas.

Stainless steels are used in an attempt to correct many of the corrosion problems; however, not all stainless steels are immune to the corrosive environment. Stainless steels in a stagnant water environment tend to degrade, because dissolved oxygen in water is necessary to maintain the protective oxide layer on the pipe wall. Valve stems made of 410 stainless steel have been corroding in this environment. By maintaining a minimum flow of water where stainless steel is present, oxygen depletion will be minimized and stainless steel corrosion will be mitigated. Maintaining an adequate dissolved oxygen content in the water also retards nickel leaching of 90/10 copper nickel tubing commonly found in SWS heat exchangers.

Where maintaining a minimum flow of water through the stainless steel pipe eliminates the oxygen depletion problem, it exacerbates carbon steel corrosion. Some large-bore carbon steel pipe developed small leaks caused by localized pitting. Small-bore carbon steel pipe also developed small leaks, but more often this condition was observed in the piping U-joints of component coolers. Carbon steel valve internal components suffered obvious degradation due to corrosion.

The greatest number of electrical failures experienced were found with the operators on MOVs. Most are corrosion related. External surfaces of buried piping, which have a protective barrier coating, show evidence of corrosion but little significant degradation.

Corrosion products, primarily from carbon steel, have caused many other less obvious problems; often these problems are diagnosed as having causes other than from corrosion products. Corrosion products have caused some flow blockage in small tubed heat exchangers and have resulted in plugged instrument lines. Often, greater torques required to operate system valves and increased pump motor currents to maintain design conditions are caused by the effects of corrosion products.

In contrast to corrosion, fewer erosion problems occur in open SWSs. With controlled low flow velocities, typically below 5 ft/s, erosion is not of consequence in piping or SWS components including heat exchangers. Even with low flow velocities in system piping, high flow velocities are experienced within SWS pumps and at throttling valves. Erosion was observed on pumps with brass impellers and required impeller replacement about every two years. Fire water systems, using the same water as the SWSs, experience erosive degradation of pump impellers and the coarse screens located in the pump suction bell.

Changes in environmental conditions are a constant threat to increased flow blockage erosion in open, once-through SWSs. Any environmental event causing increases in silt intrusion into the SWS results in increased erosion.

In addition to corrosion, biofouling and biofouling control are challenges inherent in the design of open once-through SWSs. Biofouling from microorganism growth tends to decrease heat exchanger efficiency, cause corrosion of base metal, reduce system flow, and pose the risk of fouling as large colonies of microorganisms slough off from piping walls.

Larger organisms, such as fresh water clams (corbicula), also present biofouling challenges and cause corrosion, including local erosion and pitting, similar to that observed from microorganisms. Plugging and fouling are the most prevalent difficulties encountered. Even with chlorination, fresh water clam removal from the SWS bays is often required. Biofouling of an 18-in. pipe between the ultimate heat sink and the SWS required hydro-lancing to restore the pipe to the required flow rate.

Other environmental debris can cause mechanical stresses exceeding those routinely expected to be found in SWSs. For example, because of the design of open, once-through SWSs, pieces of wood or similar debris enter the system and have been observed to lodge between moving and stationary parts. In one instance, large volumes of wood, leaves, and debris limited power production because the physical capacity of the traveling screens was exceeded. Similarly, periodic fish migrations have resulted in flow reductions by overloading traveling screen capacity.

The major source of service water pump failure is bearing degradation. Trending of pump vibration has been a good predictor of bearing failure.

3.3.2 Recirculating System Observations

Plant-Specific Environment: arid conditions with mild river water makeup.

Corrosion is the most common problem in recirculating SWSs as noted by the site visit team. In terms of severity and quantity, erosion and sediment problems are few compared with those in open systems and generally affect limited portions of the recirculating system. Biofouling is limited to some slime and algae formations.

Corrosion, although the most common cause of problems in the recirculating SWS, is less severe and more predictable than in other open portions of SWSs. Water treatment products and the addition of scale and corrosion inhibitors have been effectively used to control corrosion. Chemical additions to the system to control pH are infrequent because of the nonaggressive nature of the make-up water source.

Before corrosion control was implemented, all small-bore carbon steel pipe suffered significant corrosion and sediment buildup so that in some cases the pipe bore became completely blocked. Portions of the original carbon steel piping were then replaced with similar piping. Following the change in make-up water sources and introduction of corrosion control, carbon steel pipe corrosion has been minimal.

Ring header supports for pond spray rings have sustained some pitting attacks, resulting in broad shallow areas of slight metal removal. Magnesium anodes are used in the ponds to protect submerged steel structures.

Underground piping inspected during an outage showed light exterior corrosion and no significant degradation. In other underground piping, some corrosion was evident where water seeped under the protective wrapping. This corrosion, though evident, is minor and has not compromised the integrity of the underground piping. A cathodic protection system is installed on some underground piping, but the system has not been used.

A division diesel generator heat exchanger was inspected when it was opened for maintenance. The visual inspection revealed some corrosion and sedimentation accumulation with a slight slime overlay. The corrosion was uniform across all surfaces and did not appear to be a problem. Some corrosion tubercles, up to approximately 1 in. in diameter and 1/2-in. high, were found in flow eddy areas. No corrosion-caused thermal performance degradation had been identified in diesel generator heat exchangers.

Sediment, which had accumulated in the spray ponds during six years of operation, had stayed below the allowable technical specification limit. Vacuuming techniques were used to remove accumulated sedimentation.

No significant flow blockage has occurred in heat exchangers because of sedimentation. In general, flow reduction caused by sedimentation has been in large cross-sectional areas of piping runs where low-fluid velocities exist. SWS modifications to reduce or mitigate sedimentation by increasing fluid velocities have been successful.

In general, fluid flows are slow enough that erosion is not a problem. No significant erosion was observed in piping or heat exchanger tubes. Some cavitation erosion was observed on SWS discharge butterfly valves. Replacing the butterfly valves with hardened steel components is expected to mitigate the cavitation erosion.

Biofouling in the recirculating SWS is not a significant problem. With a spray pond temperature range of 32°F to approximately 77°F and no biocides currently used in the system, some biological activity is expected. Live algae bloom has been noted at certain times of the year in the spray pond bulk water. A slight slime overlay was observed in some piping and on the surface of heat exchanger tubing. Biocidal treatment has been implemented in the pond.

Although active Asiatic clam populations are known to exist at the river make-up water intake, no evidence of clams, clam debris, or clam larvae have been found in the SWS.

After 5 years of operation, eddy current testing was performed on three SWS heat exchangers. Results of the examination revealed that it was not necessary to plug tubes for any structural deficiencies.

Vibration, usually from motor operation, has caused air line leaks in air-operated valve actuators. The leaks occur at the tubing compression fittings at the valve actuators in proximity to the motors.

A water hammer problem with the SWS was a problem during startup. The problem has been mitigated by adding a keep-full or jockey pump and by motor-operated valve sequencing that starts each pump against a highly throttled discharge valve. All short- and long-term consequences from the water hammer problem are believed eliminated by the plants experiencing the problem.

Instrumentation flow orifices, venturis, and sensing lines have not experienced any plugging problems since replacing small-bore carbon steel pipe and adding corrosion inhibitors.

Pump motors in the SWS are routinely monitored and surveillance tested. It has not been necessary to disassemble the motors for inspection because of adverse monitoring results or surveillance data.

3.3.3 Closed System Observations

Plant-Specific Environment: salt water marine with high ambient humidity.

The most common causes of closed SWS problems observed by the site visit team were corrosion and biofouling, and the greatest number of modifications to any of the SWS types was observed with the closed system. Most of the modifications in progress are to correct or mitigate corrosion problems that were developing on the raw water side of the SWS. A significant number of modifications are planned to correct biofouling.

Two corrosion problems are evident at the intake structure: concrete pedestal cracking due to rebar corrosion and aluminum embedment failure caused by corrosion from the marine environment. Other indicated degradation includes long-term degradation of the structure due to rebar corrosion, and attack from marine and groundwater chemicals on the concrete binders and cements.

The SWS pumps, having accumulated about five years of operation and being submerged for about seven years, were removed for routine examination. After removing the surface oxidation coating, the pumps were found to be in extraordinarily good condition. The pumps' pristine condition can be attributed to an effective cathodic protection system used to control the corrosive marine environment at the submerged pumps. However, corrosion was observed under the pump shaft sleeves during the routine inspection, necessitating replacement of the lower pump shaft.

A water hammer problem noted during startup was corrected by a combination of administrative and hardware modifications. Plant personnel believe that all short- and long-term consequences from the water hammer problem have been eliminated. Accelerated aging and degradation effects resulting from the water hammer problem are unknown at this time.

In many of the non-safety-related piping, carbon steel used in the drains and the intake screen wash system has been replaced because of the effects of corrosion and biofouling. Inspection of the removed pipe spools revealed uniform corrosion with biofouling of the inner pipe wall. Through-wall failures have occurred where corrosion was localized; these locations are difficult to predict from external examinations of pipe. That pipe has been almost totally replaced with lined or stainless steel pipe.

Non-wetted valve parts have failed under stress-induced corrosion. In some cases, corrosion has caused valve inoperability, and in others, catastrophic valve failure. A series of failures, due to corrosion of ductile iron packing followers caused by the damp marine environment, valve leakage, and possibly other factors, occurred on some SWS valves.

Butterfly valves in the SWS have failed because of corrosion between the monel shaft and the packing follower. The packing followers were carbon steel and, as the corrosion process proceeded, carbon steel metal oxides were formed (which are volumetrically larger than the original carbon steel) and eventually caused binding of the operating shaft.

The check valves were showing signs of premature wear, primarily in the valve seat area and the check disk. New elastomer seals were installed in the seat area and the check disk was painted with a protectant, thus reducing the rate of degradation to an acceptable level until the planned modification to replace the existing check valves with aluminum-bronze valves is completed.

Valve body to valve-liner adhesion is a problem. Loss of the valve liner integrity is not obvious during normal operation and once lost, irreparable corrosion rapidly ensues.

The heat exchanger copper-nickel tubes were replaced with titanium alloy tubes selected specifically to resist salt water chemistry.

Corrosion of electrical equipment is evident but not significant, primarily because it is predictable. Electrical equipment protected by coatings or water-tight enclosures show minimal or no visual degradation. Surface corrosion is evident on exposed wire, cable, and electrical contacts, as well as on electrical conduits, boxes, and enclosures. Surface corrosion is quite visible on motor cables not protected by insulation or coatings. No failures due to corrosion were identified in electrical power circuits at the utility visited.

The premature failure of the rubber/elastomer expansion joints at the SWS heat exchangers is an excellent example of air-induced aging degradation caused by improper installation. This conclusion, reached by both the plant's engineering staff and the component manufacturer's engineering representative, was supported by evidence presented to the site visit team. The failure is a series of longitudinal cracks or tears in the expansion joint that coincide with each of the bolt holes in the expansion joint mounting flange. The hypothesis is that during installation an oversized torque or similar wrench was used which, when twisted, placed a severe stress on the elastomer material, resulting in surface degradation. The degradation was not immediately observable but was manifested only after repeated operating cycles. Replacement by proper installation solved the problem.

Preventing biofouling is a never-ending battle against the marine environment because the SWSs are assaulted by vegetation, microorganisms, barnacles, mollusks, and other aquatic life forms. The shells of marine organisms, primarily barnacles, are continuously swept into the heat exchangers, resulting in tube plugging. These barnacles must be routinely removed to maintain the flow rate required to mitigate postulated accident conditions.

Heat treatment, chlorine injection, and manual cleaning are common biological control processes used by the utilities. Heat treatment (thermal backflush), with or without chlorine, is performed on a routine basis as determined by the rate of organism growth. Chlorine is injected at regular intervals, usually several times a week. Manual cleaning is performed during refueling outages, or as necessary to maintain required system flow rates.

A continuous cycle of SWS cleaning and inspection for degradation is necessary; any identified degradation is then repaired or replaced. However, long-term aging trends can only be estimated because the cleaning/replacement cycle is continuous. As the time between cleaning and inspection is extended (possibly due to system modification or process changes), the effects of aging and degradation stressors on the SWS must be closely monitored until the long-term trends are known and predictable.

4 Root-Cause Methodology

In Phase I of the SWS study a lack of accurate, uniform root-cause analysis (RCA) was shown to lead to incorrect conclusions regarding the degradation mechanisms that result in component failure. In response to this identified need, a subtask was organized to provide a prototype method for identifying and documenting the root cause of component degradation and failure. This section discusses the background and development of this proposed methodology for performing RCA. The approach, methodology, and categorization of a generic root-cause scheme are explained.

Section 4.1 explains the regulatory basis for the actions being taken by the NRC in the Generic Letter on Service Water System Problems Affecting Safety-Related Equipment (54 FR 43209-10). It also presents the impetus for the definition of a uniform methodology for RCA. Section 4.2 explains the root-cause process, the method structure, and a root-cause categorization based on component design considerations.

4.1 Task Development

The safety-related portions of the nuclear plant SWS are the final link in the pathway of heat dissipation to the ultimate heat sink; for this reason, it was selected for an NPAR system study. The SWS draws water from various sources, and depending on the plant, is composed of a host of materials, differing configurations, and a high variability in operating and maintenance practices.

Historically, maintenance performance involves a balance among the preventive (replace before failure), the predictive (measure and replace before loss of capability), and corrective (repair/replace after the failure) measures predicated principally on economic considerations, i.e., cost effectiveness. For nuclear power, however, operating safety takes on a dominant role because of the potential impact of an accidental release of radioactivity on the general public. The safety issue, coupled with the high cost of replacement power during plant shutdowns, has led to an increased bias toward emphasizing preventive and predictive maintenance programs.

4.1.1 Safety-Related Event Reporting

A system for rapid reporting of significant events to the NRC has been established and is depicted in Figure 4.1. Like the LCOs, these events are also grouped by a perceived increase in potential core damage that could result from the event. The SWS-related events fall under "degradation in emergency response capability" in the one-hour category and "conditions that could prevent residual heat removal" in the four-hour notification group.

For any of the events shown in Figure 4.1, an LER must be issued by the plant within 30 days after the discovery that such an event took place. Per Section 50.73 of 10 CFR 50, part of this report must contain the following information to incorporate all the essential elements of a failure RCA:

- a complete event scenario description
- pre-event plant conditions
- inoperable components that contributed to the event
- cause of component failure
- failure mode, mechanism, and effect
- estimate of safety system unavailability
- discovery method.

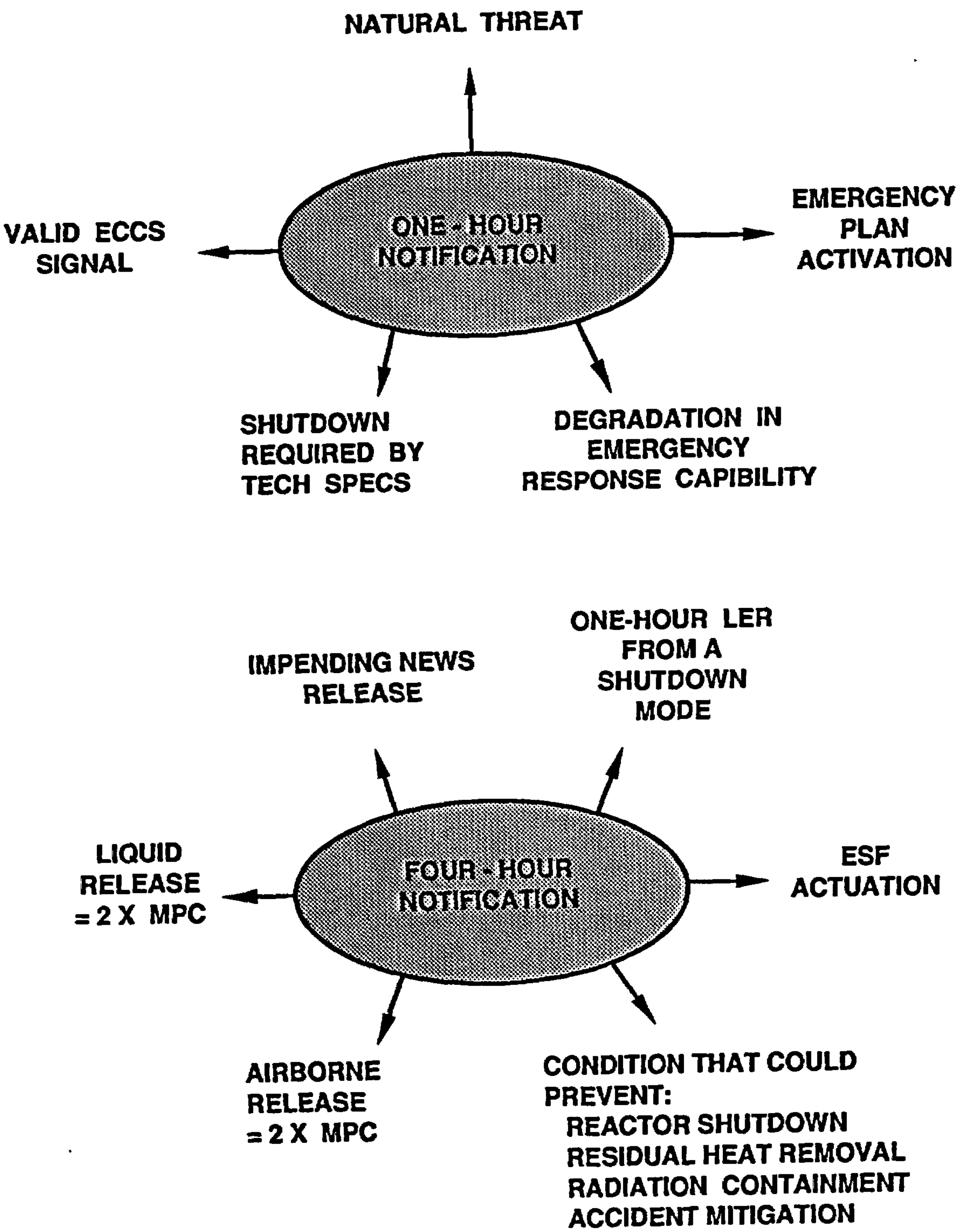

Figure 4.1 U.S. Nuclear Regulatory Commission Reportable Events

4.1.2 Component Failure Root Cause

To better understand the search for the failure root-cause, the following definitions are provided:

- failure mode - the manner or state in which a system, structure, or component fails. For example, a heat exchanger can be declared inoperable (to have failed) if it:
  - does not have pressure boundary integrity (a leak),
  - restricts the flow of working fluid to less then a required limit (a block) or,
  - does not provide the required heat transfer capacity under minimum design flow conditions (degraded heat transfer coefficient).
- aging mechanism - specific process that gradually changes the characteristics of a system, structure, or component with time or use. Example: A deposition process that reduces the ability of a heat exchanger to pass the required flow rate.
- stressor - an agent or stimulus that can produce aging mechanisms. For the above cited deposition process, the stressor might be high silt content in the process stream that settles out (deposits) in low flow areas of the heat exchanger.

Component failure root cause is thus "the process of determining the fundamental degradation stressors responsible for component failure, such that, if corrected, prevents recurrent failure from that mechanism" (Jarrell et al. 1989; Stratton and Jarrell 1989). This clearly demonstrates that the utilities are required by 10 CFR 50 to perform a failure RCA for a wide range of events that affect safety-related systems. It is the goal of this RCA research study to provide a general framework defining the process requirements for performing such an investigation.

The prime goal of the NPAR program is to identify and characterize aging mechanisms that could cause safety-related components to undergo premature or undetected degradation. Figure 4.2 illustrates an abbreviated version of the principal products of the program. The program's investigative approach is to collect technical information from various sources pertaining to a specific safety-significant component or system, then analyze it in a manner that allows correlations to be made regarding the identity and characteristics of degradation in the component (system) of interest. A determination of the measurement accuracy and the completeness of the degradation effects on component (system) performance relative to the design criteria is then made to ensure safety function capability. Any discrepancies found between the design criteria and the performance measurement are then documented in a recommendation for a revision to the regulatory requirements.

4.2 Development of a Generic Methodology

The need for a structure to promote a more uniform approach to RCA and reporting grew from efforts to define equipment-aging safety issues. This section traces the evolution of that need as perceived by the Nuclear Plant Service Water System Aging Degradation Assessment (Jarrell et al. 1989), and presents the logic used in the analytical approach and the actual elements of the RCA model derivation. The source of data most commonly used by NPAR researchers for determining the degradation and failure mechanisms is utility-generated data from a central data repository. Several of these aging studies indicate that the component failures listed in such databases are inadequately specified and/or inaccurate when analyzed for component degradation mechanisms (Murphy et al. 1984). Since these studies are to be at least contributing factors in mandated utility maintenance and surveillance requirements, the impact of inaccurate degradation assessment can be manifested as 1) an increase in the actual component degradation rate, the result of repeated and unnecessary preventive maintenance, inspection, and restoration; 2) an inefficient use of already scarce utility resources; and 3) a contribution to an increase in actual core damage risk because the correct degradation stressor was not measured.

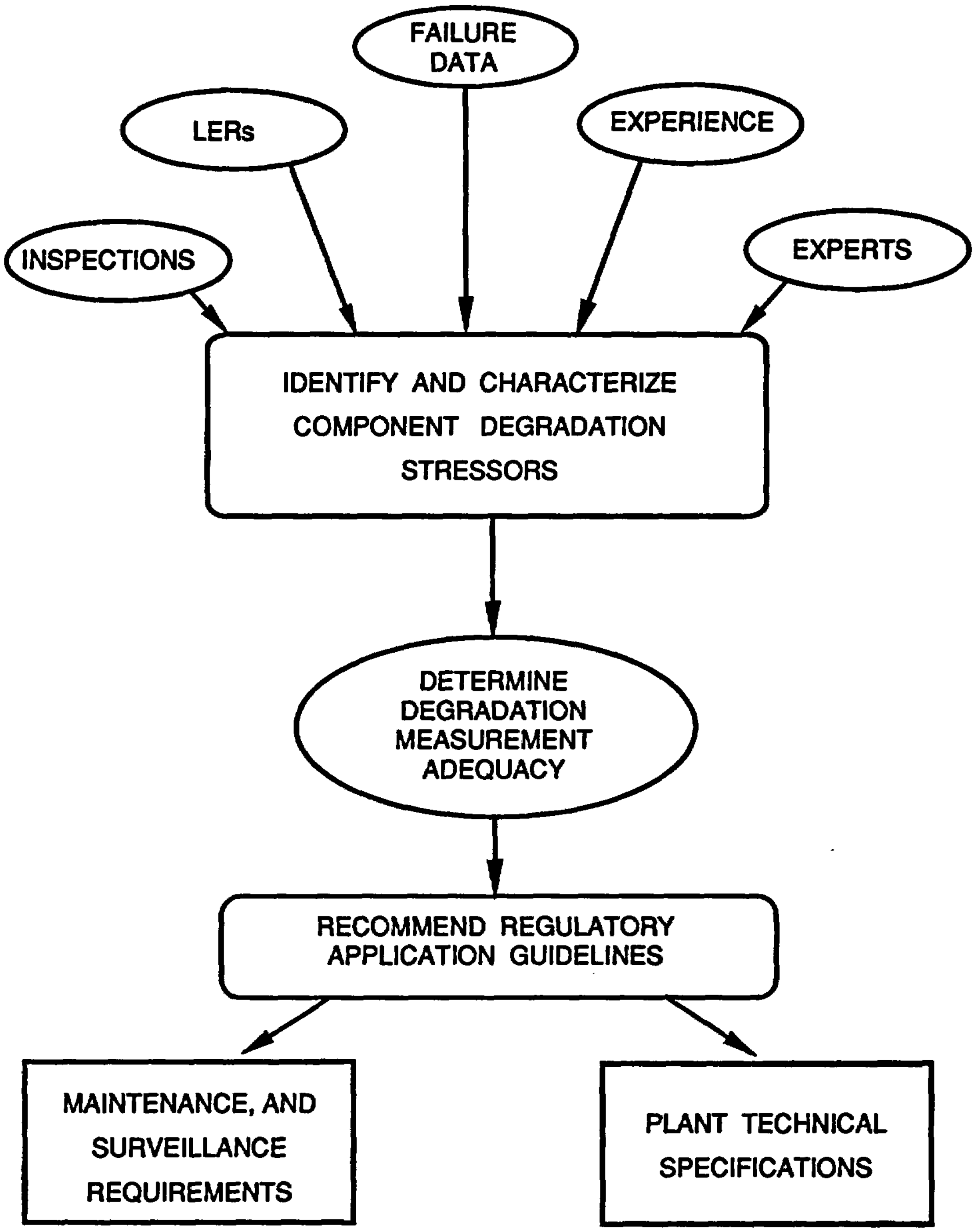

Figure 4.2 Abbreviated NPAR Program Overview

As a direct result of discovering misleading database information, the Nuclear Plant Service Water System Aging Degradation Assessment in the NPAR program (Jarrell et al. 1989) has undertaken a task to establish a methodology for conducting and documenting an RCA investigation designed to allow a more comprehensive and accurate diagnosis and documentation of the component degradation mechanisms. From a regulatory perspective, this would have the effect of:

- standardizing the inputs, methods, and deliverables of an acceptable RCA
- allowing resident inspectors to determine the adequacy of RCA methods used at their plant.

The usefulness of a formalized component failure root-cause methodology to the utility industry will vary widely because of the large differences in individual plant analysis methods, level of diagnostic expertise available, completeness of the knowledge of degradation mechanisms, and adequacy of current documentation.

4.2.1 Root-Cause Process Approach

The first step (see Figure 4.3) in the root-cause process approach was to choose and constrain a functionally significant service water component. In this case a component cooling water to service water heat exchanger was chosen because it is vital to the system function and has a demonstrated history of failures. Some 37 instances of heat exchanger failures that resulted in declaring at least portions of the SWS inoperable are documented in the approximate seven-year span addressed in the reference report by Lam and Leeds (1988).

To ensure that the focus is on the actual degradation mechanisms, the AEOD database was supplemented by visiting and examining the different SWS types at three plants, as described in Section 3.2.2, Site Information Sources. A total heat exchanger failure population of 82 events was then organized by the physical location and mechanism of the failure. From this, a set of plausible scenarios encompassing all identifiable modes of heat exchanger failure was developed.

At this point, a root-cause failure analysis was performed by a systems engineer on this specifically selected group of actual in-plant heat exchanger failures. The analytical process used by the engineer to arrive at a conclusion was explored and documented using methods from knowledge acquisition techniques found in model-based artificial intelligence technology. This solution process was then abstracted and the logic used to create the root cause of failure model shown in Figure 4.4.

A general methodology for solving the root cause of component failure was demonstrated for this general heat exchanger example in the Phase I report. It is believed this method can be expanded, at least partially automated, and applied to other components, systems, and system interactive failure investigations. Application of such a systematic approach to failure resolution benefits the users by providing:

- a logical framework for assembling the elements needed to produce a viable solution
- a method for proceeding with the analysis process
- a means of checking the reasoning invoked by the process
- a potential for automatic documentation of the investigation effort and its results.

4.2.2 Root-Cause Analysis Methodology

The proposed model of the root-cause analysis process consists of these reasoning activities: fault recognition, fault localization, fault specification, and root-cause evaluation. As shown in Figure 4.4, these activities are usually thought of as a two-step process, i.e., fault determination (what happened) and the root-cause evaluation (why it happened). By breaking these activities down into more discrete steps, it is noted that the processes are familiar to the utility system engineer. They are so familiar, in fact, that most of the processes go on almost automatically as soon as the problem is identified. This step-by-step processing does, however, have the distinct advantage of systematically identifying and documenting the areas where information is incomplete. The following items are shown on Figure 4.4:

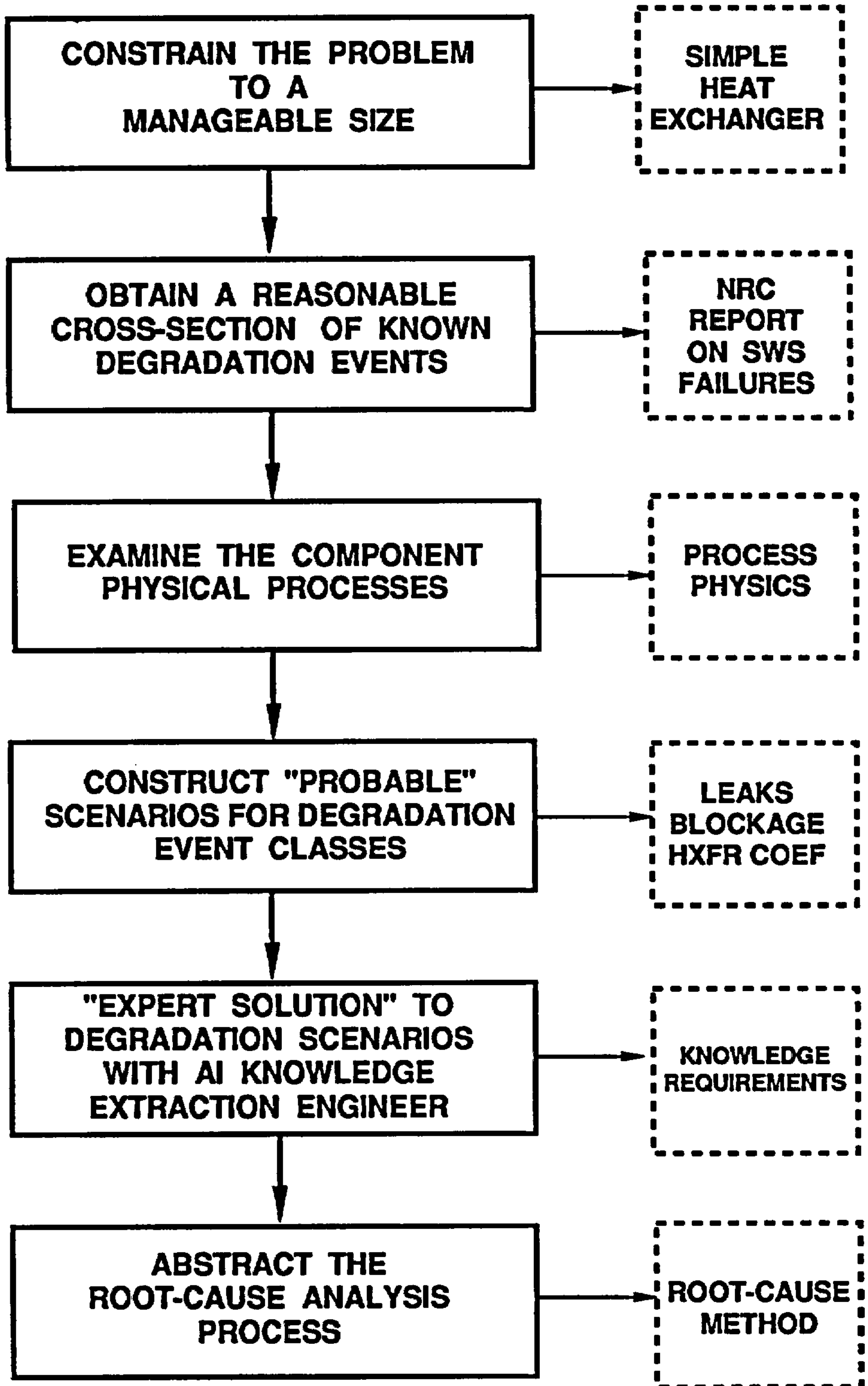

Figure 4.3 Analytical Approach

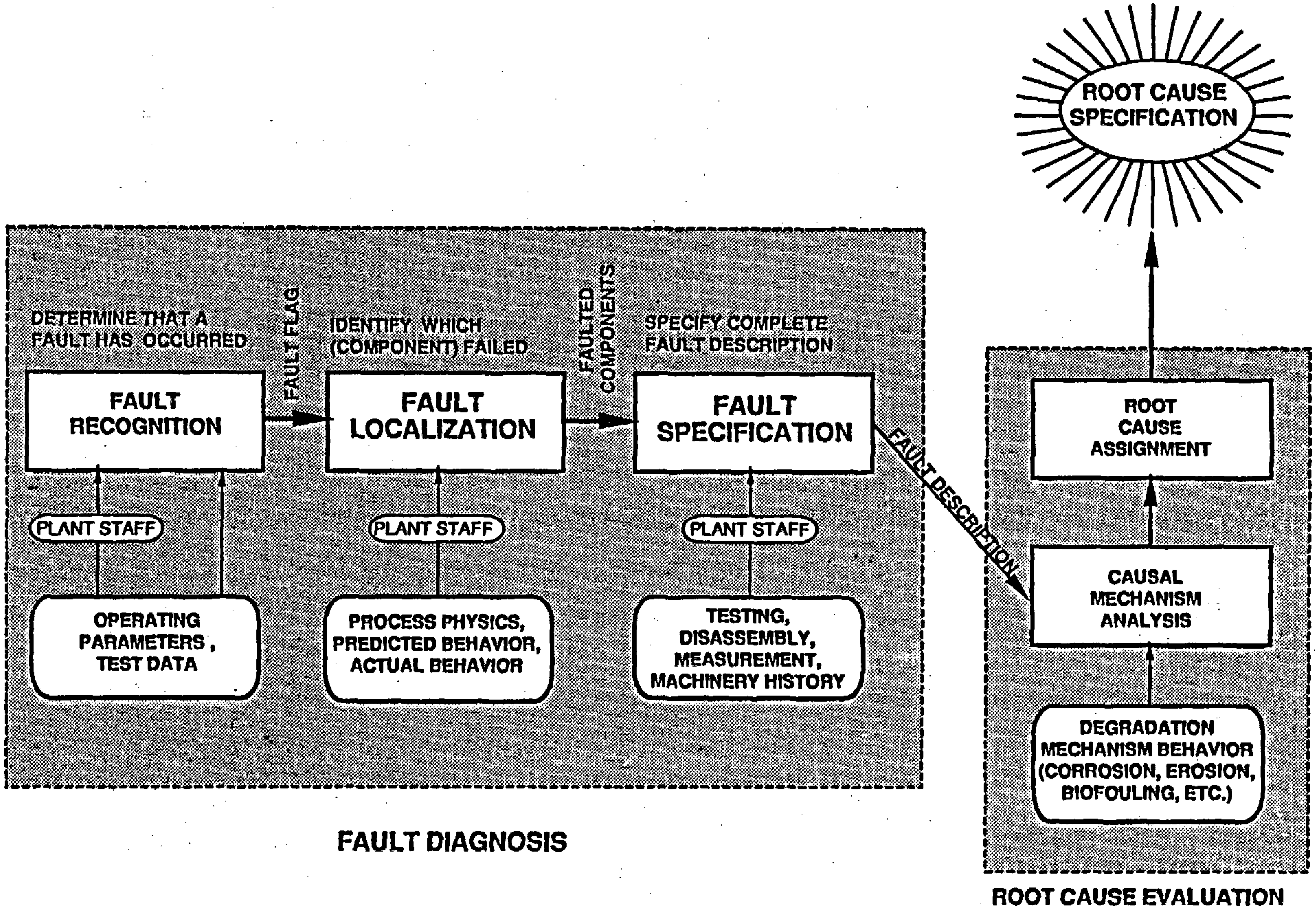

Figure 4.4. Generic Root-Cause Analysis Methodology

- *Fault Recognition* - involves reading (instrument data and related history), calculating, and comparing sensory and process information to determine if a fault condition is going to occur or presently exists. The result of this activity is the development of component and fault knowledge, notification that a fault condition exists, and an activation of further evaluation. For vital plant components, this function is implemented by alarm annunciators in the control room, while local alarms are used to indicate fault conditions of a less immediate nature.

- *Fault Localization* - processes a wider range of information than fault recognition. Its purpose is to isolate the fault to a specific component and possibly to a specific part of the component. This activity may also suggest tasks to be performed for the purpose of acquiring missing information. Plant personnel are guided in this preliminary determination by prescribed diagnostics such as annunciator response cards for most plant transient events.

- *Fault Specification* - integrates information and conclusions developed in the fault recognition and localization activities to provide a complete description to the fault. It also incorporates any additional troubleshooting or research information gathered by the plant staff as they gather pertinent data through testing, inspection, measurement, or history files. A complete record of what occurred and how it occurred is contained in an adequate fault description.

4.2.3 Basic Failure Categorization

The final step in the search for solutions to the cause of a failure must be to examine the sources of potential errors during the life cycle of the component. Any attempt to set up a scheme for categorizing events must have its foundation in the specification of the design bases of the affected system. For the SWS, this is 10 CFR 50 Appendix A, General Design Criteria 44, 45 and 46. As was explained for Figure 1.2, Chapter 1, the specific design criteria for the plant are to be found in the safety analysis report and technical specifications at the plant.

Following a component failure (see Figure 4.5), at least some degree of failure specification is performed by all plants. The relationship of the failed component's environmental stressors to the envelope of design parameters specified by the design engineer must first be examined. In proceeding down the right-hand path in Figure 4.5, if the design specifications were known to be exceeded (for example, a system pressure was taken above its design limit), then several additional questions remain. Does the operating or maintenance procedure under which the component was being operated accurately reflect the design specification? If not, then clearly an error was made when the procedure was prepared. If all the operational parameters are correctly specified, which could be expected by properly executing the procedure (including any transients that can be shown to be within the design envelope), then one must conclude that the execution of the procedure was in error. This is because it is a given that the design specification was exceeded in this investigative branch.

If the design specification was not exceeded, then the application of the component for this use becomes suspect. This is the case when a component variable is not specified in the design, but unanticipated stressors result in placing the component in an environment beyond its capability. An example might be the service water pump impeller that must be replaced after 18 months of operation because of severe erosion. The tacit assumption made by the design engineer, that pure liquid would be the working fluid, did not match the application where the river water contained a high concentration of suspended impurities. This is a prime example of what has been termed "component aging." Had the stressor responsible for failure been anticipated (included in the design specification), then two options still remain. If a manufacturing flaw can be verified, such as improper heat treatment or improper materials, then 1) the manufacturer is culpable, or, failing all of the above, 2) the design considerations and calculations must be reviewed for invalid practices.

These five categories of failure--aging, design, production, execution, and procedure--then form the basis of assigning an improvement area for preventing future failures.

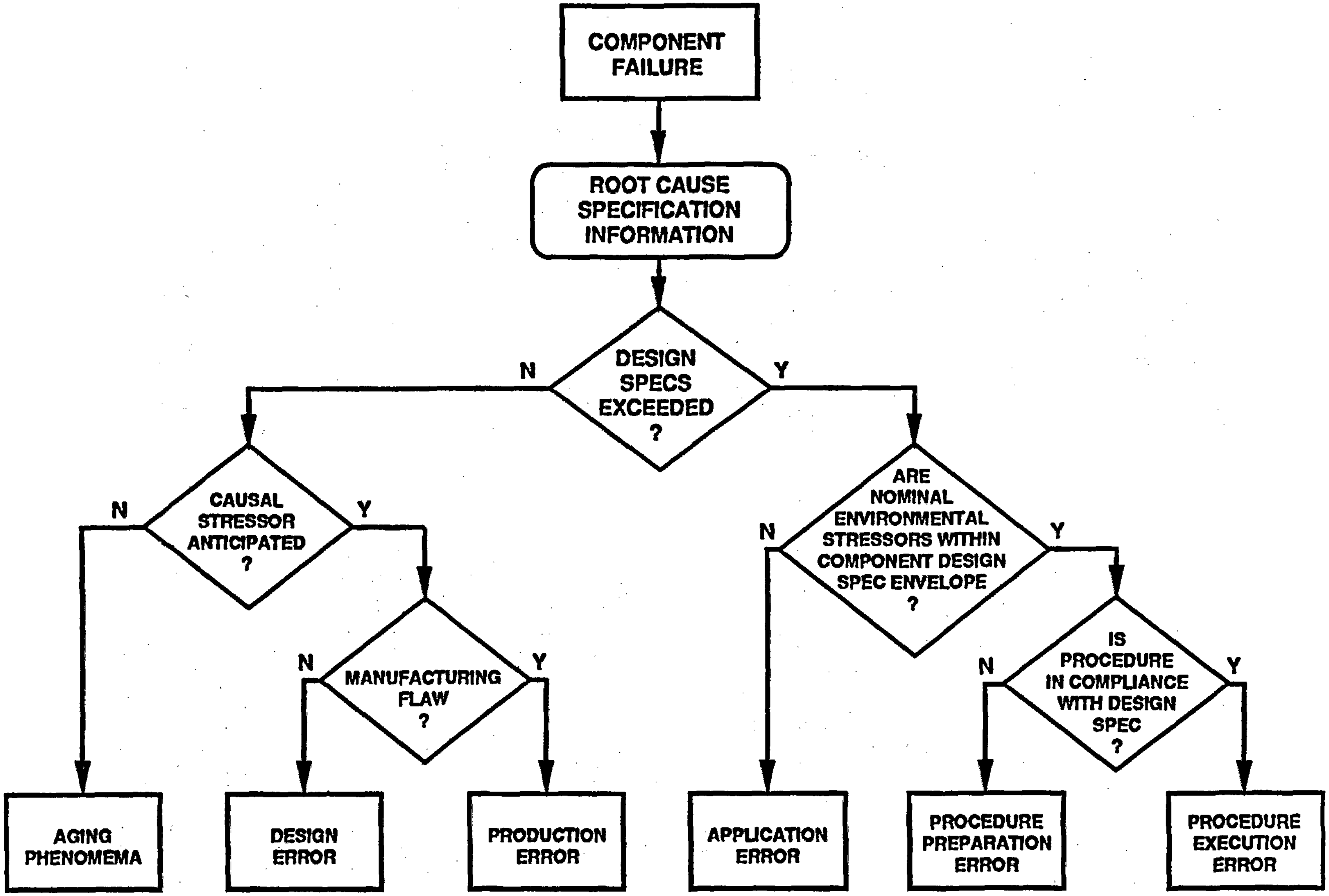

Figure 4.5 Basic Failure Categorization Scheme

4.3 Service Water System Diagnostics and Root-Cause Analysis Case Study

This section presents a case study on the evaluation of room cooler thermal hydraulic performance and EDG heat exchanger failure analysis, which was conducted in support of a SWS event at a commercial nuclear power plant. This study illustrates using the diagnostic and root-cause methodologies developed by the NPAR SWS task. PNL was requested to evaluate the measurement techniques and calculation methods used in determining the thermal-hydraulic capabilities of the safety-related portion of the SWS, hereafter called essential service water (ESW), at a large central station generating facility. The need for such an evaluation developed from 1) failure events in the EDG to ESW heat exchangers, and 2) utility efforts to initiate compliance with Generic Letter 89-13, which requires thermal performance verification of safety-related SWS components.

4.3.1 Situation Scenario

After approximately 2-1/2 years of commercial plant operation, an emergency diesel expansion tank high-level alarm was energized during routine EDG testing. This condition indicated apparent in-leakage from the ESW system (having a line pressure of approximately 100 psig) to the EDG engine closed cooling water system (slightly above atmospheric pressure). The Division 1 EDG was declared inoperative and, during the subsequent LCO, the utility examined the suspect heat exchangers and found through-wall pitting in one tube of the dual heat exchanger units. The defective tube was replaced with a new tube of like material (90/10 Cu/Ni), the heat exchanger was tested, and the EDG was returned to service.

Approximately one week later, a similar EDG tube failure occurred. The heat exchanger was repaired, eddy current tested, and approximately half the tubes were cleaned. A third failure took place two weeks later, at which time the utility completely retubed and hydrostatically tested both the Division 1 EDG heat exchanger units. A consultant was called in to examine the degraded heat exchangers and identified the degradation mechanism as local microbiologically influenced corrosion (MIC) accelerated by the elevated thermal conditions following EDG testing.

Suspecting that a global MIC condition could exist in the ESW system, the utility started an inspection of all Division 1 heat exchangers (see Figure 4.6) to determine their physical condition. This examination revealed that the remaining Division 1 heat exchangers did not exhibit extensive pitting corrosion. This was shown by a maximum tube wall pitting depth of less than 20% along with moderate general corrosion and light siltation. The lack of evidence of a widespread MIC attack was attributed to the elevated local temperature during diesel testing, combined with the stagnation conditions found during standby service in the EDG heat exchangers. These conditions were thought to promote micro-organism growth and the resulting degradation of the copper-nickel tube wall.

Prior to the system-wide heat exchanger inspection, the individual heat exchanger flow rates were measured. This pre-inspection flow measurement was performed using nonintrusive ultrasonic flow measurement devices to determine fluid velocity; the resulting component mass flow was then analytically determined. In the ESW-served heat exchangers, as-found flows varied from as low as approximately 20% to as high as 300% of component design flow. Following the inspection, piping modifications were made to provide sufficient driving head to achieve design cooling water throughput in some low-flow components which had fully opened throttling valves. With the modifications complete, the system flow balance was estimated, using a computer simulation, then physically balanced to design flow requirements using the ultrasonic flow detectors.

4.3.2 Root-Cause Issue Resolution

The above chronological summary provides a context for resolving two basic root-cause issues relative to this ESW system event:

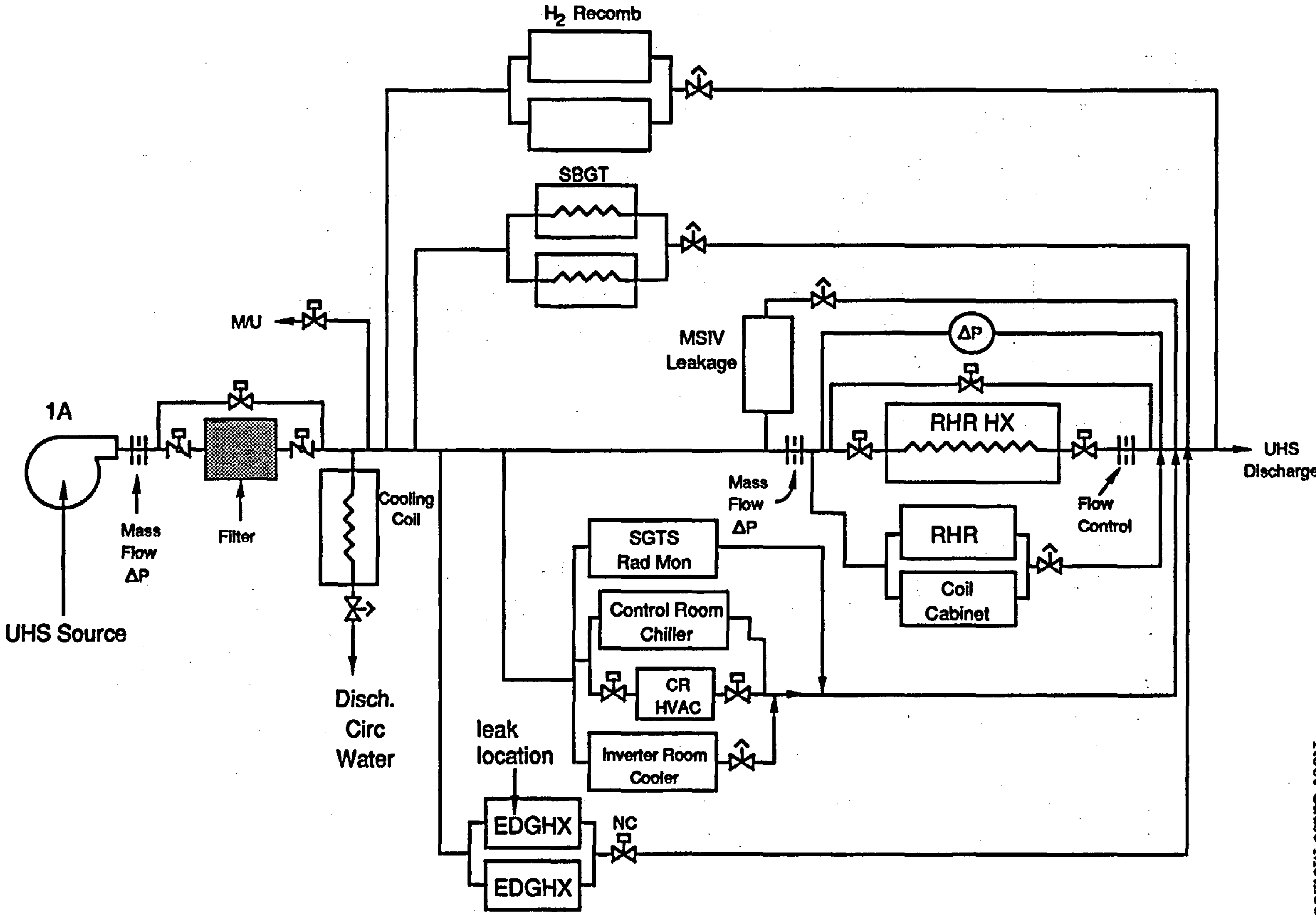

Figure 4.6 Simplified Service Water Flow Schematic

1. What was the degradation mechanism responsible for the EDG heat exchanger tube wall failure?

2. Why were the as-found component flows reduced to as low as 20% of the component heat exchanger design flow rates?

The following section endeavors to answer the first of these questions using the methods developed by this NPAR task.

4.3.2.1 Emergency Diesel Generator Tube Wall Failure Analysis

Following the methodology outline presented in Section 4.2.2 and Figure 4.4, the main steps taken to determine the root cause of the failure were the following:

- *Fault Recognition* - A fault condition was indicated by an annunciator alarm which identified a high liquid level that existed in the EDG closed cooling system head tank during EDG testing. The EDG test was discontinued, and the alarm indication was verified by a physical check of the sight glass on the side of the head tank, which confirmed the above-normal liquid level. At this point, the existence of a pressure boundary fault was further validated by checking other possible sources of pressurized liquid in-leakage at interfaces to system boundaries (make-up sources) for proper valve alignment. It should be noted that the ESW and EDG systems are de-energized (not pressurized and not running) under normal plant operating conditions.

- *Fault Localization* - When the boundary valve alignment checks indicated that all potential fill valves were in their shut position, and with no previous history of EDG-closed cooling water fill-valve failure, the most probable source of higher pressure liquid at the EDG closed cooling boundary was the ESW system. The decision was made to physically open and examine the service water pressure boundary internal to the EDG heat exchangers. This examination subsequently revealed a through-wall localized perforation near the tube sheet in the copper-nickel alloy tube wall. This provided a positive location of the fault at the EDG heat exchanger.

- *Fault Specification* - The complete specification of a component failure, or fault, provides all the relevant facts that contribute to any degradation stressors which *could* have an effect on the failed component. From first principles modeling, the possible degradation mechanisms are shown in Table 4.1.

Due possibly to the prevalence of MIC in SWS failures, a predilection tends to appear once the existence of MIC-type organisms is positively identified. In this case, a MIC expert was consulted, a positive identification of MIC organisms was produced from an examination of the samples, and MIC was declared the single operative EDG mechanism by the utility root-cause investigation team.

Figure 4.6 is a simplified drawing of one division of the ESW and shows component flow paths and the percentage of design flow through each component prior to the EDG heat exchanger failure event. When the ESW system components are operating (normally during EDG testing and plant cooldown), most heat exchangers display a flow rate considerably below the design values. The exception appears to be the failed EDG heat exchangers which have 250% to 300% of design flow. During normal plant operation, the ESW system is in a standby mode, i.e., the ESW pumps are deenergized and the ESW side of the components are in a stagnated, water-filled condition. Thus, the system had been experiencing a high flow/stagnation cyclic operation for approximately four to five years (this system was in operation prior to commercial operation of the plant). Table 4.1 shows an abbreviated listing of the modeled degradation modes, mechanisms, and stressors that were searched for possible negative influences on heat exchanger performance. As shown by the table, of the four possible mechanisms which could be responsible for a breach in the heat exchanger pressure boundary, two were shown to exist in the failed component: 1) MIC (active during stagnant standby service), and 2) high-flow-rate erosion during emergency diesel operation.

Table 4.1 Heat Exchanger Degradation Hierarchy

Modes (that are recognized)	Mechanisms (process to degrade the component)	Sample Stressors (a possible cause for the observed degradation)
Flow Blockage	Biofouling	clams, mussels, kelp
	Accumulation	high solids content, mobile corrosion products
	Foreign Debris	incomplete maintenance, detached pipe coating, detritus
Pressure Boundary Penetration	Corrosion	galvanic action, MIC, pH, biocides
	Erosion	high flow velocity, suspended solids, vapor collapse
	Structural Impact	water hammer, improper maintenance
Heat Transfer Degradation	Biofouling	slime, algae
	Accumulation	high solids, mobile corrosion products
	Foreign Debris	oil, pesticides, fertilizer

By consulting appropriate references (Dawson et al. 1990; Uhlig and Revie 1985) the copper/nickel (90/10) threshold for fluid-to-wall shear stress mass transfer effects (metal removal by erosion) was determined. With the assumption of mildly aggressive water conditions, the erosive limit was shown to be at a water velocity of approximately 7 ft/s. Calculating the tube flow area shows that from 11 to 13 ft/s existed in the heat exchanger tubes during ESW system operation. Thus, by considering all available information on fluid velocities, fluid composition, and system geometry, clear mechanisms for both erosion and MIC were shown to exist. These mechanisms are part of a more complete fault specification.

4.3.2.2 Causal Mechanism Analysis

A full description of the fault can now be passed to the next task: finding a mechanistic correspondence that exists between the root-cause hypothesis generated by the data analysis and the behavior predicted by the degradation model.

The erosion-corrosion cycle, which existed during the pre-balance condition, was composed of alternating high-fluid velocity during ESW operation (13 ft/s) and subsequent stagnation while in the standby mode. This had the effect of thinning and removing corrosion product film, as well as some of the freshly-exposed base metal by the high velocity fluid shear stress. Alternately, during the stagnation conditions encountered during standby operation, the bacteria causing the MIC phenomena were provided with the ideal environment in which to grow, i.e., stagnant water, thermal enrichment (a warm heat exchanger), and copper freshly exposed by the scouring action of the high-velocity fluid. Modeling techniques (Dawson et al. 1990) were then employed to provide a correlation between the momentum transfer effects and the stability of semi-protective films that control the rate of corrosion. This analytical model was coupled with the existing environmental conditions to validate the conclusion that through-wall perforation of the heat exchanger tubing would require both MIC and erosion-corrosion to produce through-wall penetration in the observed timeframe.

This establishes a reasonable relationship between the observed degradation and a model of the degradation process. The root cause of the failure is therefore attributable to the following two conditions:

1. excessive flow through the EDG heat exchanger tubing, which resulted in fluid velocities capable of removing corrosion product film and base metal from the tube wall

2. repeated MIC attack on the cleaned tube inner wall surface, presumably enhanced by thermal and stagnation conditions.

Actions taken by the plant were to reduce the ESW water flow through the heat exchanger to within design conditions (mitigates condition 1) and to pursue a program of biocide control of the MIC bacteria (designed to control condition 2); thus, both degradation mechanisms were addressed by the corrective actions.

4.3.3 Categorization and Conclusions

Carrying this example to its conclusion, the final question to be answered requires identifying what must be changed in the design, manufacture, implementation, operations, or maintenance cycle of the component to stop the degradation event from recurring.

4.3.3.1 Failure Categorization

The RCA of the event indicated the following: *through-wall perforation of the EDG to ESW heat exchanger was the result of MIC and erosion-corrosion in a cyclic combination.*

From this statement of cause, two specific issues must be addressed to ensure completion of the fault remediation:

1. MIC-pitting attack on the heat exchanger tube wall

2. erosion of the tube wall by high velocity fluid.

As stated in Section 4.1.1, efforts to avoid future failures start with the component design process, where a clear and obvious set of performance criteria intended for the life of the component is evident. Using the logic diagram of Figure 4.5, we will proceed to categorize the improvement areas for each of the two identified root causes.

MIC Attack

The root-cause specification gives a mechanism, MIC, which was little known at the time of the design specification phase for the EDG heat exchangers. Common design specifications list expected application environmental factors, such as maximum pressure, temperature, fluid attributes, fluid velocity, and duty cycle consistent with operation for a nominal 40-year design life. Thus, since there is no mention of MIC issues in the design specification, the answer to the design specification being exceeded for the MIC mechanism is "No" (follow the left-hand branch on Figure 4.5). Similarly, for the next decision block, this causal stressor was not anticipated, leaving the condition designated as "aging." This agrees with the definition given in the introductory section of this report.

While this condition is recognized as being outside the original design envelope, future heat exchanger design efforts would be expected to profit from such experiences and "design out," or at least be based on aging mitigation using biocidal controls for bacteria shown to be indigenous.

Erosion (High-Flow Velocity)

The design flow velocities for the EDG's heat exchangers were 4 and 5 ft/s. Measured velocities in the 11 to 13 ft/s range place the operation outside the design specifications (follow the right-hand branch on Figure 4.5). The high ESW flow through the EDG cooler was not in compliance with the flow balance procedure. An application/execution error appears to have occurred.

The plant's root solution to these two degradation mechanisms was to establish a biocide program to control the MIC organisms (mitigate future MIC attack), and to provide new system balancing methods to accurately set component flow rates to near-design values (eliminate the erosion mechanism).

4.3.3.2 Summary of Conclusions

The ESW problems, their identified root causes, and the root cause initiators are summarized by Table 4.2.

Table 4.2 Root Solution Summary

Problem	Root Cause	Solution	Category
EDG heat exchanger tube pressure boundary penetration	1. Microbiologically influenced corrosion	1. Biocide injection to control aging	Aging Phenomena
	2. High velocity water erosion	2. Design basis component flow rate	Application/Execution Error that Promoted Aging

This example case study has demonstrated links between the design specifications of safety-significant system components, operation of the components within the total specification set, surveillance testing that measures the effects of all the active degradation stressors, and the necessity of a systematic evaluation of the root cause of component failure. This completes the loop discussed in Section 1.3 concerning the feedback mechanism required to effectively implement design-basis regulated operation.

The importance of understanding the complete set of operative degradation mechanisms is that it ensures, as far as is possible, that the previously failed component is now being operated within the specified design conditions envelope. The advantages of operations within the envelope are that 1) historically, other similar components have survived for the duration of the design life under the specified conditions while providing satisfactory performance, i.e, it is a proven, cost-effective manner of operation, and 2) the regulatory requirements of the operating license are being met.

5 Degradation Analysis

The Phase I degradation analysis applied a root-cause methodology to plant-specific failure data to determine failure mechanisms responsible for aging. Undocumented information obtained from plant personnel by means of the questionnaire protocol was also utilized during root-cause analysis. Analysis results indicate that these techniques lead to successful identification of the primary age-related failure mechanisms that affect SWS operability.

In Phase I, effort was concentrated on data from an open SWS. In Phase II this same methodology was applied to two additional types of SWSs in an effort to ensure a complete study of all potential degradation mechanisms that could affect SWSs.

5.1 Source of Data

Data obtained in Phase I of this study were derived from the records of a single commercial reactor with a fresh water, open-cycle SWS and consisted of 324 plant maintenance records. These records document all SWS failures, surveillance tests, and inspections occurring over a 21-month period. The successful data-gathering techniques of Phase I (system analysis followed by formal protocol plant visits) were repeated in Phase II to acquire data from a closed-cycle-type SWS using saltwater, and a fresh water recirculating SWS. The result is a database that allows cross-plant data to be compared for common mechanism identification and failure modes, thus leading to a more confident assessment of all potential aging mechanisms.

During plant visits for Phase II, 554 maintenance records, covering 9 years and 10 months of operation, were obtained from a plant using a fresh water, recirculating SWS. A second plant with a salt water, closed-cycle SWS provided 968 maintenance records spanning 20 years of operation, although only records dating back through 1980 were used. These records document all SWS failures, surveillance tests, and inspections that took place at each plant during the time span indicated above.

5.2 Component History Database

Relevant information obtained from the Phase II site visits to a recirculating and closed SWS was entered into the database developed in Phase I (refer to Table 5.1). The type of information and level of detail on plant records from the two additional plants parallel those of the records previously studied in Phase I. The conclusion reached in Phase I regarding plant record keeping is also applicable to the additional data acquired during this phase: close work with plant data has shown several areas in which plant maintenance record keeping needs to be improved. Specific improvements are presented in Section 5.1 of the Phase I report (Jarrell et al. 1989).

5.3 Analysis of Plant Data

This section examines analysis of the content of the component history database; an outline of the approach taken to analyze and interpret the data; comparison of cross-plant data for common tendencies and relations; and an aging assessment of SWSs based on this limited data, with a summary of conclusions based on that assessment. Test results from the Phase I report are used to compare and substantiate the outcome of the additional SWS assessments.

5.3.1 Failure Classification

Each entry in the database was categorized based on the type of event documented; these categories are summarized in Table 5.2. A Category 1 designates records that indicate that a failure occurred. Records indicating that a problem was discovered during a surveillance test or an inspection are placed in Category 2. Some of these "problems" are classed as a failure, while others simply indicate a degraded state of the component. The third category consists of records of surveillance tests for which no problems were found. Category 4 records work which was done in support of events covered by Categories 1 through 3. Design changes and resulting diagram changes fall under Categories 5 and 6, respectively. Records indicating that work was performed,

Table 5.1 Component History Database Contents

Database Field	Information Contained
Plant	Name of plant document came from
Document Type	Type of document data came from (machinery history, LER, inspections report, etc.)
Document Number	Individual document identification
Component Code	Individual component identification
Component Type	Type of component
Subcomponent	Specific part within component boundary
Start Date	Date problem reported, or date document initiated
Stop Date	Date installation and testing completed
Category	Assigned based on reason document initiated and event outcome
Comments	Description of maintenance work and testing
Failure	Specific failure; if it can be determined that a failure occurred
Root Cause	Root cause of failure, problems, symptoms or other observations
Failure Mechanism	Failure mechanism responsible for failure

but do not indicate why the work was necessary, are classed as Category 7. Records in Category 8 document maintenance work that was cancelled. Category 9 is

Table 5.2 Maintenance Document Categories

Category	Description
1	Failure[a]
2	Surveillance and/or inspection; problem found[a]
3	Surveillance and/or inspection; no problem found
4	Support work
5	Design change
6	Drawing change only
7	Work done; reason not stated[a,b]
8	Cancelled
9	Unclear[b]

(a) Root-cause analysis carried out on these categories.
(b) Not enough information to place in any of the previous categories.

reserved for records that are incomplete and cannot be placed in Categories 1 through 8. Entries in Categories 1, 2, 7 and 9 were further analyzed using a formalized root-cause logic.

5.3.2 Root-Cause Analysis of Plant Data

It is essential, in the process of evaluating the effect of aging on the plant, to be able to identify, with a high degree of certainty, the failure mode and mechanism responsible for each failure or problem, and whether this mechanism is age-related or not. Once this has been accomplished, the effects of aging on risk may be evaluated. The root-cause analysis methodology discussed in Section 4.2 was utilized in determining the failure mechanisms responsible for the events documented in the component history database.

Results of the Phase I study indicated that the information currently available on plant maintenance records is not sufficient for assessing aging. The undocumented detailed information used by plant personnel in reaching their conclusions about each case is needed to obtain useful results from a root-cause analysis. To recover some of this information, PNL personnel involved in the process interviewed plant maintenance personnel before performing the actual analysis; this additional data enabled PNL personnel to assign root causes to events that did not otherwise have sufficient documentation to determine the root cause of the event. While these interviews improved the accuracy of the root-cause analysis results, this type of analysis would be most effective if it were carried out at the plant by those who actually participated in the maintenance work.

The first step in applying root-cause analysis to the plant data was to assign an initial root cause to each event classified as belonging to Category 1 or 2. The root cause was then analyzed by PNL personnel who had communicated with the maintenance personnel at the site and assigned a certainty level, based on expert opinion. PNL personnel who had visited the sites, confirmed or refuted the results of the root-cause analysis from the documentation of the plant maintenance records. A certainty level, based on expert opinion, was then assigned to each root-cause analysis, as well as to the failure mechanism. Table 5.3 describes how each level of certainty was assigned.

Table 5.3 Root-Cause Analysis Certainty Assignments

Classification	Certainty
1	<95%
2	<75%
3	<50%
4	<25%

5.3.3 Aging Analysis

The analysis process began by extracting components from the database containing only failures associated with aging or unknown failures and certainty values of 95% or greater (Table 5.3, category 1). Table 5.4 explains the types of failure mechanisms used to analyze the SWS systems.

Figures 5.1 through 5.6 illustrate failures per year per type of failure mechanism. The failure mechanisms per year were determined by the year in which the work was performed. The year chosen was based on the component's work start date; if that date was not provided, the ending year date was used. If neither the work beginning or completion date was recorded in the records, the failure was not entered into the graph. This loss of information accounted for only a small percentage of records not being displayed: open system, less than 10%; closed system, less than 1%; open recirculating system, approximately 0%. It can be stated that the results will show an accurate representation of the failure relations, given the data provided.

5.3.3.1 Open SWS Analysis

Figures 5.1 shows the number and relationship of failure rates per year and the type of failure mechanism associated with each failure for an open SWS. Figure 5.2 shows the distribution of components to types of failures. As determined in the Phase I report, corrosion was the largest contributor to failure, compared to other aging mechanisms. However, it may appear from Figure 5.1 that there was a problem in 1986. It should be noted that the data collection was taken from 1985 through 1986 and partially through 1987, thereby skewing the data to 1986, since there is a complete record of 1986 but not of 1985 or 1987. A majority of the degradation and failure events of less than 95%- cause certainty were categorized as corrosion-related, but because of the lower certainty value, were not included in Figure 5.1. Figure 5.2 exhibits, as mentioned, corrosion as the major contributor to failure, with biofouling and wear being the next largest contributors to failures. Figure 5.2 also confirms that valves are more likely to corrode rather than to wear or be subjected to biofouling. This figure indicates that pumps and motors

Table 5.4 Service Water System Principal Aging Degradation Mechanisms

Degradation Mechanism	Abbreviation	Definition	Sample SWS Manifestations
Corrosion	COR (A1)	Destructive attack of a metal by reaction with its environment.	rust, valve contact surface adhesion, formation and transport of metallic oxides, leaching (selective alloy metal removal), through-wall-perforation
Erosion	ERO (A2)	Metal removal by high-speed liquid or suspended solid impingement.	pump impeller metal removal, elbow wall metal removal, heat exchanger tube wall metal removal
Cavitation	CAV (A3)	Local fluid pressure drops below saturation producing voids that subsequently collapse, resulting in metal removal.	pump impeller erosion, pump casing erosion, valve control surface loss
Vibration	VIB (A4)	Low displacement, high-frequency cyclic stress.	fretting (e.g., baffle-tube wall thinning), or corrosion fatigue
Fatigue	FTG (A5)	High frequency stress below the yield strength of a material resulting in failure by cracking.	progressive fracture (fatigue or corrosion fatigue) from points of high stress concentration
Thermal Cycling	THC (A6)	Stress changes in a material caused by variations in temperature.	pipe coating fracture, tube-to-tube sheet break
Wear	WER (A7)	Removal of surface material by sliding contact between it and another surface.	bearing surface removal
Biofouling	BIO (A8)	Buildup of a biological species.	heat exchanger tube plugging, pipe flow area reduction, slime mold preventing heat transfer
Accumulation	ACU (A9)	Chronic inorganic solids precipitation or radial deposition on fluid or heat transport surfaces.	bottom buildup in pipe, flow area reduction
Structural Impact	STI (A10)	A small number of cyclic loads applied to a material at a high rate and short duration.	denting or physical deformation of otherwise undamaged material (water hammer, exterior impact)
Foreign Debris	DBR (A11)	Rapid accumulation of material of significant size from a source outside the component.	pipe coating material, gasket material, wrenches, cloth maintenance items, detritus
Unknown	U	Less than 95% confidence in the accuracy of the fundamental degradation stressor.	symptoms of degradation given as cause, incomplete information, contradictory information

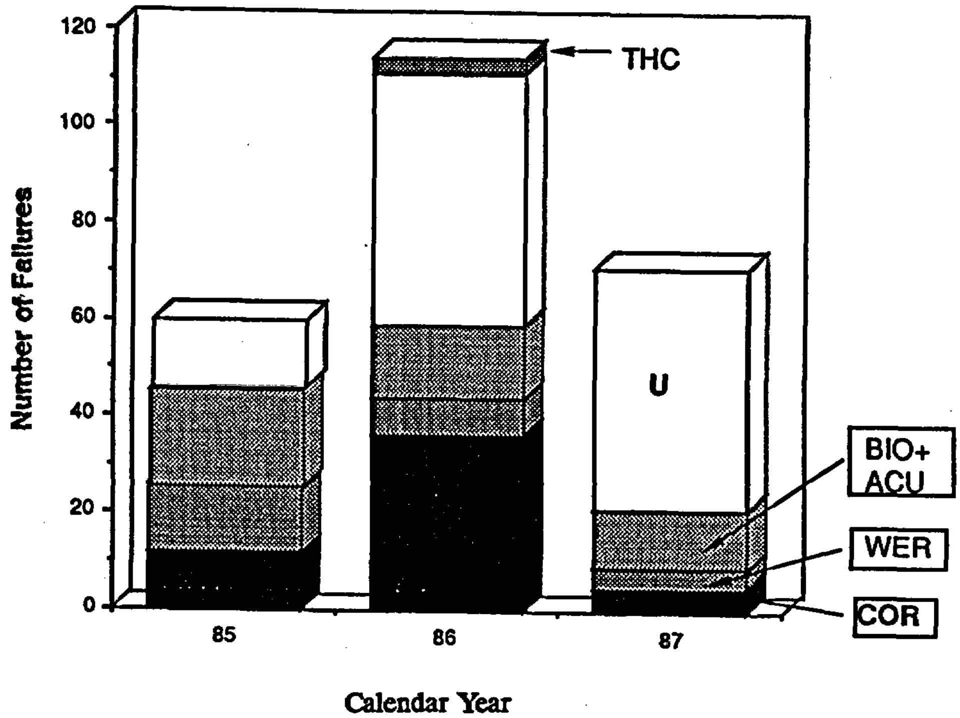

Figure 5.1 Open Service Water System Failure Rate Distribution

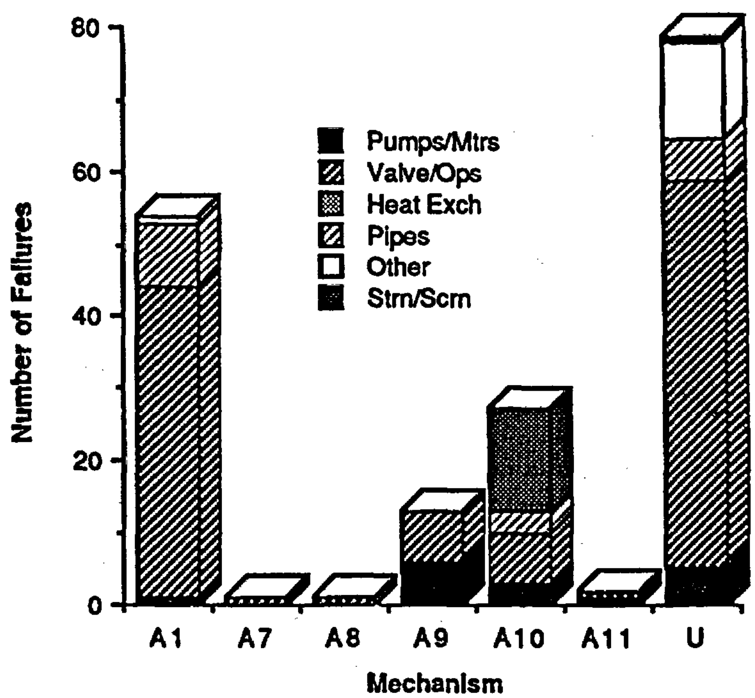

Figure 5.2 Open Service Water System Component Failure Distribution

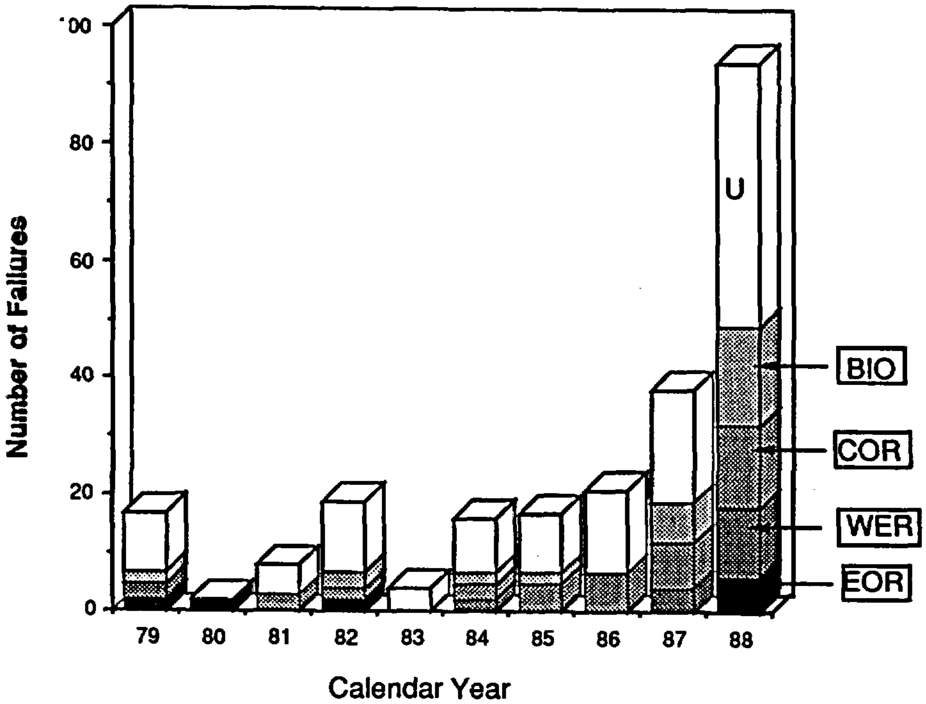

Figure 5.3 Closed Service Water System Failure Rate Distribution

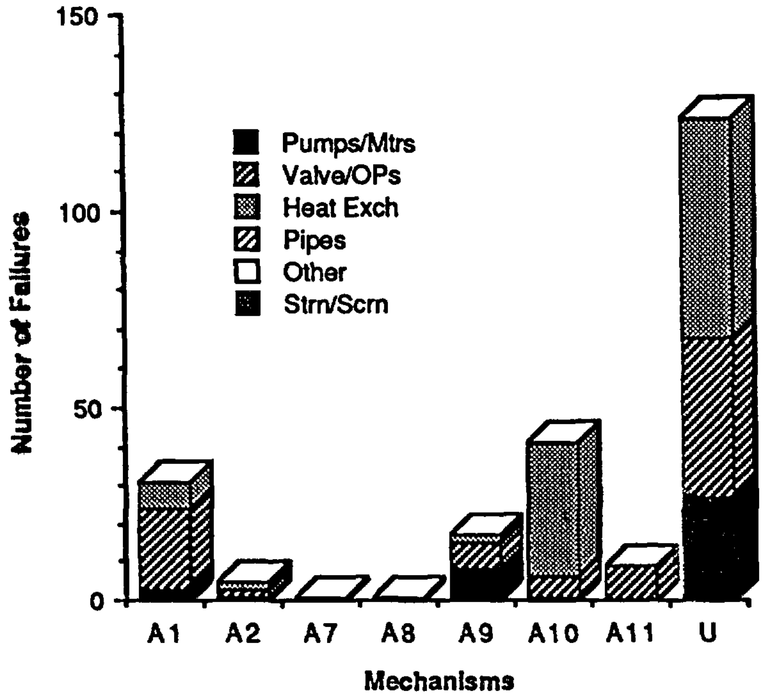

Figure 5.4 Closed Service Water System Component Failure Distribution

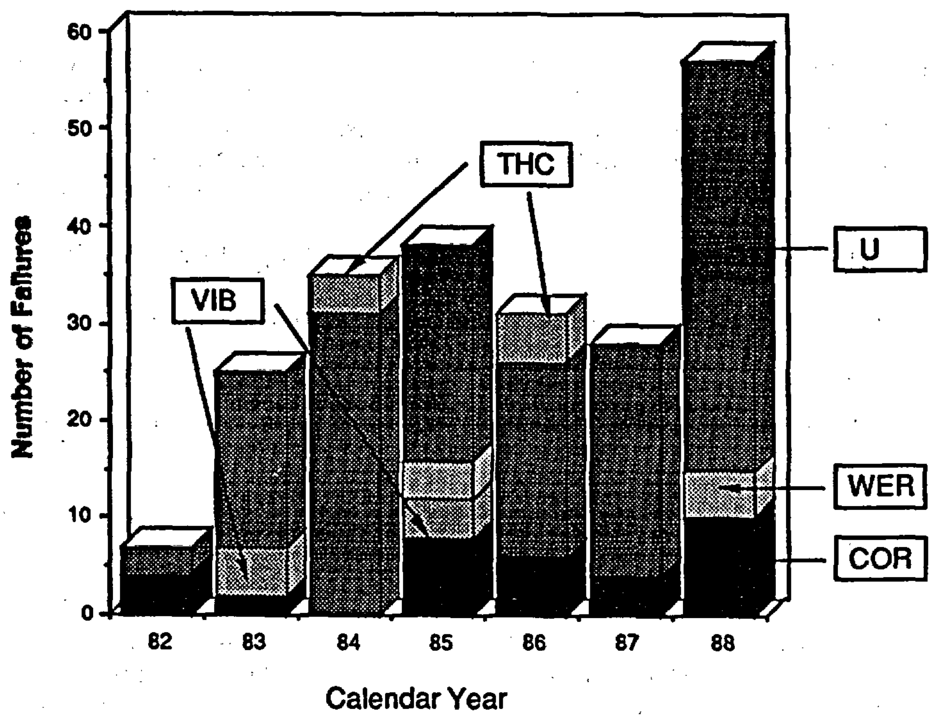

Figure 5.5 Recirculating Service Water System Failure Rate Distribution

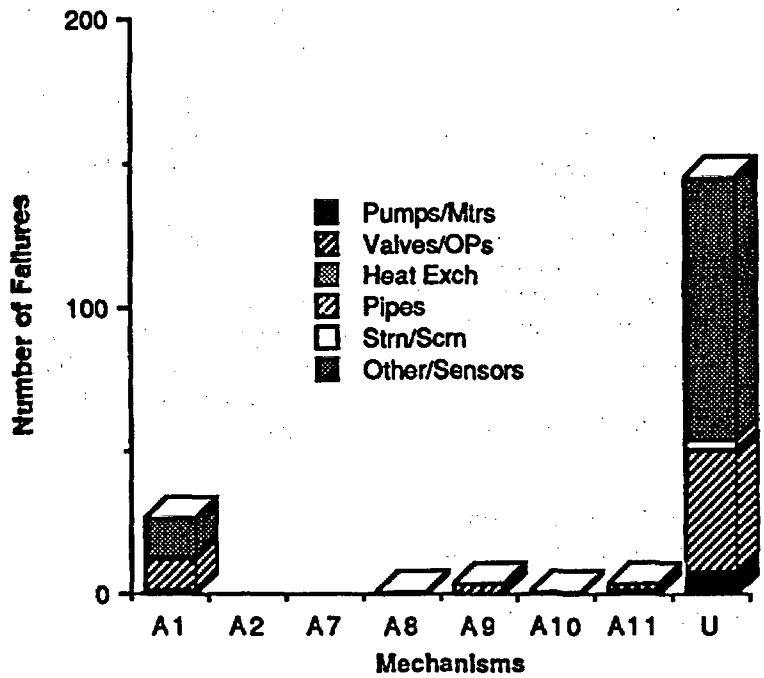

Figure 5.6 Recirculating Service Water System Component Failure Distribution

are more likely to fail due to wear, and that screens and strainers are more likely to fail because of biofouling. The largest number of failures were of the unknown category. Of that category the valves class contributed approximately 70% to the number of failures.

5.3.3.2 Closed SWS Analysis

Figure 5.3 and Figure 5.4 display the failures for a closed SWS. Figure 5.3 specifically shows failure rates per year from pre-1980 through 1988. From 1980 to 1988 the number of reported failures increased, with a dramatic increase in 1988. This apparent trend is the result of action imposed by the NRC on all nuclear power plants to maintain a more accurate description of the maintenance work being performed.

Figure 5.4 shows that the largest failure is the result of biofouling in the heat exchangers, with corrosion of the valves the second largest contributor to failures. Wear in valves, pumps, and motors also contributes a high fraction of failures. The classes of heat exchangers, valves and pumps, and motors create the main components that would be most likely to fail.

5.3.3.3 Recirculating SWS Analysis

Figures 5.5 and 5.6 display the failures for a recirculating SWS. Figure 5.5 specifically shows number of failures per year from 1982 through 1988. From 1984 to 1986 the number of failures appeared to decrease, but showed a marked increase in 1988. As mentioned in the closed system analysis, this trend was due to the action imposed by the NRC on all nuclear power plants to maintain a more accurate description of the maintenance work being performed. The decrease from 1984 to approximately 1987 was due to the corrosion treatment program instituted by the facility.

The recirculating SWS's primary failure mechanism, as shown in Figure 5.6, was corrosion. Corrosion appeared to mainly affect sensors and valves. The unknown category in Figure 5.6 consisted of three component classes--pumps and motors, valves, and sensors. The recirculating SWS's main failure component was the sensors. The number of unknown root causes in Figure 5.5 indicated that not enough critical information was provided to ascertain the root cause of the failure.

5.3.4 Principal Degradation Mechanisms

The major findings in examining the three types of SWSs are as follows:

Corrosion, fouling (biological and inorganic), and wear were the three major failure mechanisms.

Valves and valve operators were the most likely components to fail.

Failure of components could be correlated with a certain type of failure mechanism (e.g., valves were more likely to fail because of corrosion, and pumps were more likely to fail because of wear).

Failure and routine maintenance information is not complete enough to perform an effective root-cause analysis.

The data indicated that the recirculating SWS had the least failures per year, assuming that the documentation was comparable to the other SWSs.

5.3.5 Control Procedures

A range of water chemistry control procedures for each system type is summarized in another publication (Johnson and Jarrell 1991). Other surveillance procedures used to counter SWS degradation are also summarized.

6 Conclusions

In keeping with the stated goals of this report, the Phase II SWS aging investigation produced the following conclusions:

1. The primary SWS degradation mechanism of corrosion, compounded by biological and inorganic accumulation, has been validated through the analysis of a substantially broadened and verified component failure database.

   Discussion

   Whereas, the Phase I report conclusion was reached almost entirely from information obtained from one type of SWS, Phase II includes an in-depth review of machinery histories from three different plant SWS types. Industry-wide data from the NRC, contained in the report issued by the AEOD (Lam and Leeds 1988), also support this primary conclusion.

2. A corollary to this primary conclusion is that, in general, the most effective means of mitigating SWS degradation is to pursue control methods that reduce the rate of corrosion and biofouling.

   Discussion

   This conclusion suggests that a necessary element to minimization of degradation in any type of SWS is effective control of water chemistry. A range of water chemistry treatments is available with options that apply specifically to one or all of the SWS types. Where confirmed biological agents are active, a biocontrol program (chemical, thermal, etc.) is an obvious requirement. Environmental regulations can limit the extent to which some of these solutions can be pursued, and a careful consideration of alternatives is necessary to select the most cost-effective solution. Any changes to the control program should be accompanied by monitoring for induced stressors [e.g., leaching (denickelfication)] of Cu/Ni heat exchangers by a chlorination biocide).

3. A demonstrated need for a systematic methodology for performing and documenting an effective root-cause analysis of component failure resulted in developing the mechanistic correspondence approach explained in Chapter 4.

   Discussion

   The formalized heat exchanger root-cause analysis method developed during this task has been shown to be an effective guide to performing RCA in a field environment. The methodology is generic in that, by developing degradation mechanistic models of components other than heat exchangers, the same technique can be applied. It is recommended that an RCA process having the elements described in Chapter 4 be required for all corrective maintenance work orders performed on safety-related portions of the SWS.

4. The most reliable and complete data source available for determining age-related degradation in nuclear power plant systems and components resides in the "corporate memory" of the engineers, operators and maintenance personnel at the plant.

   Discussion

   Despite vast improvements in reporting criteria and computerized media, the root-cause failure knowledge gained by operating and maintaining the plant systems is not being communicated in sufficient depth to allow confident conclusions. Implementing the RCA methods developed for this report and documenting the failure specification and degradation correspondence is designed to assist in filling this information/ documentation void.

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

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

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

NUREG/CR-5379
PNL-7916
Vol. 2

2. TITLE AND SUBTITLE

Nuclear Plant Service Water System Aging Degradation Assessment

Phase II

3. DATE REPORT PUBLISHED

MONTH	YEAR
October	1992

4. FIN OR GRANT NUMBER

B2911

5. AUTHOR(S)

D.B. Jarrell, L.L. Larson, R.C. Stratton, S.J. Bohn, M.L. Gore

6. TYPE OF REPORT

7. PERIOD COVERED (Inclusive Dates)

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

Pacific Northwest Laboratory
Richland, WA 99352

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

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

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The second phase of the aging assessment of nuclear plant service water systems (SWSs) was performed by the Pacific Northwest Laboratory (PNL) to support the U.S. Nuclear Regulatory Commission's (NRC's) Nuclear Plant Aging Research (NPAR) program. The SWS was selected for study because of its essential role in the mitigation of and recovery from accident scenarios involving the potential for core-melt, and because it is subject to a variety of aging mechanisms. The objectives of the SWS task under the NPAR program are to identify and characterize the principal age-related degradation mechanisms relevant to this system, to assess the impact of aging degradation on operational readiness, and to provide a methodology for the management of aging on the service water aspect of nuclear plant safety.

The primary degradation mechanism in the SWSs, as stated in the Phase I assessment and confirmed by the analysis in Phase II, is corrosion compounded by biologic and inorganic accumulation. It then follows that the most effective means for mitigating degradation in these systems is to pursue appropriate programs to effectively control the water chemistry properties when possible and to use biocidal agents where necessary.

A methodology for producing a complete root-cause analysis was developed as a result of needs identified in the Phase I assessment for a more formal procedure that would lend itself to a generic, standardized approach. It is recommended that this, or a similar methodology, be required as a part of the documentation for corrective maintenance performed on the safety-related portions of SWSs to provide an accurate focus for effective management of aging.

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

Nuclear Power Production
System Aging
Service Water System
Component Degradation
Root Cause Analysis
Failure Mechanisms

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)
Unlimited

(This Report)
Unlimited

15. NUMBER OF PAGES

16. PRICE

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