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Aging Assessment of Westinghouse PWR and General Electric BWR Containment Isolation Functions

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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

A study was performed to assess the effects of aging on the Containment Isolation (CI) functions of Westinghouse Pressurized Water Reactors and General Electric Boiling Water Reactors. This study is part of the Nuclear Plant Aging Research (NPAR) program, sponsored by the U.S. Nuclear Regulatory Commission. The objectives of this program are to provide an understanding of the aging process and how it affects plant safety so that it can be properly managed. This is one of a number of studies performed under the NPAR program which provide a technical basis for the identification and evaluation of degradation caused by age.

Failure data from two national databases, Nuclear Plant Reliability Data System (NPRDS) and Licensee Event Reports (LERs), as well as plant specific data were reviewed and analyzed to understand the effects of aging on the CI functions. This study provided information on the effects of aging on component failure frequency, failure modes, and failure causes. Current inspection, surveillance, and monitoring practices were also reviewed.

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EXECUTIVE SUMMARY

An aging assessment of Containment Isolation (CI) functions for Westinghouse pressurized water reactors (PWRs) and General Electric (GE) boiling water reactors (BWRs) has been performed as part of the Nuclear Plant Aging Research (NPAR) program. The NPAR program is sponsored by the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, Division of Engineering Technology. Its goal is to provide a technical basis for understanding and managing the effects of aging in nuclear plants. For both PWRs and BWRs, containment isolation is one of the important safety functions for mitigating the unlikely events of design basis accidents (DBAs).

The CI function is an engineered safety feature (ESF) that prevents and minimizes the release of radioactivity. It is not a single, well defined system in either GE BWRs or Westinghouse PWRs. For the Westinghouse PWRs, the containment isolation function consists mainly of containment isolation valves and valve operators. Automatic containment isolation is initiated by the engineered safety features actuation system (ESFAS), which is not included in this study. However, for BWRs, most of the containment isolation valves and valve operators are in the nuclear steam supply shutoff system, which also initiates the containment isolation signals. For this reason, the failure data of switches, bistables, radiation detectors and transmitters in the CI functions of the BWRs were analyzed in this study.

This study reviewed the safety analyses related to the containment isolation function and focussed on those components which were most risk-significant based upon specific accident scenarios. Previous NRC studies indicated that immediate detection of the failures is important in further reducing the safety significance of failures of the components in the containment isolation function.

The failures of the components in the CI functions reported to the Nuclear Plant Reliability Data System (NPRDS) (3,519) and Licensee Event Report (LER) (1,062) databases for the 1988 through 1993 time period were reviewed and analyzed to understand the effects of aging on failure frequency, failure modes, and failure causes. The results show that valves and valve operators were the most frequently failed components, and that most failures reported to the NPRDS were due to aging. Specifically, 76% of PWR and 80% of BWR valve failures reported to the NPRDS were aging-related, while 74% of PWR and 66% of BWR valve operators were aging-related.

A review of the failure frequency curves for individual PWR valve types indicated variations between valve types. The globe valve was most frequently failed in PWRs, as opposed to gate valves for BWRs. In PWRs, globe and gate valves showed a rise in failure frequency with age, while check and butterfly valves did not show a similar increase. BWR valves demonstrated more consistency for the different valve types, with increasing rates of failure occurring after 15 years of service.

The most commonly reported failure modes for the valves included leakage (both internal and external) and failure to close. The leakages were caused by aged seat/discs, packing, and gaskets. Corrosion of valve internals and resulting corrosion product/dirt buildup were found to be primarily responsible for valves which failed to close.

Pneumatic valve operators contributed to the majority of the valve operator failures for both PWRs and BWRs. The operating data showed a continuous increase in the failure frequency when plotted against component age at failure. For the containment isolation function, failure to close is the most significant failure mode, and the primary causes of pneumatic valve operator failures were degradation of the diaphragm, solenoid valve, limit switch, and problems associated with the air supply to the operators.

Primarily due to the redundancy provided in the containment penetrations design (e.g., series valves), these component failures did not result in any loss of containment function. However, some of the failures did affect plant operation, and resulted in power reductions, unit shutdowns, and delayed startups. A review of the methods of detection for these failures indicated that testing and surveillance programs detected the majority of them. However, some failures, especially for valve operators, were detected by operational abnormality. Thirty-one percent of the failures of pneumatic valve operators for PWR CI functions and 27% for AC electric valve operators for BWR CI functions were detected as operational abnormalities, which highlights the importance of continued attention required for these components.

A representative BWR-6 type plant was visited to obtain information on the inspection and maintenance activities related to components which provide the containment isolation function. The majority of the inspections performed are in accordance with Technical Specifications, ASME Section XI inservice testing, and Appendix J requirements. The maintenance records at this plant were reviewed, and found to be evenly divided between preventive and corrective maintenance type activities. A close correlation was established between the insights gained from the analyses of the national operating data and this plant's corrective maintenance records (e.g., the fraction of failures due to aging, failure causes, failure modes, and sub-components failed).

As a result of this Phase I study, the aging processes of the components of the containment isolation functions in both PWRs and BWRs are better understood through the aging characterization of major components. The components most frequently affected by aging degradation should have higher priorities in aging management. The information obtained from this study provides a technical basis for future work.

The following conclusions are made based on the findings from this study:

- A large portion of the component failures in the CI functions reported to NPRDS are aging-related, 76% for PWRs and 80% for BWRs. The results of this study also have shown that aging-related failures increase as the components in the CI functions age, and the failure modes and causes are affected by aging.

- Even though a large number of component failures in the containment isolation functions were reported to NPRDS and LERs, most of them were not safety-significant. However, for both PWRs and BWRs, during the period studied, there were several failures of components in the CI function that affected the plant operation, either resulting in unit off-line, or in a reduction of power operation.

- Many reported failures were caused by regular maintenance items, such as valve packings. Utilizing the insights from the operating experiences and aging studies may be beneficial in reducing these aging-related failures.

- This study characterized the aging degradation of different types of valves and valve operators separately, and the results show that different component types show different aging effects. For operating data analyses, it is recommended that the groupings be as small as possible, as long as there are enough data available for meaningful analyses.

ACRONYMS

AFW	Auxiliary Feedwater
BWR	Boiling Water Reactor
CI	Containment Isolation
CCW	Component Cooling Water
CVCS	Chemical and Volume Control System
DBA	Design Basis Accident
ECCS	Emergency Core Cooling System
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GE	General Electric
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
HPSI	High Pressure Safety Injection
LER	Licensee Event Report
LLRT	Local Leak Rate Test
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPSI	Low Pressure Safety Injection
MFW	Main Feedwater
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
NPAR	Nuclear Plant Aging Research
NPRDS	Nuclear Plant Reliability Data System
NRC	Nuclear Regulatory Commission
NSSSS	Nuclear Steam Supply Shutoff System
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RTS	Reactor Trip System
RWCU	Reactor Water Clean-Up
SG	Steam Generator
SI	Safety Injection
SLB	Steam Line Break
SSPS	Solid State Protection System
TMI	Three Mile Island

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1. INTRODUCTION

1.1 Background

By the year 2014, 48 commercial nuclear power plants in the United State are projected to reach 40 years of operation [Ref. 1]. As this aging population increases, degradation of the systems and components in these plants becomes an important consideration to the nuclear community. The distribution by plant age for the operable plants is shown in Figure 1.1.

Almost a decade ago, the United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research, Division of Engineering initiated an ambitious research program for assessing aging effects on equipment and systems in nuclear power plants. The program, entitled "Nuclear Plant Aging Research" (NPAR), seeks to improve the operational readiness of systems and components that are vital to nuclear power plants and their safety by understanding and managing aging degradation. Since its inception, the NPAR program has produced a wealth of knowledge on the aging of systems and components; so far, 18 systems have been studied (primarily by the national laboratories), and this report is the 19th and the last study. By the end of 1995, 29 components will have been studied [Ref. 2]. Most of the information is published in NUREG reports; some examples are given in References 3 through 10. The insights gained during the NPAR program are summarized in Reference 1, which is designed for use by industry and by the NRC in understanding and managing the aging of systems, structures, and components in nuclear power plants.

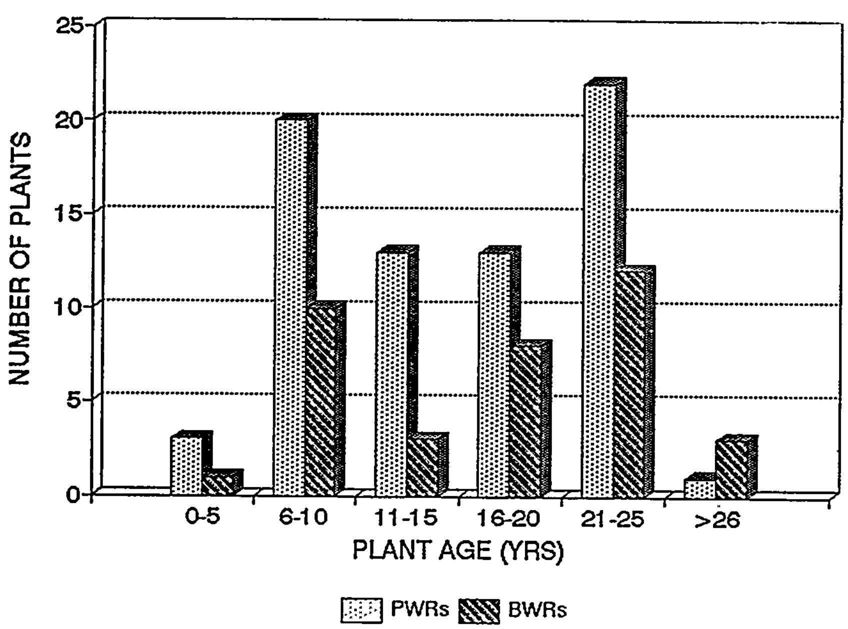


Figure 1.1 Age distribution of operable plants

The main goals of the Nuclear Plant Aging Research (NPAR) program are to understand aging and to identify ways to manage the aging of safety-related systems, structures, and components (SSCs) in nuclear power plants. The following are the technical objectives of the program [Ref. 11].

- identify and characterize aging effects which, if unmitigated, could cause degradation of SSCs and thereby impair safety,

- develop supporting data and information to facilitate management of age-related degradation,

- identify methods of inspecting, surveillance, and monitoring of SSCs, or of evaluating their residual-life to ensure the timely detection of significant aging effects before safety function is lost,

- evaluate the effectiveness of storage, maintenance, repair, and replacement practices in mitigating the effects of aging and diminishing the rate and extent of aging degradation,

- provide technical bases and support for the License Renewal Rule and the license renewal process, and develop a regulatory guide on the format and technical information content for renewal applications.

1.2 Objectives and Scope of This Study

The primary containment is classified as an Engineered Safety Feature due to its importance in containing loss of coolant accident (LOCA) releases and shielding equipment. Consequently, the components related to the containment's function are subject to stringent regulations and standards, including 10 CFR 50, Appendix J [Ref. 12]. When a design basis accident (DBA) occurs, either a LOCA or a main steam line break (MSLB), the containment isolation (CI) function is the engineered safety feature (ESF) that prevents and minimizes the release of the radioactivity. Section 3 discusses in more detail the importance of the CI function to the safety of the plant personnel and the public.

In accordance with the NRC-NPAR Program Plan, the primary goals of this phase I Containment Isolation Function Aging study were to identify and characterize aging and service-wear effects which, if unchecked, could cause components and systems to degrade and thereby impair plant safety; these goals include an aging characterization of the system and its important components. In addition, a preliminary review of current inspection, surveillance, and maintenance practices was made to identify areas where improvements could more effectively detect and mitigate aging degradation.

The components in the CI function are designed to handle the extreme loads (temperature, pressure, humidity, radioactivity) associated with DBAs. Thus, it is very important to develop proper aging management programs for them so that they can function when needed. Aged components will degrade further during DBAs, probably at accelerated rates due to the harsh environment, which may cause these components to fail. One of the main goals of this study was to generate information needed for their proper management so that a minimum number of CI components fail during DBAs. The following were the specific objectives of this study:

- characterize the aging processes of the components in the CI function,

- identify trends associated with aging of different components in the CI function,

- understand the causes and mechanisms of aging for each different type of component,

- make a preliminary review of inspection, surveillance, and monitoring practices, and their effectiveness.

In this study, a large number of operating data were analyzed, from both old and new plants. Thus, one underlying objective of this study was to gather information from analyzing the operating experience of older plants to predict what to expect for newer plants, so that effective aging management programs can be prepared.

Due to the many data on failures of the components for the containment isolation function, the scope of this study was limited to the failures in the Westinghouse plants (PWRs) and the General Electric plants (BWR). This decision was an arbitrary one, based only on the number of the operating plants.

For Westinghouse PWRs, the containment isolation (CI) function consists mainly of containment isolation valves and valve operators. The automatic containment isolation is initiated by engineered safety features actuation system (ESFAS), which is not included in this study because it was a subject of an earlier NPAR study [Ref. 10]. However, for BWRs, most of the containment isolation valves and valve operators are in the nuclear steam supply shutoff system, which also initiates the containment isolation signals. For this reason, the failure data of switches, bistables, radiation detectors, and transmitters in the CI functions of the BWRs were analyzed in this study. More detailed descriptions of the CI functions are given in Section 2.

1.3 Methodology of Analysis

A detailed analysis was performed of the following data bases summarizing the actual operating experience of CI functions:

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

In reviewing these data, decisions were made as to whether or not the failures were aging-related. We note that different reviewers using the same information may reach different conclusions, based upon their personal knowledge and experience. To be consistent, a set of guidelines was used to define what constitutes an "aging-related failure"; these guidelines represent a minimum set of criteria which must be satisfied to classify the failure as aging-related.

The first criterion is that the failure satisfies the NPAR definition of aging [Ref. 11], defined as being the cumulative changes with time that may occur within a component, structure, or system which, if unchecked, may result in loss of function and impairment of safety. Factors causing aging include:

a. natural internal chemical or physical processes during operation,

b. external stressors (e.g., radiation, humidity) caused by storage or the operational environment,

c. service wear, including changes in the dimensions and/or relative positions of individual parts or subassemblies caused by operational cycling,

d. excessive testing, and

e. lack of maintenance.

In addition, the circumstances of the failure must meet the following criteria:

1. The component must have been in service for at least 6 months before the failure; this eliminates failures of infant mortality.

2. The failure must be the result of operation or service over some period, and not be due to a temporary, instantaneous event.

Three categories of aging were used; "aging", "possibly aging", and "non-aging." When the failure met all the criteria above, and its description clearly indicated the cause as aging, it was classified as "aging." In many cases, terms such as aging, old, obsolete, or age of the component are used in the description. When there was insufficient information, but the mode, and cause of failure, and the component's age indicate that the failure was probably due to aging, it was categorized as "possibly aging." When human factors, including errors, procedural problems, and design problems were the clear causes, the failure was categorized as "non-aging."

The data classified as "aging" and "possibly aging" were combined and termed "aging-related failures." Only these failures were analyzed to characterize aging degradation. Each data base was analyzed to obtain and the following information:

- Effects of aging on the failure frequencies for important components
- Effects of aging-related failures of components on plant operation
- Aging fraction of component failures
- Modes of component failure
- Causes of component failure

Plant-specific data for the CI function supplemented the generic data bases; this included six years of maintenance records from an operating BWR. The data represented a plant age of less than 10 years. An analysis, similar to that used for the data base records, was made on the plant data to identify aging characteristics. This included a determination of the aging fraction, modes and causes of failure as well as an identification of the components failing most frequently. These findings then were compared with those from other data bases to check the results.

1.4 Organization of the Report

In Section 2, the containment isolation functions for both PWRs and BWRs are described. Section 3 of this report discusses the safety significance of the containment isolation function in mitigating the consequences of a design basis accident (DBA) for PWRs and BWRs. Section 4 discusses operational and environmental stresses, which cause aging degradation. Operational stresses include normal operational stresses, stresses induced by testing, and those induced by human performance.

Section 5 gives the results of the analyses of PWR component failure data, and those of BWRs are discussed in Section 6. The results of the analysis of operating data from a BWR plant are given in Section 7.

Section 8 reviews current utility practices on testing and maintaining the CI function. The summary and conclusions are given in Section 9. Appendix A describes different types of containments for PWRs and BWRs.

2. DESCRIPTION OF THE CONTAINMENT ISOLATION

The containment isolation components function to isolate the containment from the outside environment in the event of any postulated accident for which isolation is required. The CI function is designed as a engineered safety feature system (Fig. 2.1) to minimize the release of radioactive material by isolating those systems which penetrate the containment and are not essential to mitigate the consequences of the accident. Design criteria for this function are contained in General Design Criteria (GDC) 54 through 57, Regulatory Guide 1.141, NUREG-0737, and Section 6.2.4 of the Standard Review Plan.

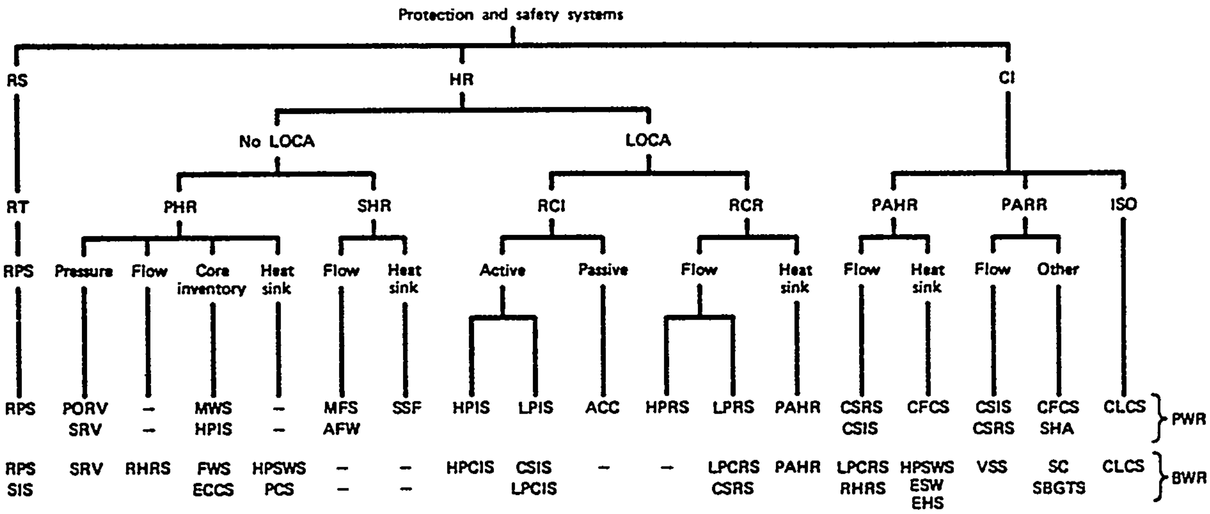
The containment isolation function is not provided by a single, well-defined system in either GE BWRs or Westinghouse PWRs. The containment isolation valves at each penetration are considered part of the associated fluid system (e.g., main steam, residual heat removal) rather than the containment. In addition, automatic isolation may be initiated by the nuclear steam-supply shutoff system (General Electric), engineered safeguards actuation system (Westinghouse), and/or by individual system isolation logic. (Actuation of the containment isolation function is discussed in more detail later).

The containment isolation function, as described in the final safety analysis reports (FSARs), consists of containment isolation valves and valve operators and, in certain cases, the isolation logics and instrumentation that provide automatic closure signals. The containment is isolated by the automatic isolation of non-essential lines which penetrate the primary containment. Those fluid lines penetrating primary containment which support engineered safety feature (ESF) systems have remote power-operated isolation valves so that they can be closed from the control room.

Among the same type of reactors, either PWR or BWR, different containment designs have arisen due to the evolution of the designs of the reactor and containment. PWRs have three different types of containments; large dry, ice condenser, and subatmospheric. For BWRs, there are 6 different classes of reactors, BWR 1-6, with three different types of containments, Mark I, II, and III. Appendix A briefly describes these different containment types for both PWRs and BWRs.

In this report, all the operating data from the Westinghouse plants are combined as were those for the General Electric plants. This approach was justified because the aging parameters of the different containment types from the same manufacturers should be about the same, except the inerted containment environment for some BWRs (discussed in the later sections). The following two sections, Sections 2.1 and 2.2, describe the containment isolation functions for GE BWRs and Westinghouse PWRs, respectively.

The primary containment is penetrated at many locations. In addition to valves, piping, and hatches for personnel and equipment, the isolation function may include a penetration pressurization system. Pressurization with nitrogen gas is used to charge a closed volume created by redundant seals in the sleeve cavities on some of the containment piping and electrical penetrations. The cavity pressure is maintained at a level above that expected to occur inside the containment. Monitoring the pressure during normal operation indicates the leak-tightness of the sleeve. Because there are few reported failures of the penetration system, no detailed description of it is given in this section. For both PWRs and BWRs, the penetration system is treated as a separate system from the CI function, even though they are part of the CI function. The reason may be that the hatches are always closed, and the penetration pressurization system is a passive system, not activated by the isolation signals. However, these penetrations were also included in our analysis, even though the number was small.



General safety functions

CI	Containment Integrity
HR	Heat Removal
RS	Reactor Subcriticality

Condition safety functions

ISO	Containment Isolation
PAHR	Post-Accident Heat Removal
PARR	Post-Accident Radioactivity Removal
PHR	Primary Heat Removal
RCI	Reactor Coolant Injection
RCR	Reactor Coolant Recirculation
RT	Reactor Trip
SHR	Secondary Heat Removal

Typical engineered safety features (ESFs)

PWR

ACC	Accumulators
CFCS	Containment Fan Cooling System
CLCS	Containment Leakage Control System
CSIS	Containment Spray Injection System
CSRS	Containment Spray Recirculation System
ESFCS	Engineered Safety Features, Containment Systems
HPIS	High Pressure Injection System
HPRS	High Pressure Recirculation System
LPIS	Low Pressure Recirculation System
MFS	Main Feedwater System
MWS	Makeup Water System
PORV	Power-Operated Relief Valve
RPS	Reactor Protection System
SHA	Sodium Hydroxide Addition
SRV	Secondary Relief Valves
SSRS	Secondary Steam Relief System

BWR

BIS	Boron Injection System
CLCS	Containment Leakage Control System
CSIS	Core Spray Injection System
CSRS	Core Spray Recirculation System
ECCS	Emergency Core Cooling System
EHS	Emergency Heat Sink
ESFCS	Engineered Safety Features Containment Systems
ESW	Emergency Service Water
FWS	Feedwater System
HPCIS	High Pressure Coolant Injection System
HPSWS	High Pressure Service Water System
LPCIS	Low Pressure Coolant Injection System
LPCRS	Low Pressure Coolant Recirculation System
PCS	Power Conversion System
RCICS	Reactor Core Isolation Cooling System
RHRS	Residual Heat Removal System
RPS	Reactor Protection System
RWCS	Reactor Water Cleanup System
SBGTS	Standby Gas Treatment System
SC	Secondary Confinement
SRV	Secondary Relief Valves
VSS	Vapor Suppression System

Figure 2.1 Reactor protection and accident mitigation systems [Ref. 13]

Most of the data that we analyzed were on valves and valve operators, mainly because these are the components which provide containment whose failures are reported most often to NPRDS. When the failures of bistables and switches are reported as those of CI function, they also were analyzed.

2.1 Description of the GE BWR Containment Isolation Function [Refs. 14-18]

Operating BWR plants in the United States have three different types of containments: 67% (24) have Mark I containment; 22% (8) and 11% (4) have Mark II and Mark III containment, respectively (Figure 2.2). General Electric boiling water reactors have a pressure suppression containment. Regardless of the containment design (Mark I, II or III), each BWR has several hundred fluid-system penetrations (associated with tens of systems), multiple electrical penetrations, and several equipment/personnel hatches. Many of these penetrations are open during normal operation. In this study, no distinction was made between containment designs, and all of the component failures are combined for analyses. For the purposes of this study, the BWR containment isolation function consists of the isolation valves, and the nuclear steam supply shutoff system (NS^4) isolation logic. In addition, several systems such as the high pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), and residual heat removal (RHR) systems have isolation logics that can isolate specific containment penetrations.

2.1.1 Containment Isolation Valve Arrangements

General Design Criterion (GDC) 54 of 10CFR50 Appendix A requires that each of the piping systems penetrating the primary containment has capabilities for leak detection, isolation, and containment with redundancy, reliability, and performance attributes consistent with their importance to safety. The ability to periodically demonstrate the operability of valves and to determine if leakage is within acceptable limits also is required.

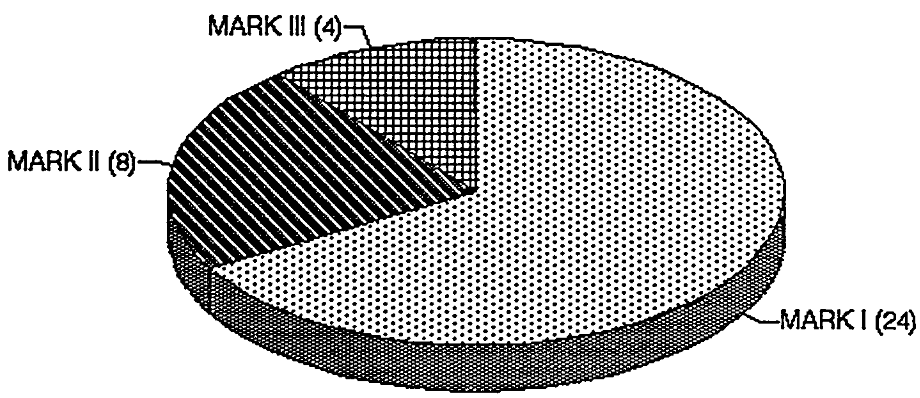


Figure 2.2 Containment types for BWRs

The number of containment isolation valves (CIVs) and their locations (i.e., inside and/or outside containment) are specified in GDC 55 through 57. Since these GDC encompass all light water reactors and are not written specifically for the BWR containment configuration, some BWR penetration designs do not correspond exactly to an individual GDC. In recognition if this, the isolation criteria within these GDC provides for alternate designs with reliability and performance capabilities that reflect the importance to safety of isolating the piping system.

Typically, for piping penetration, two isolation valves are used; one is located close to the containment wall on the inside, and the other is on the outside. However, for the closed systems inside, only one exterior valve is acceptable, as shown in Figure 2.3. Consequently, the number of valves inside and outside is not equal, with more outside valve configurations. This figure shows 14 different configurations of valves allowed for the containment isolation function.

GDC 55 specifies the arrangements of the containment isolation valves for those lines that penetrate the containment and are part of the reactor's coolant pressure boundary. Unless it can be demonstrated that the containment isolation provisions for a specific type of line (such as instrument lines) are acceptable on some other basis, this criterion requires two containment-isolation valves (one inside and one outside containment). The valves may be locked closed, automatic isolation valves, or a combination thereof. A simple check valve cannot be used for the automatic isolation valve outside the containment. As shown in Table 2.1, typical BWR penetrations that conform to GDC 55 include the main steam lines, reactor water clean-up (RWCU) suction line, the RCIC and HPCI steam supply lines, and the RHR head spray line.

Penetrations that also fall under the scope of GDC 55 are the feedwater lines and the high pressure core spray (HPCS) (or HPCI), low pressure coolant injection (LPCI), and low pressure core spray (LPCS) injection lines. Since these lines can potentially provide makeup to the reactor vessel, automatic isolation is not always warranted. These penetrations generally have an inboard check valve and an outboard gate valve with remote power-operated isolation capability.

GDC 56 applies to lines that connect directly to the containment's atmosphere and penetrate primary containment. The isolation requirements are identical to GDC 55. Lines which satisfy the requirements of GDC 56 include the containment vent and purge lines, floor and equipment drain lines, and the containment spray lines.

Other lines which fall under the scope of GDC 56 include the various emergency core cooling system (ECCS) pump suctions from the suppression pool and the test return lines. However, placing a CIV inside the containment could subject it to post-LOCA hydrodynamic forces, and result in an overall decrease in the system's reliability. Since these lines terminate in the suppression pool, they are isolated from the containment's air space. These lines generally credit this water seal, and feature a single, normally open (ECCS suction) or normally closed (test return) motor-operated CIV outside the containment.

GDC 57 states that each line that penetrates the primary containment and is neither a part of the reactor coolant pressure boundary (RCPB) nor connected directly to the containment's atmosphere is required to have at least one automatic, locked closed, or remote power-operated CIV located outside the containment. A simple check valve cannot be used as the automatic isolation valve. The reactor recirculation pump and motor cooling lines and the traversing incore probe (TIP) drive guide tubes can fall under the requirements of GDC 57.

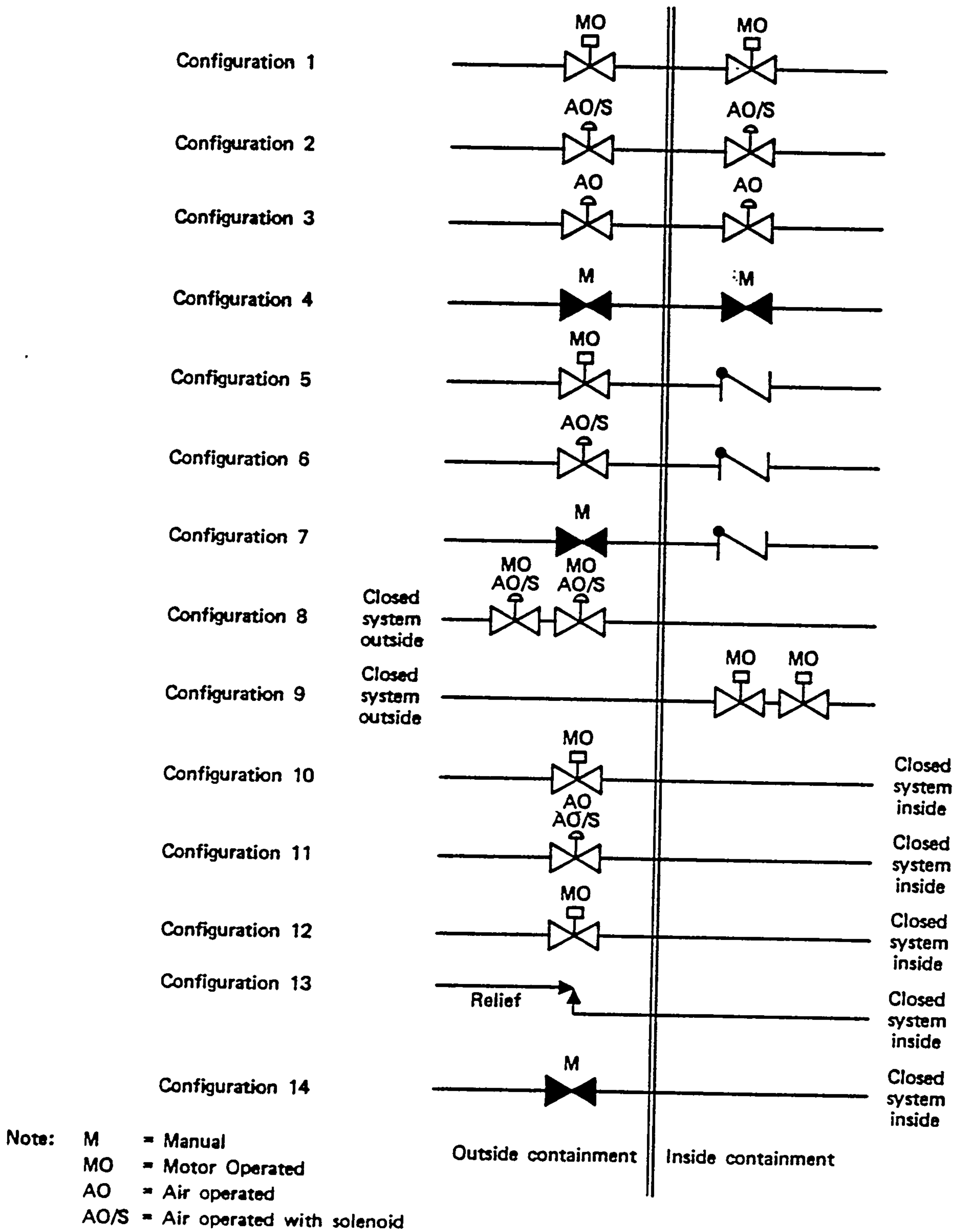


Figure 2.3 Allowed configurations of containment isolation valves [Ref. 19]

Table 2.1 Description of Typical BWR Penetrations

Penetration Description	NS[4] Isolation Group	Isolation Valve Type	Location	Normal Position	GDC	Isolation Signals
Main Steam Lines (MSIVs and MSIV Drains)	1	AO Globe AO Globe MO Gate MO Gate	Inside Outside Outside Outside	Open Open Open Closed	55	A,E,F,G,H,O,RM A,E,F,G,H,O,RM A,E,F,G,H,O,RM A,E,F,G,H,O,RM
MS Line Drain	1	MO Gate MO Gate	Inside Outside	Open Open		A,E,F,G,H,O,RM A,E,F,G,H,O,RM
Reactor Recirc Sample	2	SO Globe SO Globe	Inside Outside	Closed Closed	55	B,E B,E
Drywell/Wetwell Ventilation Supply Lines	3	AO Butterfly AO Butterfly SO Globe SO Globe	Outside Outside Outside Outside	Closed Closed Open Open	56	B,D,I,RM B,D,I,RM B,D,I,RM B,D,I,RM
Drywell/Wetwell Ventilation Exhaust Lines	3	AO Butterfly AO Butterfly SO Globe SO Globe	Outside Outside Outside Outside	Closed Closed Open Open	56	B,D,I,RM B,D,I,RM B,D,I,RM B,D,I,RM
Reactor Closed Cooling Inlet Header	4	MO Gate MO Gate	Outside Outside	Open Open	56	B,D,RM B,D,RM
Reactor Closed Cooling Outlet Header	4	MO Gate MO Gate	Outside Inside	Open Open	56	B,D,RM B,D,RM
Suppression Pool Cleanup Suction/Return	4	MO Gate MO Gate	Outside Outside	Closed Closed	56	B,D,RM B,D,RM
Equipment and Floor Drains	4	AO Gate AO Gate	Outside Outside	Open Open	56	B,D,RM B,D,RM
Reactor Recirc Valve Hydraulic Lines	4	SO Globe SO Globe	Outside Outside	Open Open	56	B,D,RM B,D,RM
Radiation Monitor Supply Lines	4	SO Globe SO Globe	Inside Outside	Open Open	56	B,D,RM B,D,RM
Traversing In Core Probe	5	SO Ball Shear	Outside Outside	Closed Open	56	C,D,RM RM

Table 2.1 (Cont'd)

Penetration Description	NS[4] Isolation Group	Isolation Valve Type	Location	Normal Position	GDC	Isolation Signals
RHR Test Return Line to Supp. Pool	5	MO Globe MO Gate	Outside Outside	Closed Closed	56	C,D,RM C,D,RM
RHR SD Cooling Suction	6	MO Gate MO Gate	Inside Outside	Closed Closed	55	C,K,N,RM C,K,N,RM
RHR SD Cooling Return	6	Check MO Globe	Inside Outside	Closed Closed	55	C,K,N,RM C,K,N,RM
RPV Head Spray	6	Check MO Globe	Inside Outside	Closed Closed	55	C,K,N,RM C,K,N,RM
Reactor Water Cleanup Suction Line	7	MO Gate MO Gate	Inside Outside	Open Open	55	B,J,M,P,RM B,J,L,M, P,RM
RCIC Steam Supply Line	NA	MO Gate MO Gate	Inside Outside	Open Open	55	S S
Feedwater Lines	NA	Check PC Check MO Gate	Inside Outside Outside	Open Open Open	55	 RM RM
HPCS Injection	NA	Check MO Gate	Inside Outside	Closed Closed	55	RM RM
LPCS Injection	NA	Check MO Gate	Inside Outside	Closed Closed	55	RM RM
Drywell Spray	NA	MO Gate MO Gate	Outside Outside	Closed Closed	56	S S
HPCS Pump Suction from SP	NA	MO Gate	Outside	Closed	56	S
LPCS Pump Suction	NA	MO Gate	Outside	Closed	56	S
HPCS Test Line	NA	MO Globe	Outside	Closed	56	S

Abbreviations for Table 2.1

Valve Operator Type

AO air operator
MO motor operator
PC positive closing
SO solenoid operator

Isolation Signals

A Reactor Vessel Low Water Level 1
B Reactor Vessel Low Water Level 2
C Reactor Vessel Low Water Level 3
D High Drywell Pressure
E Main Steam Line High Radiation
F Main Steam Tunnel Area and Turbine Building Area High Temperature
G Main Steam Line High Flow
H Main Turbine Inlet Low Steam Pressure
I Containment or Drywell Ventilation Exhaust Radiation High
J Reactor Water Cleanup (RWCU) System High Differential Flow
K RHR System Area High Ambient Temperature or High Differential Temperature
L High Temperature at the RWCU Non Regenerative Heat Exchanger Outlet
M Standby Liquid Control (SLC) System Actuation
N Reactor Vessel Pressure High (Shutdown Cooling Mode)
O Low Main Condenser Vacuum
P RWCU Pump Room or Heat Exchanger Area High Temperature or High Ventilation Differential Temperature
RM Remote Manual
S System Specific Isolation Logic (Non NS[4])

Table 2.2 shows the numbers of different containment isolation valves for a typical BWR-6 plant.

Table 2.2 Typical Number of CI Valves in BWRs

Containment	Outside the Containment	Inside the Containment	Total
Mark I	111	129	240
Mark II	40	216	256
Mark III	88	223	311

2.1.2 Isolation Logic

The primary objective of containment isolation is to provide protection against releases of radioactive materials to the environment as a result of accidents in the nuclear steam supply system (NSSS), auxiliary systems, and support systems. This objective is accomplished by automatic isolation of appropriate lines that penetrate the primary containment. Containment isolation is automatically actuated when specifically defined limits are reached.

It is neither necessary, nor desirable, to close every isolation valve simultaneously with a common isolation signal. For example, if a process pipe were to rupture in the drywell, it would be important to close all lines open to the drywell, and some effluent process lines, such as the main steam lines. However, under these conditions, it is essential that the containment and core-cooling systems are operable. Therefore, several specific signals are used to isolate the various process and safety systems.

The containment isolation system is divided into twelve designated containment isolation groups.[1] Each group consists of a different group of valves and/or equipment that will automatically isolate, secure, or startup. The nuclear steam supply shutoff system (NS^4) supplies the isolation signals to Groups 1 through 7. Isolation Groups 8 through 12 consist of the RCIC and the ECCS systems, each with their own isolation logic. These systems were the subject of previous aging studies [Ref. 20, 21], and are not discussed here.

Nuclear Steam Supply Shutoff System (NS^4)

The nuclear steam supply shutoff system provides isolation signals to the first seven groups of the containment isolation function. From information provided by the reactor's process instrumentation, the NS^4 determines, which system(s) should be isolated, and provides the appropriate signals. Local sensor elements send input signals to the NS^4 from the nuclear boiler system, standby liquid control system, reactor protection system (RPS), main condenser system, and leak detection system, providing information on selected plant parameters to the NS^4 logic. Table 2.3 presents the typical isolation signals for each NS^4 isolation group.

NS^4 Logic for Group 1 is arranged in a one-out-of-two-taken-twice manner, similar to the RPS logic. The main steam isolation valve (MSIV) isolation logic portion of Group 1 is composed of trip system "A" consisting of channels "A" and "C", and trip system "B" with channels "B" and "D". This is not an inboard-outboard type arrangement. A trip signal in either channel will trip its corresponding system. Both trip systems must be activated to initiate an MSIV closure. When an isolation is initiated, all eight MSIVs receive closure signals.

The main steam line drain (MSLD) portion of Group 1 is an inboard-outboard logic. Unlike the MSIV system, the MSLD trip system is arranged with channels "A" and "B" in trip system "A", and channels "C" and "D" in trip system "B". MSLD isolation logic is a two-out-of-two logic requiring both channels in a system to trip causing the corresponding isolation of the inboard or outboard CIV.

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[1]This system description is generally based on a BWR 6 with a Mark III containment. Other BWR designs are conceptually similar, but may have different terminology, systems, isolation groups, and closure signals.

Table 2.3 Typical BWR NS[4] Initiating Signals

Group 1 - Main Steam Isolation Valves (MSIVs) and Main Steam Line Drains
Reactor Vessel Low Water Level 1 Main Steam Line High Radiation Main Steam Tunnel Area and Turbine Building Area High Temperature Main Steam Line High Flow Main Turbine Inlet Low Steam Pressure (Mode Switch in Run) Main Condenser Low Vacuum
Group 2 - Reactor Water Sample Valves
Reactor Vessel Low Water Level 2 Main Steam Line High Radiation
Group 3 - Primary and Secondary Containment Ventilation and Purge Systems
Reactor Vessel Low Water Level 2 High Drywell Pressure Containment or Drywell Ventilation Exhaust Radiation High[2]
Group 4 - Miscellaneous Balance of Plant (BOP) Systems (including: Reactor Closed Cooling Water and Instrument Air
Reactor Vessel Low Water Level 2 High Drywell Pressure
Group 5 - Residual Heat Removal (RHR) System and Traversing In-Core Probe (TIP) System
Reactor Vessel Low Water Level 3 High Drywell Pressure
Group 6 - Residual Heat Removal System (Shutdown Cooling Mode)
Reactor Vessel Low Water Level 3 RHR Area High Temperature or High Ventilation Differential Temperature Reactor High Pressure (Shutdown Cooling Mode)

[2]Some plants do not consider this an NS[4] isolation signal

Table 2.3 (Cont'd)

Group 7 - Reactor Water Cleanup (RWCU) System
Initiating Signal Reactor Vessel Low Water Level 2 RWCU High Differential Flow RWCU Non Regenerative Heat Exchanger Outlet High Temperature Standby Liquid Control System Initiation RWCU Pump Room or Heat Exchanger Area-High Temperature or High Ventilation High Differential Temperature

The isolation logic for the remaining isolation groups (2 through 7) also is arranged in an inboard-outboard type logic. Channels "A" and "B" provide input to trip system "A" (outboard), and channels "C" and "D" provide input to trip system "B" (inboard).

Logic power to the NS[4] is 120VAC from RPS buses A & B. Bus "A" will supply the outboard logic channels, while Bus "B" supplies the inboard logic channels.

AC power for the NS[4] is supplied for instruments, status lights, and isolation valve solenoids; this power is 60Hz, 120VAC, single-phase power from various buses. The design basis of these power supplies is such that the loss of a logic power supply will not prevent an isolation.

2.2 Description of Westinghouse PWR Containment Isolation Function [Refs. 22-25]

The Westinghouse PWR plants in operation in the United States also have three different types of containments; about 70% (35) have large dry containment, and 16% (8) and 14% (7) have ice condenser and subatmospheric containment, respectively (Figure 2.4). Although Westinghouse PWRs typically have fewer fluid system penetrations than BWRs, nonetheless a significant number are open during normal operation. Automatic closure is necessary to:

1. Minimize and limit the atmospheric release of radioactive materials in the event of a LOCA by isolating those lines penetrating the containment that are not required for operating the engineered safety systems.

2. Avoid the reactivity effects that could result from excessive cooldown of the reactor coolant system, and prevent overpressurization of the containment in the event of a steam line break accident by isolating the containment and the steam generators.

For this study, the Westinghouse containment isolation function was defined as the containment isolation valves and valve operators associated with the fluid systems that penetrate the containment. However, the logic system(s) that provide closure signals to selected valves are also described briefly in this section.

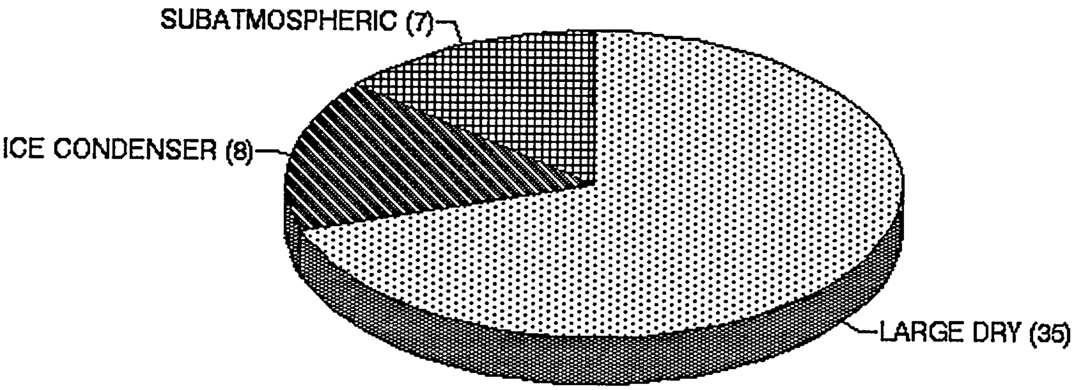


Figure 2.4 Containment types for Westinghouse PWR plants

2.2.1 Arrangements of the Containment Isolation Valves

As discussed in Section 2.1.1, General Design Criteria (GDC) 54 to 57 specifies the general design capabilities of each piping system that penetrates primary containment. GDC 54 requires leak detection, isolation and containment capabilities for each fluid penetration, consistent with their importance to safety.

GDC 55 states the arrangements for the containment isolation valves for those lines that are part of the reactor coolant pressure boundary (RCPB). They require two valves, one inside and one outside the containment. The CIVs can be locked closed or automatically isolated. Check valves cannot be used for automatic isolation outside the containment. With the exception of charging, letdown, and safety injection, Westinghouse PWRs typically do not have RCPB lines which penetrate the primary containment.

The Westinghouse design typically has several exceptions to GDC 55. The RHR pump suction lines from the reactor coolant system (RCS) hot legs each contain two remote power-operated valves that are closed during normal operation. The valves are interlocked and cannot be opened when the RCS pressure is greater than the RHR design pressure. Lines which penetrate containment and perform a safeguard function, such as the RHR, safety injection and reactor coolant pump seal injection lines, may be required to remain open following an accident. Their associated containment valves have remote power-operated closure capability to allow them to be isolated after the safeguards function is completed. Table 2.4 illustrates selected Westinghouse CIV arrangements and the applicable GDCs to which they are designed.

Table 2.4 Description of Typical Westinghouse PWR Penetrations

Penetration Description	Isolation Valve Type	Location	Normal Position	GDC	Isolation Signals
Main Steam Lines (MSIVs and Drains)	AO Globe AO Globe AO Globe AO Globe	Outside Outside Outside Outside	Open Closed Open Open	57	3 3 3 1
Feedwater Lines	EH Gate AO Gate AO Gate AO Gate	Outside Outside Outside Outside	Open Closed Closed Closed	57	4 4 4 4
CVCS Normal Letdown	AO Globe AO Globe AO Globe AO Globe	Inside Inside Inside Outside	Cycles Open Cycles Open	55	1 1 1 1
Steam Generator Blowdown	AO Globe AO Globe	Outside Outside	Open Open	57	1 1
Reactor Coolant Sample	SO Globe SO Globe	Outside Inside	Open Open	55	1 1
Instrument Air Supply	Check AO Gate	Inside Outside	Open Open	56	- 1
Service Water from Non-Nuclear Fan Coils	AO Butterfly AO Butterfly	Inside Outside	Open Open	56	1 1
Component Cooling Water to RCPs	Check MO Gate	Inside Outside	Open Open	57	- 2
Component Cooling Water from RCP Thermal Barriers	MO Gate MO Gate	Inside Outside	Open Open	57	2 2

Table 2.4 (Cont'd)

Penetration Description	Isolation Valve Type	Location	Normal Position	GDC	Isolation Signals
Containment Atmosphere Purge Makeup	AO Butterfly AO Butterfly AO Butterfly AO Butterfly	Inside Inside Outside Outside	Closed Cycles Closed Cycles	56	5 5 5 5
Containment Atmosphere Purge Exhaust	AO Butterfly AO Butterfly AO Butterfly AO Butterfly	Inside Inside Outside Outside	Closed Cycles Closed Cycles	56	5 5 5 5
CVCS Normal Charging	Check MO Gate	Inside Outside	Open Open	55	 6
High Head SI to Cold Legs	Check MO Globes MO Gate	Inside Inside Outside	Closed Locked Throttled Closed	55, 57	- - 6
Containment Spray	Check MO Gate	Inside Outside	Closed Closed	56, 57	- 6
Containment Sump to RHR Pump	MO Gate	Outside	Closed	56	6
Service Water to Fan Coolers	MO Butterfly	Outside	Open		6

Abbreviations for Table 2.4

Valve Operator Type

AO air operator
MO motor operator
SO solenoid operator
M manual
EH electro-hydraulic operator

Isolation Signals

1. Containment Isolation Actuation Signal Phase A
2. Containment Isolation Actuation Signal Phase B

Isolation Signals (Cont'd)

3. Main Steam Isolation Signal
4. Main Feedwater Isolation Signal
5. Containment Ventilation Isolation Signal
6. Remote Manual

GDC 56 covers lines that penetrate the primary containment and are in direct communication with the containment's atmosphere. The isolation requirements are identical to GDC 55. Such lines are of two types; the first communicates directly with the atmospheres inside and outside the containment, for example, the atmosphere purge line. The second type encompasses those penetrations for non-nuclear safety class lines penetrating the containment, and includes service air and fire protection.

Other penetrations that fall within the scope of GDC 56 include the residual heat removal (RHR), and containment spray (CS) pump suction lines from the containment's recirculation sumps. These penetrations do not conform exactly to the requirements of GDC 56, but meet its intent. Each line has a remote power-operated gate valve. These valves are enclosed in valve chambers that are leaktight at containment design pressure. Each line from the containment sump to the valve is enclosed in a separate concentric guard pipe, which also is leaktight. A seal is provided so that neither the chamber, nor the guard pipe is connected directly to the containment sump or to its atmosphere.

GDC 57 requires at least one automatic, locked closed or remote power-operated CIV located outside the containment for lines that are not a part of the RCPB nor connected to the containment's atmosphere. A simple check valve cannot be used as the automatic isolation valve. The GDC implicitly credits a closed system as the second isolation barrier. A closed system is one which satisfies all of the following requirements:

1. The system does not communicate with either the Reactor Coolant System or the containment atmosphere.

2. The system is protected against postulated missiles and pipe whip.

3. The system is designated seismic Category I.

4. The system meets Quality Group B or C standards.

5. The system is designed to withstand temperatures at least equal to the containment's design temperature.

6. The system is designed to withstand the external pressure from the containment structural acceptance test.

7. The system is designed to withstand environmental and transient conditions resulting from either a loss-of-coolant accident or a main steam line break.

The steam generator shell, and all connected lines are designed as Seismic Category I, Quality Group B, and are missile protected. This design allows these components to be considered as an extension of the containment. In conformance with GDC 57, isolation valves are provided outside the containment on all lines connected to the shell side of the steam generator. These valves are either normally closed or close automatically to effect steam generator isolation, except for the steam supply lines to the auxiliary feed pump turbine and safety valve lines which may operate intermittently.

The Westinghouse design can also credit closed systems outside the containment as the second isolation barrier to satisfy the GDC requirements. Containment penetrations for the component cooling water (CCW), RHR, Safety Injection (SI), portions of the chemical and volume control system (CVCS), and the containment spray systems may credit a closed system outside the containment. Table 2.5 shows typical number of containment isolation valves for different types of containments of Westinghouse plants.

Table 2.5 Typical Number of CI Valves for Westinghouse Plants

Type of Containment	Inside of Containment	Outside of Containment	Total
Large Dry	78	76	154
Ice Condenser	15 (5 check valves)	82	97
Subatmospheric	57	78	135

2.2.2 Isolation Logic

Containment isolation is automatically actuated by the engineered safety features actuation system (ESFAS). This system measures temperatures, pressures, flows and levels in the reactor coolant system containment and various auxiliary systems.

The ESFAS instrumentation is segmented into three distinct, but interconnected modules:

- Field transmitters or process sensors and instrumentation.
- Signal processing equipment.
- Solid State Protection System (SSPS) including input, logic, and output bays.

<u>Field Transmitters or Sensors</u>

To meet the design demands for redundancy and reliability, as many as four field transmitters or sensors measure plant parameters. In many cases, transmitters or sensors that input to the ESFAS are shared with the Reactor Trip System (RTS). In some cases, the same channels also provide inputs from the control system.

<u>Signal Processing Equipment</u>

Generally, three or four channels of process control equipment are used for the signal processing or plant parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and compares the measured input with setpoints established by safety analyses. If a measured value exceeds a predetermined setpoint, an output from a bistable is forwarded to the SSPS for a decision evaluation. The channels are separated up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS, while others provide input to the SSPS, the main control board, the unit computer, and one or more control systems.

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient for the required reliability and redundancy. If a parameter is used for input to the SSPS and a control function, four channels with a two-out-of-four logic typically are present.

Solid State Protection System (SSPS)

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables. To meet the redundancy requirements, there are two trains of SSPS, each performing the same functions.

The bistable outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various transients. If a required logic matrix combination is completed, the system will send actuation signals via the master and slave relays to specified components.

Primary containment isolation is just one of the system outputs. The following is a typical listing of the ESFAS protection functions that are used to initiate engineered safety features and supporting equipment:

Safety Injection Actuation Signal
Containment Spray Actuation Signal
Control Room Ventilation Isolation Signal
Containment Isolation Actuation Signal-Phase A
Containment Isolation Actuation Signal-Phase B
Main Steam Isolation Signal
Main Feedwater Isolation Signal
Containment Ventilation Isolation Signal

As shown, the last five ESFAS signals isolate portions of the primary containment. These ESFAS functions are discussed below and summarized in Table 2.6.

Containment Isolation

There are two separate containment isolation signals, Phase A and Phase B. Phase A isolates all automatically isolable process lines, except the component cooling water (CCW), at a relatively low containment pressure indicative of primary or secondary system leaks. For these types of events,forced circulation cooling using the reactor coolant pumps (RCPs) and steam generators is the preferred method of removing decay heat. CCW is required to support RCP operation.

Phase A containment isolation is actuated automatically by a safety injection signal or manually from the control room.

The Phase B signal isolates CCW at a relatively high containment pressure, indicative of a large break LOCA or a steam line break (SLB). For these events, forced circulation using the RCPs is no longer needed.

Table 2.6 Typical ESFAS Containment Isolation Signals and Initiating Parameters[1]

1. Phase A Containment Isolation a. Safety Injection (SI) Signal • Containment Pressure High -1 • Pressurizer Pressure Low • Steam Line Pressure Low • High Differential Pressure Between Steam Lines • High Steam Flow in Two Steam Line Coincident with:[2] Low Low $T_{avg.}$, or Steam Line Pressure Low • Remote Manual SI initiation[1] b. Manual
2. Phase B Containment Isolation Signal a. Containment Pressure High-3 b. Manual
3. Main Steam Line Isolation Signal a. Containment Pressure High-2 b. Steam Line Pressure Low c. Steam Line Pressure Negative $Rate_{avg}$, High d. High Steam Flow in Two Steam Lines Coincident with:[2] Low Low T_{avg}, or Steam Line Pressure Low e. High Steam Line Flow Coincident with Safety Injection and Low Low T_{avg}[3] f. High High Steam Line Flow Coincident with Safety Injection[3]
4. Main Feedwater Isolation Signal a. Steam Generator Level High High b. Safety Injection Signal c. Low T_{avg} and Reactor Trip
5. Containment Ventilation Isolation Signal a. Phase A Containment Isolation b. Manual or Automatic Safety Injection[1] c. Containment Radiation d. Manual Containment Spray (CS) Actuation

Notes:

1. The ESFAS initiation parameters are based on several Westinghouse PWR designs. No single plant ESFAS incorporates all of them.
2. Applicable to three-and four-loop plants.
3. Applicable to two-loop plants.

Isolation of the Main Steam Line

Isolation of the main steam lines provides protection in the event of a steamline break (SLB) inside or outside the containment. Their rapid isolation limits the accident to the blowdown from one steam generator (SG). For a SLB upstream of the main steam isolation valves (MSIVs), inside or outside the containment, closing the MSIVs limits the accident to the blowdown from only the affected SG. For an SLB downsteam of the MSIVs, closing the MSIVs terminates the accident as soon as the steam lines depressurize. MSIVs will also insolate on receipt of a high containment pressure signal which could also result from a LOCA.

Feedwater Line Isolation

The primary functions of the feedwater isolation signal is to stop the excessive flow of feedwater into the SGs. The function is necessary to mitigate the effects of the resulting high water level in the SGs, which could cause excessive cooldown of the primary system.

This function is actuated by SG Water Level-High High, or by an SI signal. In the event of a SI, the unit is taken off line and the turbine generator must be tripped. The main feedwater (MFW) System also is taken out of operation and the auxiliary feedwater (AFW) System is automatically started.

Containment Ventilation Isolation (CVIS)

The containment ventilation system can provide a path from its atmosphere during normal operation. In addition to the provisions of Phase A and to manual containment-spray isolation, these valves close when radiation in the containment is high.

3. SAFETY SIGNIFICANCE OF THE CONTAINMENT ISOLATION FUNCTION

The containment isolation function is one of the most important measures for mitigating of a design basis accident. Hence, periodic testing is required, and any technical specification violations are reported in accordance with the applicable regulations. As a result, many LERs have been generated related to the CI function. Also, due to the large number of valves and valve operators, and due to periodic testing, many failures of components within the CI function have been reported to the NPRDS. In this section, we discuss the safety significance of the performance of the containment isolation function, and the significance of the reported component failures and the LERs.

3.1 Safety Significance of the Containment Isolation Function

The Containment Isolation Function differs from most of the systems previously addressed in the aging studies because its function is to mitigate, rather than prevent core damage. The PRA treatment of core damage mitigation systems differs from the modeling of the so-called "front line" systems that prevent core damage. The relative importance of a front line system, such as high pressure coolant injection, can be ascertained from its contribution to the plant core damage frequency. However, the importance of containment function requires a more detailed evaluation that considers the risk associated with plant operation. Within this context, risk is defined as the chance (probability) of an undesirable consequence, such as a loss, injury, or death.

Risk = (frequency of core damage) X (probability of a release to the environment) X (the consequences of the release)

A Level 1 PRA is used to estimate the core damage frequency. It consists of definition of accident sequence(s), analyses of plant systems and operation, and development of a data base including initiating event, component failure, and human error. In addition to the estimate of core damage frequency, a level 1 PRA can provide insight into the dominant causes of core damage. However, it cannot estimate the risk associated with plant operation. Many accident sequences with relatively high frequencies have low early containment failure probabilities and therefore, low risk.

In addition to quantifying the accident sequence described above, a level 2 PRA examines the physical processes of the accident and the containment's response. The timing of containment's failure, the mode of failure, and the radionuclide inventory that is released to the environment are estimated for each accident sequence, or group of sequences. Similar releases are binned together into release categories. A Level 2 PRA analysis can give some qualitative or comparative risk insights based on the characteristics of a release category and its associated frequency.

A Level 3 PRA includes an analysis of the radionuclide transport through the environment. It provides a quantitative, plant specific assessment of the public health and economic consequences of the spectrum of postulated plant accidents. An analysis of this scope quantitatively assesses plant risk since it estimates both the frequencies and the consequences of the accident sequences [Ref. 26, 27].

As discussed, a Level 2 PRA can provide qualitative risk insights related to the containment's performance during accidents. In part, release categories are differentiated by the timing and magnitude of the predicted release. As shown in Table 3.1, early releases and significant release fractions have higher consequences. An early release does not allow time for significant reduction in radionuclides due to sedimentation, plate-out, or condensation. In addition, public protective actions are less effective. Large release fractions imply significant failures of isolation, bypass[1], or a gross failure of the containment.

Beginning with the Reactor Safety Study [Ref. 28], PRAs have generally shown that the dominant contributors to risk are accident sequences that bypass the containment or result in its gross structural failure. In contrast, leakage through the closed CIVs has a small effect on total plant risk. Recent studies continue to confirm this insight, even for containment leakage rates that are significantly above design, approaching 100% per day [Ref. 29]. In comparison, the technical specification maximum allowable leakage rate ranges from 0.1% to about 1.0% of the containment's air, by weight, at maximum peak containment pressure.

Table 3.1 Qualitative Evaluation Consequence

Release Category Parameter [1] (Note 1)	Consequence Evaluation Less Serious → More Serious	Notes
Containment Failure Timing	no failure → late → early	2-5
Containment Failure Mode	no failure → large catastrophic or containment bypass	3

Notes:

1. Other parameters including the location of the failure, its mechanism (over temperature, over pressure), the energy and duration of the release can be important for consequence analysis. However, for the risk insights associated with failures of the CIS, failure timing and mode are sufficient.
2. Containment failure timing is based on the onset of core damage. Failure time can be influenced by the accident sequence. A loss of decay heat removal generally would be associated with a late overpressurization failure. An ATWS sequence could challenge containment early. Containment bypass sequences, by definition, are early failures.
3. No failure includes containment isolation valve (CIV) leakages up to 100% per day. Gross isolation failures such as both CIVs failing to close in systems that communicate with the environment are not included.
4. Late containment failure will generally allow time for protective actions, such as evacuation, and a reduction of the health effects (risk) associated with a sequence.
5. Late failure can allow deposition mechanisms such as settling and plateout to reduce the release to the environs.

Other PRA insights related to containment isolation include:

- Leakage paths equivalent to 1 inch in diameter for BWRs, and 4 inches in diameter for PWRs, do not have significant offsite consequences. In general, Integrated Leak Rate Test (ILRT) failures fall within this category. Ruptures of the CI valve body and the gaskets or failures of the cover flanges can have significant consequences.

[1]Such as a LOCA outside the containment with a failure to isolate.

- Those lines that form closed systems outside the containment with pressure retention capability in excess of the containment's ultimate pressure, provide an additional level of isolation. They can be considered an extension of the containment.

This information will be used in Sections 5 and 6, to focus the NPRDS search on those penetrations that are more risk-significant based on accident sequence i.e., containment bypass scenarios, line sizes, and piping configuration outside the containment.

3.2 Significance of Reported Failures Related to CI Function

The NRC, as part of the Three Mile Island (TMI) Action Plan, evaluated the need for revisions to the method used to test the containment integrity to ensure that there were no undetected gross openings (i.e., no undetected gross loss of containment isolation capability). This concern arose following a discovery that two three inch containment exhaust bypass valves were left open for approximately 1.5 years at one nuclear plant [Ref. 30]. A review of LERs applicable to containment isolation failures was performed, and an overview and assessment of the loss of capability for containment isolation is reported in NUREG/CR-4220 [Ref. 31].

The results of this analysis, combined with a later more refined review of the same data were:

1. Approximately one-third of the events (1258 of a total of 3447) were leaks that were immediately detected, and therefore, posed little threat to the containment's integrity, and

2. Reportable events for tested components (i.e., valves) residing in direct air paths which could pose a threat of undetected leakage and a potential hazard were 16% of the overall total (552 events of the total of 3447).

The findings of this study concurred with the low risk-significance obtained from the PRA analyses discussed in Section 3.1. The estimated dose increases attributable to undetected leakage were very small, therefore, the increased leakage levels were found not to be risk significant. The test methods used to ensure containment integrity (discussed in Section 8) were found to be adequate.

4. OPERATIONAL AND ENVIRONMENTAL STRESSES

Degradation from aging occurs in materials subjected to stresses over time (e.g., operational and environmental). These processes are well understood when one type of material is exposed to one kind of stress. However, with complex composite materials, or for component's comprised of many different materials (which is typical for most components), these processes are difficult to understand, particularly when accounting for the synergistic effects of several stresses. Extensive laboratory testing and material analyses are necessary to characterize these complex phenomena. Also, because aging is time-dependent, considerable time would be needed to completely understand the characteristics of the aging processes in an operating plant. Therefore, it is essential that inspection and maintenance resources are applied throughout a component's life to detect aging degradation.

Aging is a degradation process (or mechanism) which exists at every level in a plant's hierarchy. If unchecked, it can limit the life of a component, system, or structure, and potentially increase the risk to plant safety; therefore, it is important to understand aging phenomena. The following are some typical aging mechanisms which cause deterioration in a material's mechanical strength or physical properties:

- fatigue stress cycles (thermal, mechanical, or electrical),
- wear,
- corrosion,
- erosion,
- vibration,
- radiation damage,
- embrittlement,
- oxidation (e.g., electrical contact),
- burning (e.g., motor winding),
- cracking or fracture, and
- accumulation of dirt/corrosion products/foreign material.

Each mechanism can occur in various materials when exposed to particular operating and environmental conditions. Abnormal conditions, such as mechanical and electrical transients, pipe breaks, exposure to harsh environments or accidents accelerate the aging process, thus weakening the material faster than normal.

This section discusses the operational, environmental, and accident parameters which can degrade the mechanical strength or electrical/chemical properties of components which comprise the containment isolation function. They include system and component level stresses, including those induced by testing, human factors, environmental parameters, and their synergistic effects.

4.1 System and Component Level Stresses

The components which provide the containment isolation function consist of active components that belong to many different systems, except the penetrations for electric cables and piping which are passive. Thus, all the aging mechanisms and stresses listed above are applicable to the CI components. Additionally, due to the importance of the CI function, they are tested more rigorously than non-safety related components. The stresses induced by testing are discussed in the following section.

Since most of the valves and valve operators that provide containment isolation function are active components, they experience operational stresses caused by the fluids they carry and by being exercised during the operation. The severity of operational stresses largely depends upon the pressure and temperature of the fluid. A previous aging study indicated that there exist differences in the frequency, modes, and causes of failure between valves handling water and steam [Ref. 20]. This study showed that the frequencies of failure for steam valves were much higher than those for the water valves due to harsher environment, mainly high temperature. Also, for water valves, in the earlier years, the main mode of failure was internal leakage caused primarily by worn seats and accumulation of corrosion products; however, in the later years, the main failure mode was external leakage caused by worn packing and worn gaskets; thus, it takes some years until packing and gaskets age and fail for the water valves. On the other hand, for steam valves, the main failure modes stay about the same over 19 years, with internal and external leaks contributing almost equally. This is mainly because packing failures start much earlier than those for the water valves, and continue to increase, causing the failure frequency of the steam valves to stay higher than that of the water valves. For common packing materials, high temperature accelerates aging degradation.

4.2 Stresses Induced by Testing

Since CI function is a safety related function to mitigate the consequences of design basis accidents, the components providing this function are tested periodically in accordance with the applicable ASME Code requirements. Typical testing includes leak rate tests for valves and stroke tests for valve operators. Section 8 has a more detailed discussion on testing and maintenance. The test frequency of each component depends on the plant's operating schedules, technical specification commitments, and maintenance and surveillance testing. Each time the valves are tested, the stresses imposed by exercising the components contribute to their aging-related degradation.

It is difficult to quantify what portion of aging degradation is caused by testing, but it is generally believed that excessive testing probably accelerates the process. However, it is also true that, as components age, more aging-related failures occur and more testing is needed to minimize the on-demand failures. There should be an optimum balance between the advantages and disadvantages of increased testing. Two of the major disadvantages of increased testing are the cost and the stresses induced. In making such decisions, the aging processes should be understood, and inspection methods and procedures should be accurate enough to detect changes in operating parameters due to aging.

4.3 Stresses Induced by Human Performance

Successful operation of the containment isolation components is a function of the initial design, manufacture, and installation of the components, and continues throughout the life of the plant. Even with rigorous training programs, humans make errors. Past operating experience verifies that errors in maintenance, installation, and procedures contribute to failures of system components.

The causes of maintenance errors, and components affected by them, are very diverse. Any component which requires maintenance at any time is subject to human errors. This category includes virtually all components in the plant; therefore, the potential here is quite large. It is only through rigorous training and the experience of personnel that maintenance errors are kept low. To address this, the ASME Boiler and Pressure Vessel Code requires components to be functionally tested following maintenance.

Typical maintenance errors include the use of wrong replacement parts, improper tightening of bolts or screws, and improper lubrication of moving components. LER 293/92-003 gave an example in which the inboard main-steam-supply isolation valve to the RCIC turbine indicated closed but was actually open. The valve could not be closed because the motor operator was detached from the yoke. The yoke's attaching capscrews had been insufficiently torqued during maintenance. The maintenance procedure incorrectly specified mild steel for the capscrew material, and the drawing of the valve gave the wrong size of capscrew. Other examples were seen where valves were degraded due to inadequate maintenance on packing. Most resulted in leakage of the coolant; however, some valves were inoperable due to over-tightening of the packing.

Some human errors accelerate aging and the resulting failures can be considered aging-related. However, because it is difficult to identify this type of aging failure, this study classified all the failures caused by human error as non-aging failures.

4.4 Environmental Effects

All the components that provide a containment isolation function are located inside the primary and between primary and secondary containment for BWRs, and inside and outside of containment for PWRs. The components in the primary containment are exposed to the most severe temperature, humidity, and radiation environments. Typical values are shown in Table 4.1 and 4.2, for BWRs and PWRs, respectively.

Table 4.1 Typical Environmental Stresses on CI Function Components in the BWR Containment

Parameter	Normal	Accident
Temperature	135-150°F	200-340°F
Pressure	16.7 psig (nitrogen)	62 psig
Relative Humidity	30-100%	100% (all steam)
Radiation	30 rads/hr	26 Mrads During LOCA 1.3×10^{6}rads/hr

Table 4.2 Typical Environmental Stresses on CI Function Components Inside the PWR Containment

Parameter	Normal	Accident
Temperature	50-120°F	300°F
Pressure	atmospheric	70 psig, max.
Relative Humidity	30-100%	100%
Radiation	50 rads/hr	150 Mrads

Since these components must function during and after design basis accidents, in some cases under extreme environmental conditions. All components which provide for containment isolation should be environmentally qualified (EQ). These EQ processes for valve operators, switches, transmitters and penetrations are very important because many sub-components of these components are made of elastomeric materials, which are known to degrade under high temperature, high radiation, and possibly high humidity. However, the EQ issue, especially under LOCA conditions, is out of scope of this study, and we consider only the aging processes during the normal operations.

4.5 Summary of Stresses

Since the containment isolation components are required for safe plant operation, and are tested in accordance with the applicable code requirements due to their safety significance, the components fulfilling this function experience stresses from operations and testing. Also, since a significant portion of the components are located inside the primary containment, the environmental stresses on these components are higher compared to those located outside containment. Table 4.3 summarizes some potential aging mechanisms that are significant to components for CI function.

Table 4.3 Aging Causes and Effects for Different Stress Sources

Stress Sources	Aging Cause/Effects	Mechanical (e.g., Valves, Valve Operators)	Inst. & Control (e.g., Switches, Detectors)
Normal Operating Conditions	Erosion, wear, corrosion, leakage	X	X
	Clogging or blocking.	X	
	Vibrations, misalignments, crack growth, loose or dislodged pieces	X	X
	Mechanical binding, distortion, rupture	X	
	Set point drift, out of calibration, loose connections		X
	Electrical shorts, grounds surface pitting, erratic signals/indicators	X	X
Testing	Wear	X	
	Vibrations	X	X
	Electrical Stress		X

5. PWR OPERATING EXPERIENCE

5.1 Analysis of NPRDS Data

As discussed in Section 2, containment isolation is a function that involves components from many plant systems; this resulted in a rather extensive NPRDS search. To review the operating data, systems were identified for both BWRs and PWRs primarily by considering the containment penetration tables in Chapter 6 of the various Final Safety Analysis Reports (FSARs). NPRDS includes containment isolation valves under specific NPRDS systems such as the Main Steam, Component Cooling Water, Low Pressure Safety Injection, or Nuclear Service Water. In addition, Westinghouse plants have a system called "Containment Isolation System", which includes all the CIVs that do not belong to other systems. The search showed that this system is the main source of data for component failures in the CI function.

As discussed in Section 1, due to the large amount of data to be reviewed, only the operating data from Westinghouse plants were analyzed in this study. An extensive number of NPRDS Westinghouse PWR containment isolation function failures were identified for the five years of interest (1989 through 1993). To reduce the data analysis, the risk insights of Section 3 were applied, and the search was revised to eliminate the following containment penetrations.

- those that form closed loops outside the containment with a pressure capability in excess of the containment ultimate pressure.
- those that were less than, or equal to four inches in diameter.

However, specific penetrations that may contribute to interfacing systems LOCA sequences or LOCA outside containment sequences were retained, based on the insights of the Reactor Safety Study, the Seabrook PRA and two generic PRA studies [Ref. 32, 33, 34].

Based on these system selection criteria, the NPRDS search was conducted on the following Westinghouse systems:

Containment Isolation System (primarily consisting of miscellaneous CIVs)
Component Cooling Water
Nuclear Service Water
Low Pressure Safety Injection (LPSI)/Residual Heat Removal (RHR)
Containment Spray
Ice Condenser
Reactor Building Penetration System (primarily consisting of personnel access hatches)
Combustible Gas Control

The main steam isolation valves also perform containment isolation functions, but they were excluded from this study because they are being studied by the Oak Ridge National Laboratory under the NPAR program.

For these systems, the NPRDS database identified 2,042 failure records, of which about 76 % were aging-related failures. Table 5.1 shows the contribution of different systems to the reported component failures in the containment isolation function; about 73% of the component failures are reported to the

Table 5.1 Failures in the Containment Isolation Functions

Contributing Systems	Total Failures	Aging-Related	Non-Aging	Aging Percent (%)
Containment Isolation	1,497	1,196	299	80
Component Cooling Water	151	106	45	70
Nuclear Service Water	131	76	55	58
LPSI/RHR	85	59	26	69
Containment Spray	128	84	44	66
Ice Condenser	18	12	6	67
Containment Fan Cooling	8	5	3	63
Reactor Building Penetration	11	6	5	55
Combustible Gas Control	35	20	15	57
Total	2,042	1,544	498	76

Table 5.2 Failures Reported to Containment Isolation System

Components	Total Failure	Aging-Related	Non-Aging	Aging Percent (%)
Valve	890	717	173	81
Valve Operator	480	379	101	79
Transmitter/Rad Monitor	36	29	7	81
Penetration	12	10	2	83
Circuit Breaker	32	20	12	63
Others	25	21	4	84

containment isolation system. The breakdown of these failures reported to containment isolation system is shown in the Table 5.2; most of the failures are caused by those of valves (60%) and valve operators (33%).

Failures of instrumentation and controls related to the containment isolation function were not reviewed because they are reported to engineered safety features actuation system (ESFAS), which was a subject of an earlier NPAR study [Ref. 10].

As shown in Figure 5.1, about 93% of the aging-related failures reported to the containment isolation system were due to failures of valves (61% of the aging-related failures) or valve operators (32%). For other systems listed in Table 5.1, only the aging-related failures of CIVs and valve operators were searched and reviewed. Thus, in this section, the aging of valves and valve operators is mainly studied by combining all the failure data from all systems shown in this Table. Failures of the personnel hatches were not studied because there were only 16 reported.

Quantification of age-dependant failure rates, as would be necessary for this aging data to be used in PRAs [Ref. 35], was considered to be beyond the scope of this study. The trends and other insights presented were based upon information presented by the plant personnel to the NPRDS and LER databases.

5.1.1 Valves

The total number of aging-related valve failures studied was 965, of which 717 were reported to the containment isolation system. The failures of containment isolation valves reported to other systems are shown in Table 5.3. To analyze aging effects on valve failures, all the failure data listed in Table 5.3 were combined.

5.1.1.1 Effects of CI function valve failures on plant operation

During the 5 years studied, there were 6 valve failures in the CI function that resulted in a delay in start-up (unit off-line). Five involved leaks (4 internal leaks and 1 external leak), and the last was a failure to close. All were detected during the surveillance testing, in-service inspection, or routine observation, and the plants were either at low power or 0% power. The leaks were caused by aged seals or worn seat/plug, and the failure to close was caused by a galled and pitted valve stem, plug assemblies, cage, spacer and bonnet gasket.

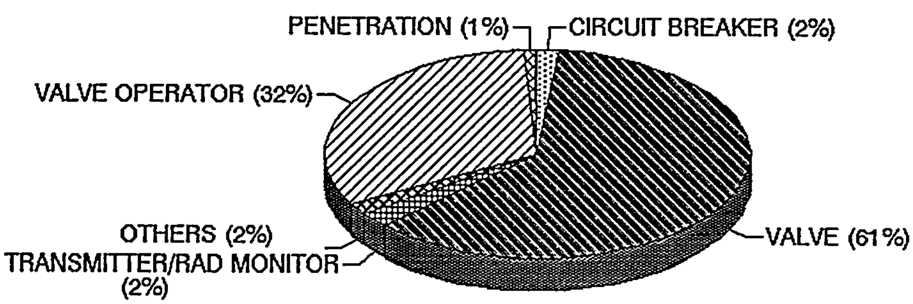


Figure 5.1 Components that failed due to aging in Westinghouse containment isolation system. Source: NPRDS (1989 - 1993), Total number: 1149

Table 5.3 Aging-Related Containment Isolation Valve Failures Reported to Different Systems

Systems	Number of Aging-Related Failures
Containment Isolation System	717
Component Cooling Water System	64
Containment Spray System	59
Nuclear Service Water System	51
LPSI/RHR	35
Combustible Gas Control System	12
Containment Fan Cooling System	10
Ice Condenser System	5
High Pressure Safety Injection System	4
Reactor Building Penetration System	3

One example of the internal leak failure is that of a butterfly valve, which was detected during the Local Leak Rate Test (LLRT) of the containment's purge air return valve. The valve was leaking past its seat, and aging of the butterfly valve spring caused the loss of its shutting power and resultant internal leak. The start up was delayed until the repair was complete, which was reported as a unit off-line.

During the same period studied (1989-93), four incidents of CI function valve failures resulted in reduced power operation. Three occurred while the plants were at power, and one occurred during a startup power ascension. In the first case, the valve leak was found, and in the next two cases, the valves failed to close on demand. In the last case, the valve failed to open on demand. Even though these demands were not CI function demands, but demands for separate systems (e.g., pressurized liquid sample line, steam generator water sample line), the CI function would have been degraded if it had been required.

In the first incident, two steam-generator blowdown containment isolation valves (at the same plant) were found to be leaking through due to normal wear. Power was reduced briefly during a containment entry to manually isolate the blowdown lines.

The cause for the second failure was a combination of selecting an improper valve design for this application and a dirty system. The cause for the third one was unknown, and the cause for the last one was mechanical binding due to aging.

All ten CI valve failures did not affect the safety of the plant due to the redundant CIV design, though they affected the cost of operation. The rest of the valve failures discussed in this section did not affect the plant operation nor the safety of the public.

5.1.1.2 Valve types

The majority of the aging-related failures are those of four major types of valves; globe valves (41%), check valves (18%), gate valves (15%), and butterfly valves (15%); other types are diaphragm valves (5%), needle valves (2%), ball valves (2%), and plug valves (2%) (Figure 5.2). Since the aging effects on valve failures may depend on the type of the valve, the four major types of the valves are studied separately in this section.

5.1.1.3 Effects of aging on failure frequency

Figure 5.3 shows the normalized aging-related failure frequency as a function of age at failure for all the different types of valves combined. This plot clearly shows that the failure frequency decreases slowly until 11 years, after which it increases rather rapidly. Even though this plot is normalized for the number of plants reporting the failures, the data for the older ages should be used carefully. Since the number of plants reporting these failures at older ages is much smaller (e.g., only 5 plants have valves that are older than 23 years, compared to 48 plants that could have reported failures of 4-year old valves), the failure frequencies at the far right side of the plot may be exaggerated.

When the normalized failure frequency is plotted for each different type of valve, it can be seen which type is affected more by aging processes (Figures 5.4 - 5.7). The shape of the failure frequency curve for globe valves are very similar to that for the combined valves, which indicates that the combined curve is mainly influenced by that of the globe valves; this is expected since 41% of all the aging-related failures are from globe valves. The gate valves also showed a similar trend as that for globe valves. On the other hand, check valves and butterfly valves showed different trends, with the failure frequency

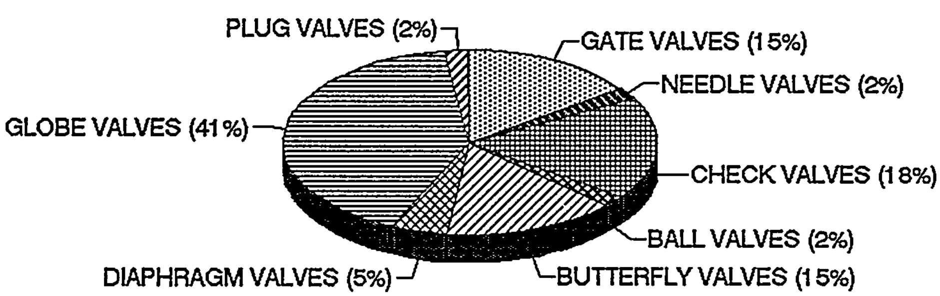


Figure 5.2 Different types of valves that experienced aging-related failures

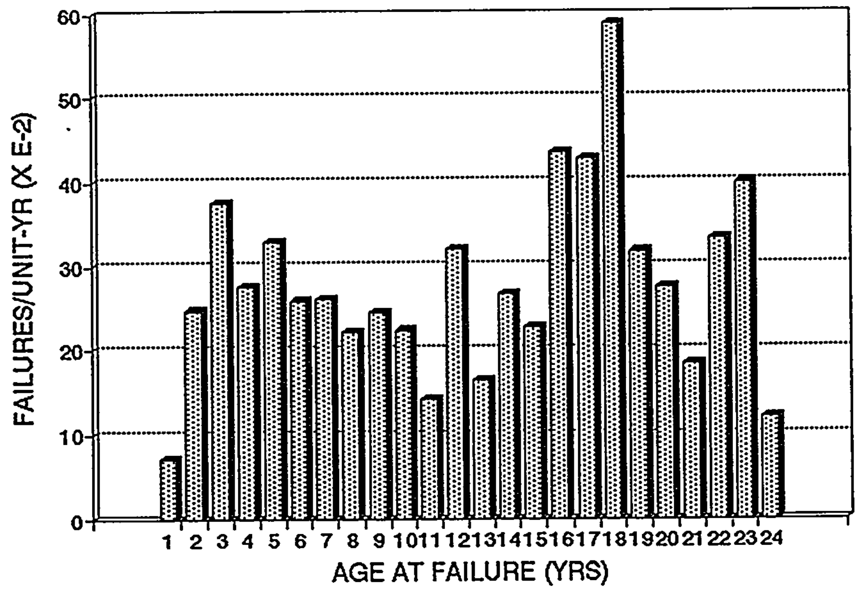


Figure 5.3 Normalized failure frequency for containment isolation valves in Westinghouse PWRs

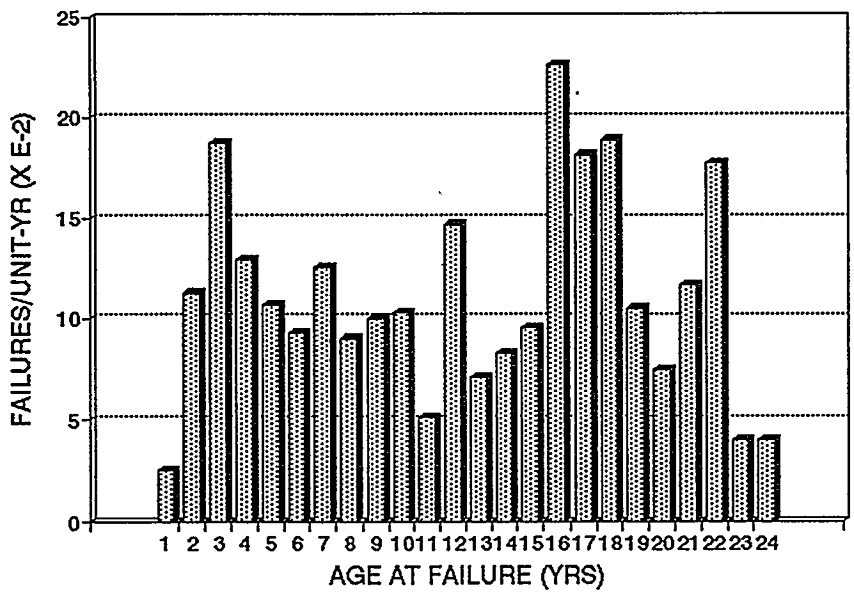


Figure 5.4 Normalized failure frequency for globe valves in the Westinghouse CI function

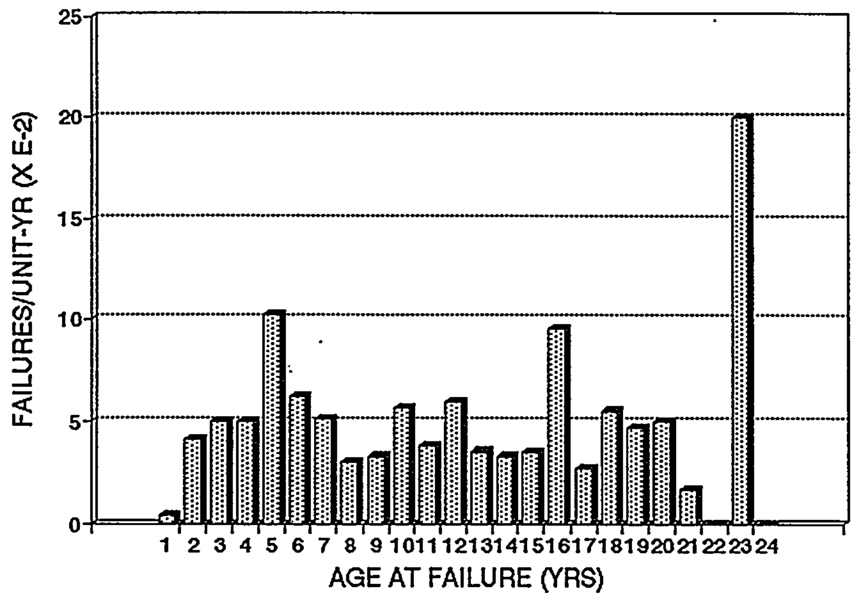


Figure 5.5 Normalized failure frequency for check valves in the Westinghouse CI function

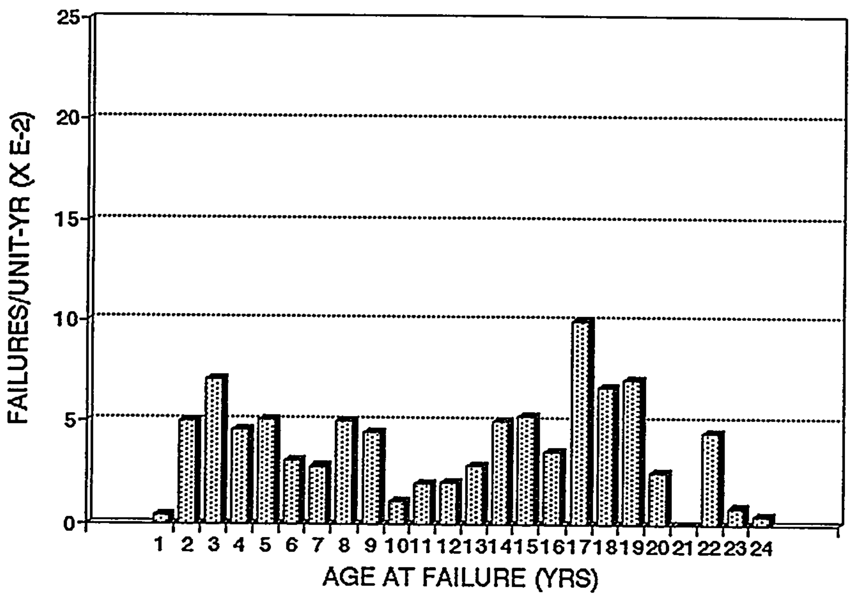


Figure 5.6 Normalized failure frequency for gate valves in the Westinghouse CI function

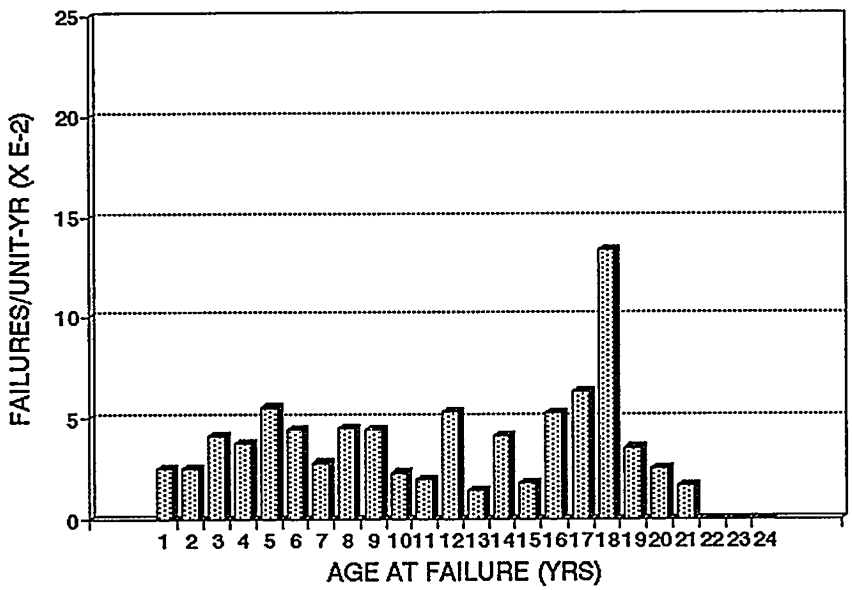


Figure 5.7 Normalized failure frequency for butterfly valves in the Westinghouse CI function

remaining almost constant except a peak at age 18 for butterfly valves and at 23 years for check valves. In the following sections, these effects of aging on failure frequencies for different types of valves will be explained using information on the modes and causes of failures.

5.1.1.4 Globe valves

There were 395 aging-related globe valve failures in the containment isolation function, which accounts for 41% of all the aging-related valve failures in the system. Approximately 54% of these globe valves were operated by pneumatic type operators, followed by solenoid operators (22%), manual operators (10%), and electric motor operators (9%).

Failure Modes

As shown in Figure 5.8, 71% of the valve failures are leaks, "internal leak" (46%) and "external leak" (25%). Other failure modes are "fail to close" (13%), "fail to open" (6%), and "fail to operate as required" (5%). To understand the effects of aging on these failure modes, the major ones were analyzed for two periods, 1 - 11 years and 12 - 24 years, based on the shape of the failure frequency curve in Figure 5.4. Figure 5.9 shows that the failure modes "internal leak" and "external leak" increased during the later years, compared to the earlier period. This will be discussed later using the information on the failure causes. This approach, dividing the failures into two groups, provides more insights to the aging process than analyzing all the data together, and in the rest of this report, it will be used whenever it is more effective.

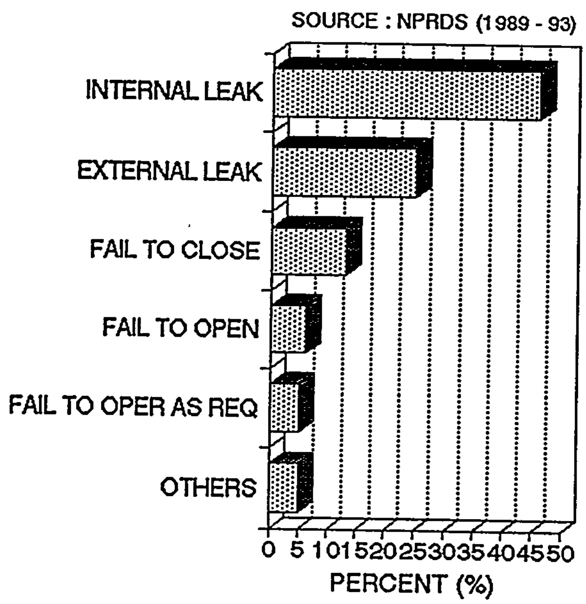


Figure 5.8 Failure modes for globe valves in the Westinghouse PWR CI function

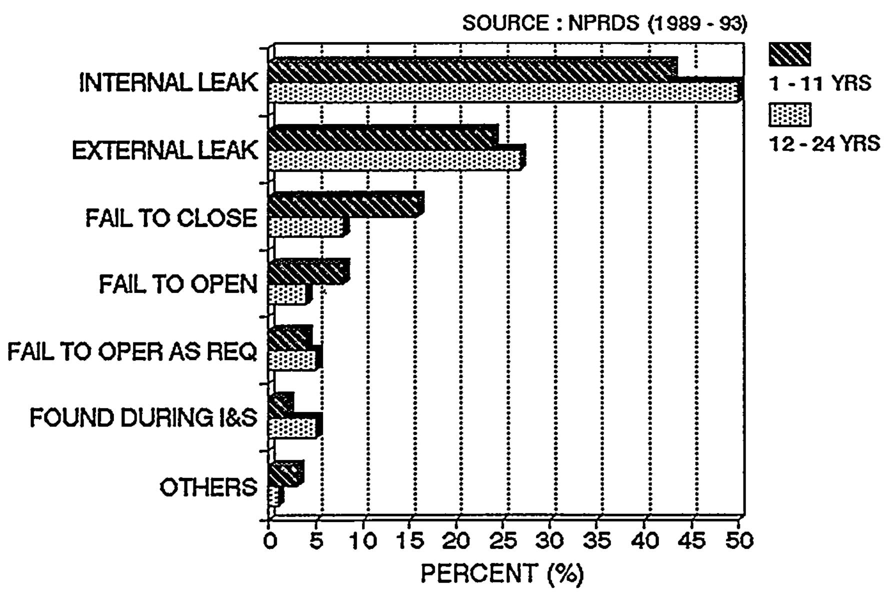


Figure 5.9 Failure modes for globe valves in Westinghouse PWR CI function during two periods

Failure Causes

Figure 5.10 shows the causes of failure reported to NPRDS for the globe valves. The predominant one is "normal/abnormal wear", causing about 61% of the failures. Other causes are "mechanical damage/binding" (9%), "dirty" (8%), "aging/cyclic fatigue" (6%), "out of mechanical adjustment" (4%), and "corrosion" (4%).

The proximate cause analysis provides more detailed information on which sub-component caused the failure. Three proximate causes, "worn or damaged seat/disc" (37%), "worn packing" (19%), "corrosion product/dirt buildup" (18%) are responsible for about 68% of the globe valve failures (Figure 5.11). Other important proximate causes are "broken/damaged stem" (6%) and "worn/damaged gasket" (6%). "Others" in this figure includes "out of adjustment packing", "failed locknut/bolt", and "failed valve fitting."

Figure 5.11 shows that the failures caused by "worn packing" and "worn gasket" increased during the later years, which explains the increase of the failure mode "external leak" shown in Figure 5.9. Worn or out-of-adjustment packings and worn or damaged gaskets result in external leaks. The failures due to "aged seat/disc" also increased as the valves got older, which may explain the increase of the failure mode "internal leak". "Aged seat/disc" includes worn or damaged seats, worn or damaged discs, and corroded seats/discs, all of which resulted in internal leaks. The NPRDS Manual [Ref. 36] states that leakage of the working fluid past the valve seat when fully closed should be reported as "internal leakage", and leakage through the valve because the valve has not closed completely is classified as "failure to close". It also states that the examples of "failure to close" include the failure to close due to corrosion, internal binding, missing or damaged parts, travel stop nuts being out of adjustment, and excessive dirt buildup that prevents the valve from closing completely. Figure 5.9 indicates that the percentage of the "fail to close" decreased during the later years, explicable by the reduced cases of corrosion product buildup during later years, as shown in Figure 5.11.

To obtain more detailed information on the effects of aging on proximate causes, the number of failures caused by the three major proximate causes were normalized and plotted as a function of age at failure (Figure 5.12). The frequency curve for the failures caused by worn seats/discs has two peaks at 12 years and 21 years, which explains the peaks at the same ages in the plot for globe valve failures (Figure 5.4). The failures caused by corrosion product/dirt buildup has a high peak at 3 years, after which the frequency stayed relatively constant. The failures caused by worn packing also peaks at 3 years, after which the failure frequency stayed low until it began to increase at 13 years. At 16 to 18 years, many valve failures were caused by worn packing, which explains the peak in the frequency of globe valve failure at 16 - 18 years (Figure 5.4).

5.1.1.5 Check valves

There were 171 aging-related failures of check valves, which accounted for 18% of all the aging-related valve failures in the CI function. Approximately 54% of the check valves had no valve operators. About 36% of the failed check valves had mechanical operators, which open the valve upon the set differential pressure and close by spring force (i.e., excess flow check valves), and about 7% of the check valves had manual operators (i.e., testable check valves).

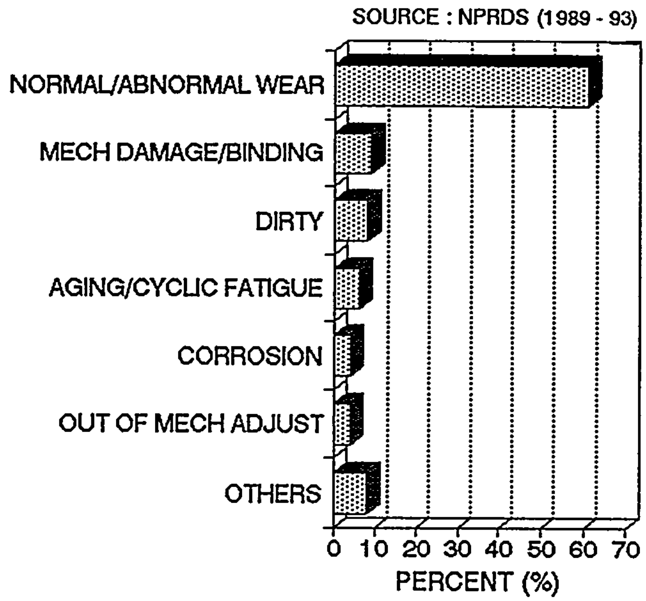


Figure 5.10 Failure causes for globe valves in the Westinghouse PWR CI function

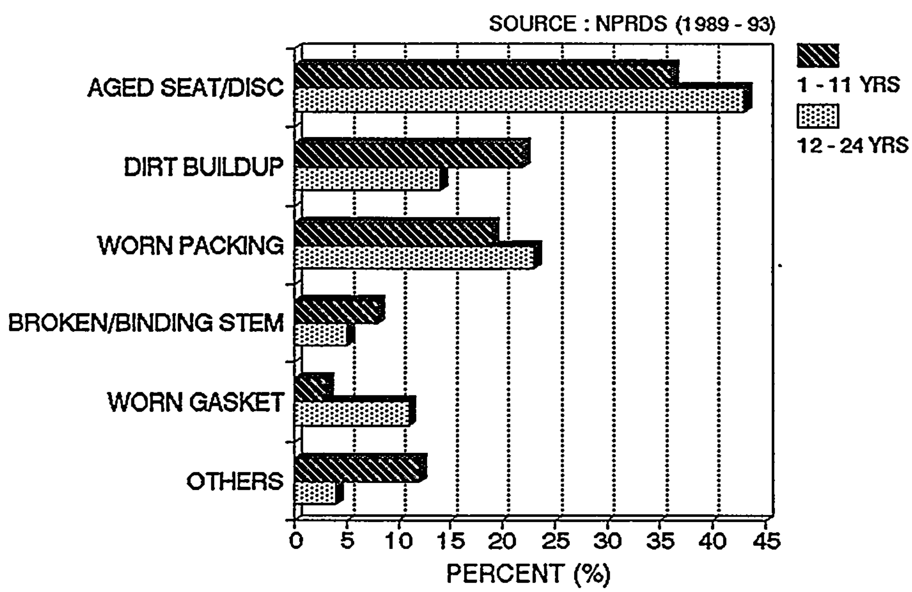


Figure 5.11 Proximate causes for globe valves failures in Westinghouse PWR CI function

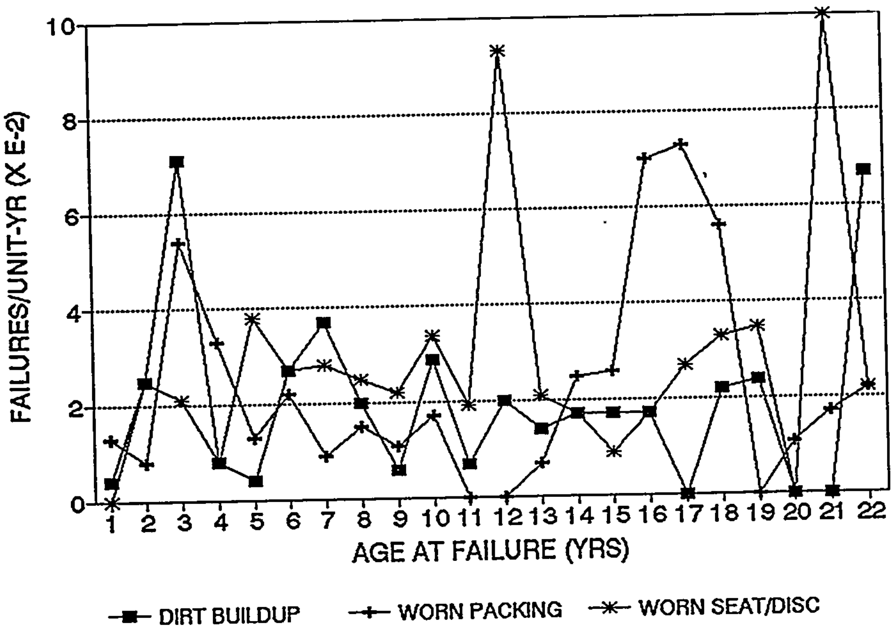


Figure 5.12 Frequency of globe valve failures caused by major proximate causes

Failure Modes

Internal leak was the main failure mode for the check valves accounting about 69% of the failures. "Fail to close" is the next major mode, accounting for about 21% of the failures. Other minor failure modes are "external leak" (4%) and "found during inspection and surveillance" (I&S) (4%).

As Figure 5.13 shows, the failure modes for check valves were not significantly affected by aging. The differences between failure modes during the early and later period are minor.

Failure Causes

The major causes of failure for the check valves are "normal/abnormal wear" (51%) and "dirty" (23%) (Figure 5.14). "Corrosion" (6%), "mechanical damage/binding" (6%), "aging/cyclic fatigue" (5%), and "particulate contamination" (4%) are responsible for the remainder.

The major proximate causes are "corrosion product buildup" (42%) and "aged seats/discs" (34%), with minor ones "worn gaskets" (4%), "aged springs" (3%), "binding/worn hinge pins" (4%), and "lubrication problems" (1%), (Figure 5.15). "Others" in this figure includes "out of adjustment seats" and "worn bearings". This analysis of proximate causes indicates that buildup of corrosion products causes either "internal leak" or "fail to close" in the check valves. It also shows that aged seat/disc and corrosion product buildup are the main causes for failures over the years, except that failures caused by the latter increased a little in later years.

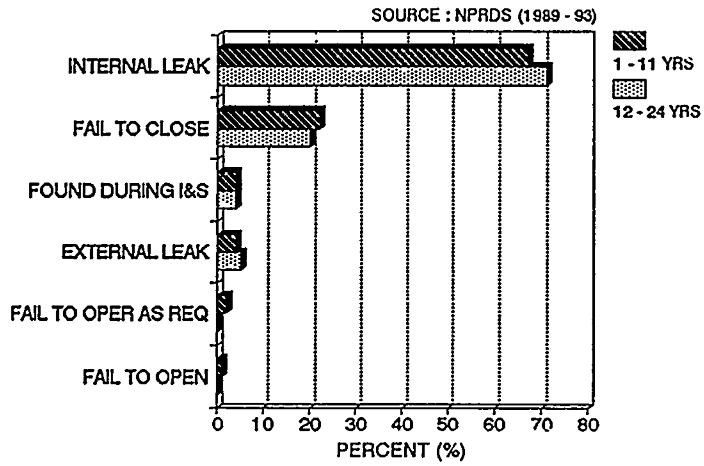


Figure 5.13 Failure modes for check valves in Westinghouse PWR CI function

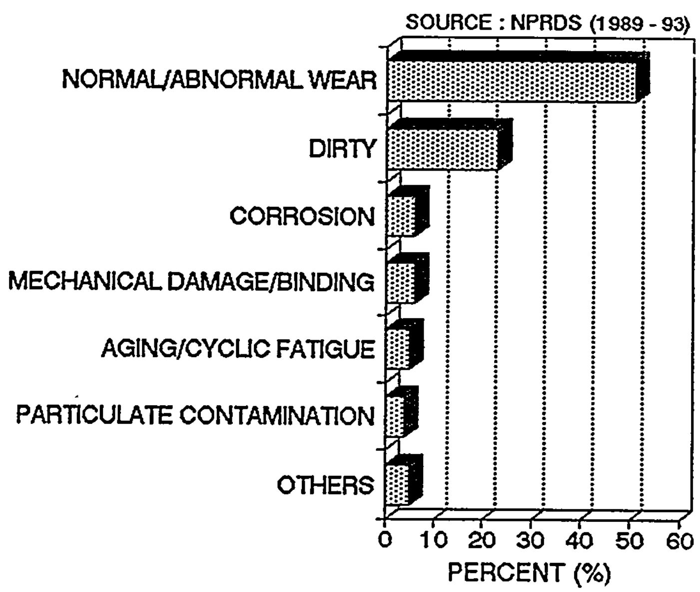


Figure 5.14 Failure causes for check valves in Westinghouse PWR CI function

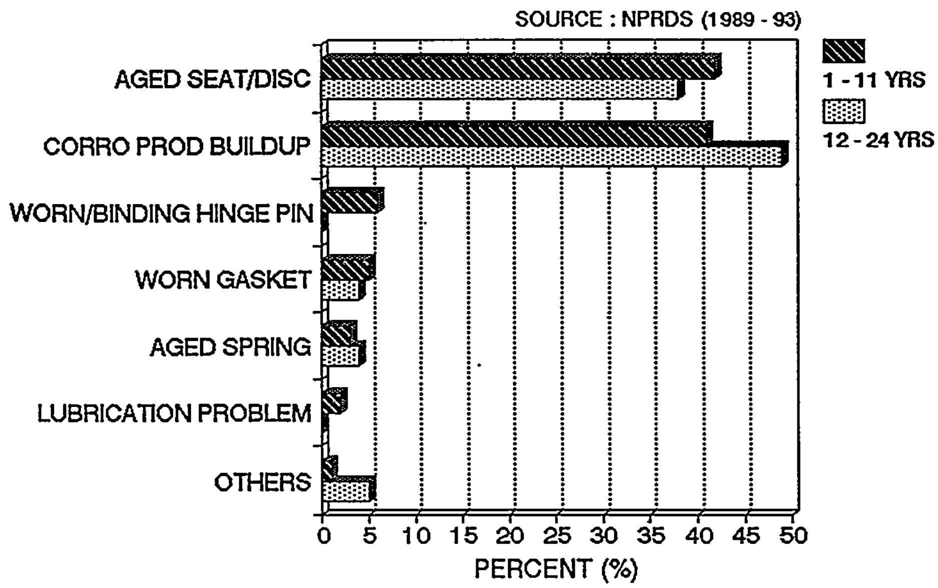


Figure 5.15 Proximate causes for check valves in Westinghouse PWR CI function

To understand the effects of aging on proximate causes, the numbers of failures of check valves caused by two major proximate causes, corrosion product/dirt buildup and worn seat/disc, were normalized and plotted as a function of age at failure (Figure 5.16). The frequencies of failures from the two proximate causes were not significantly affected by aging except that there may be 6 -7 year peak cycles for both. This figure also shows the reasons for the two peaks at 5 years and 16 years in Figure 5.5, which is the frequency curve for check valve failure.

5.1.1.6 Gate valves

There were 147 aging-related failures of gate valves in the CI functions during the 5 years accounting for about 15% of all the aging-related valve failures. Approximately 52% of the these gate valves were operated by electric motor operators, and 25% manually. About 15% of the valves had pneumatic operators, and the rest are operated by solenoid operators or mechanically.

Failure Modes

Most of the failures for gate valves resulted in leaks, "external leak" (39%) and "internal leak" (38%). The other modes are "fail to close" (11%), "found during I&S" (7%), "fail to open" (3%), and "fail to operate as required" (2%). Figure 5.17 shows that with age the percentage for internal leaks increased, while that for external leak decreased a little.

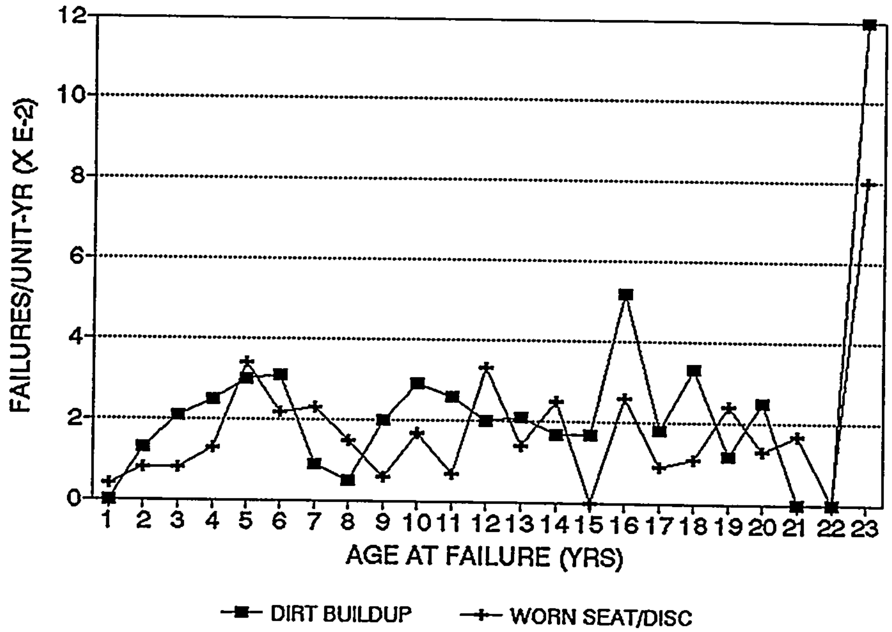


Figure 5.16 Frequency of globe valve failures caused by major proximate causes

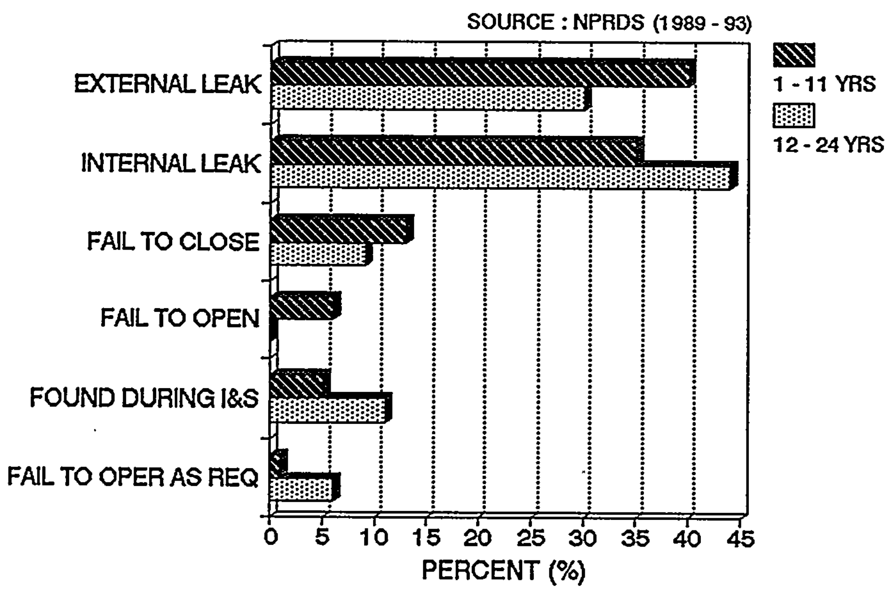


Figure 5.17 Failure modes for gate valves in Westinghouse PWR CI function

Failure Causes

About two thirds of the gate valve failures were caused by "normal/abnormal wear" (67%), followed by "dirty" (7%) and "mechanical damage/binding" (7%). Minor causes were "aging/cyclic fatigue" (5%), "corrosion" (4%), "lubrication problems" (3%), and "foreign material" (3%) (Figure 5.18). "Others" in this figure include "out of mechanical adjustment" and " particulate contamination".

The major proximate causes for the failures of gate valves were "aged seats/discs" (26%), "worn packing" (24%), "corrosion product/dirt build-up" (22%), and "worn gaskets" (12%) (Figure 5.19), while the minor ones are "broken/binding stems" (5%) and "lubrication failures" (2%). "Others" in this figure include "out of adjustment seats," "out of adjustment packing," and "locknut/bolt failure." As the gate valves age, worn/damaged seats/discs and buildup of corrosion products caused significantly more failures compared to earlier years.

The frequencies of failures caused by the three major proximate causes, aged seat/disc, corrosion product/dirt buildup, and worn packing, were plotted as a function of age at failure in Figure 5.20. The failures caused by worn seats/discs peaked at 3 years, after which they stayed low until 14 years, when they began to increase. The failures caused by worn packing show more visible effects of aging; they stay roughly the same for the first 9 years, after which they decrease to a very low number for about 4 years. Then, starting at 14 years, they began to increase rather fast. The same trend was shown by the failures of globe valves caused by worn packing (Figure 5.12). Again, Figure 5.20 explains the shape of the frequency curve for gate valve failures shown in Figure 5.6.

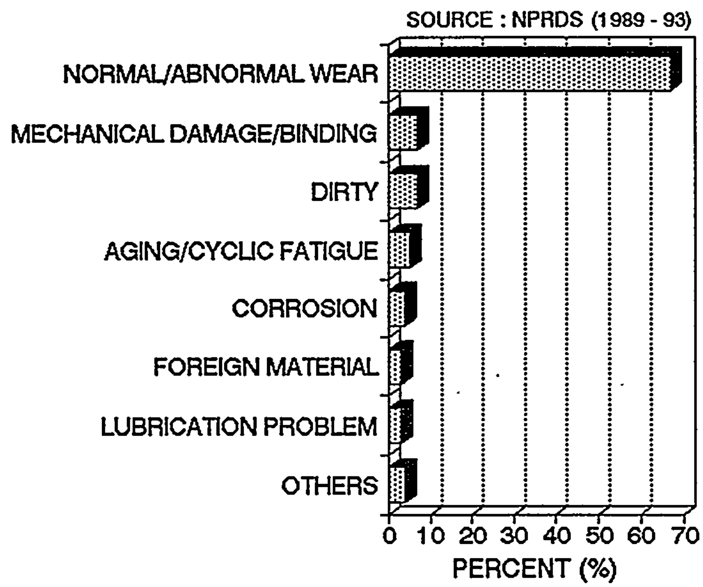


Figure 5.18 Failure causes for gate valves in Westinghouse PWR CI function

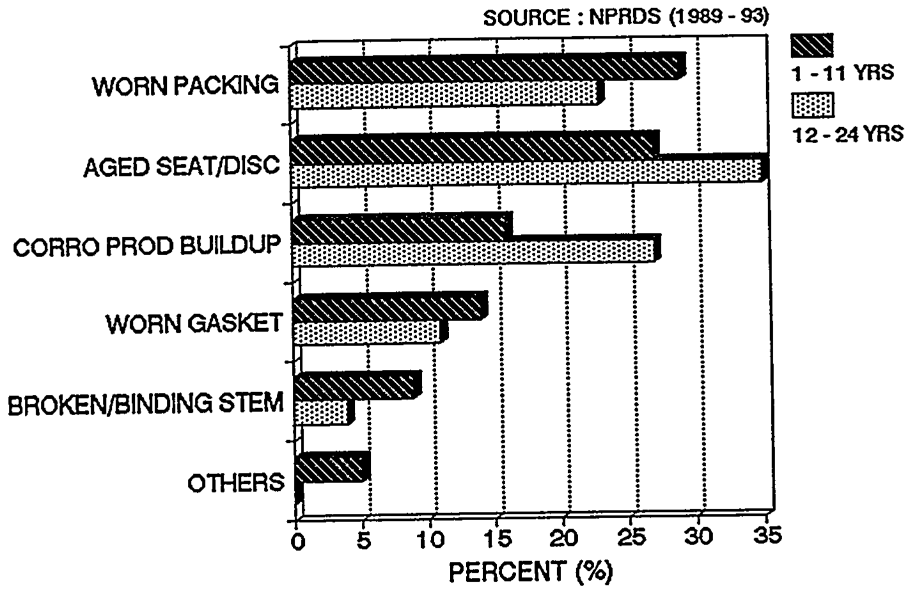


Figure 5.19 Proximate causes for gate valve failures in Westinghouse PWR CI function

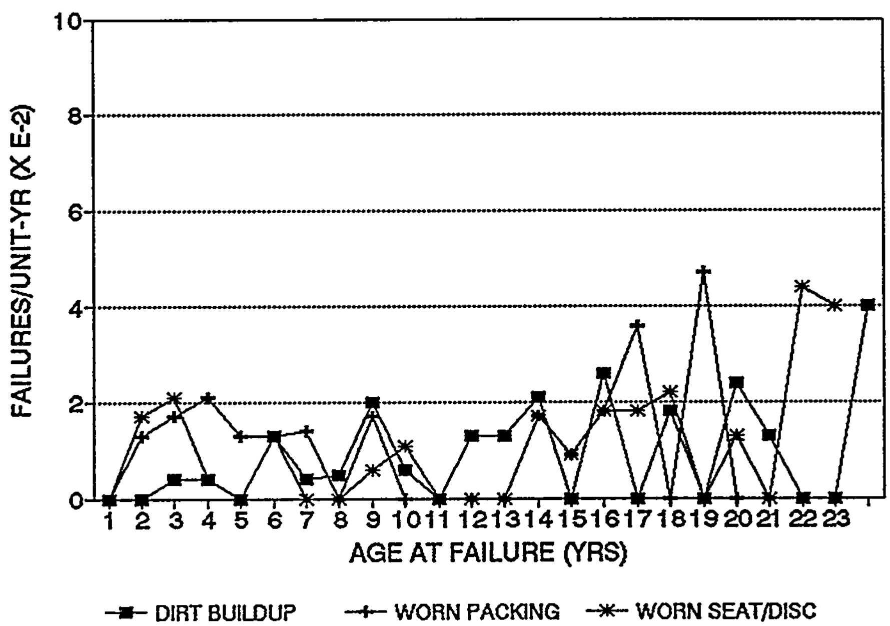


Figure 5.20 Frequency of gate valve failures caused by major proximate causes

5.1.1.7 Butterfly valves

There were 139 failures of butterfly valves in the CI functions, accounting for about 14% of the valve failures. About 46% of the butterfly valves had pneumatic operators; 30% of the valves were operated by electric operators, and 16% manually. The rest are operated by hydraulic operators.

Modes of Failure

The major failure mode for butterfly valves was "internal leak" (65%). The minor failure modes were "failure to close" (17%), "external leak" (11%), and "fail to operate as required" (4%). As the butterfly valves aged, the portions of "failure to close" and "external leak" increased, while that for "internal leak" decreased (Figure 5.21).

Causes of Failure

Approximately 54% of the failures were caused by "normal/abnormal wear", and about 17% of the failures were due to "dirty" internals. Other causes were "aging/cyclic fatigue" (9%), "out of mechanical adjustment" (6%), "corrosion" (4%), and "foreign material" (3%), (Figure 5.22). "Others" in this figure include "lubrication problems" and "loose parts."

The major proximate causes for the butterfly valve failures were " corroded/worn/damaged seats/discs" (44%) and "corrosion product buildup" (30%). This shows that most of the failures are caused by corrosion and aging of seat and disc. Also, it is noticeable that "out of adjustment stems" is a negligible cause of failure for butterfly valves (7% of the failures). Other minor proximate causes are "worn packing" (6%), "worn gasket" (4%), and "worn seal ring" (4%). As the butterfly valves got older, worn/damaged/corroded seat/disc caused more failures, which apparently resulted in more "fail to close" failures, as shown in Figure 5.21 and 5.23. This result is quite different from those for other types of valves; most of the increased proximate cause of "aged seat/disc" resulted in increased failure mode of "internal leak". Butterfly valves usually are equipped with a liner on their seat, typically made of an elastomer material, which provides good sealing when they are new. However, when the liner is aged and damaged, it tends to allow a larger leak compared to a metal seat. Also, a good seal for the whole circumference of the disc, which usually is the same as that of the piping, is needed for tight closure. In short, butterfly valves are prone to large leaks when they are old, which probably have been reported as failures to close, which may explain the increase of "fail to close" mode in the later years.

5.1.2 Valve operators

There were 668 reported failures of valve operators in the containment isolation function, which constitutes about 33% of all the component failures. Of these, approximately 73% were determined to be aging-related. In this section, the effects of aging on the frequency, modes, and causes of failures of the valve operators are discussed. The effects of these failures on the plant operation also are discussed.

Only two major types of valve operators AC electric motor operator (22%) and pneumatic operator (74%), contributed for approximately 96% of all the aging-related failures. Thus, only the data from these two types were analyzed.

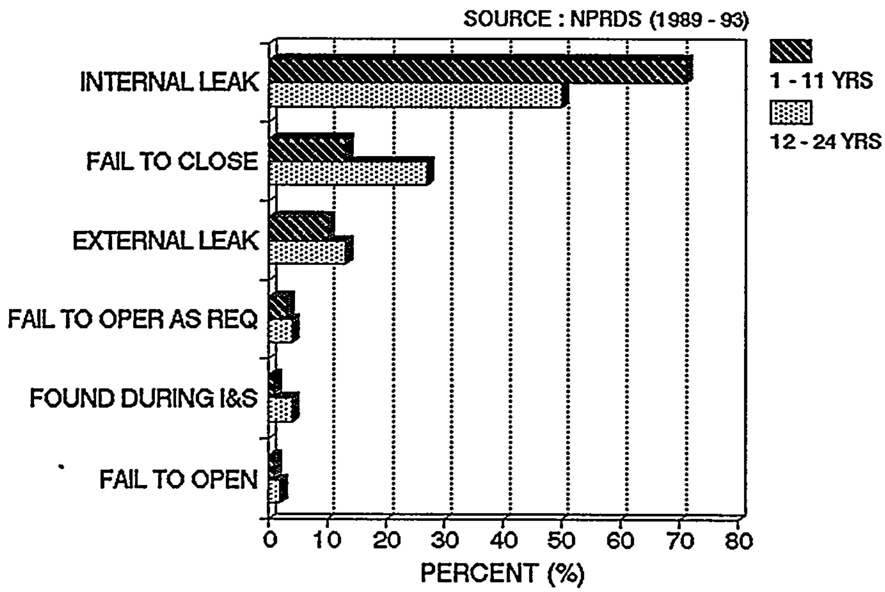


Figure 5.21 Failure modes for butterfly valves in Westinghouse PWR CI function

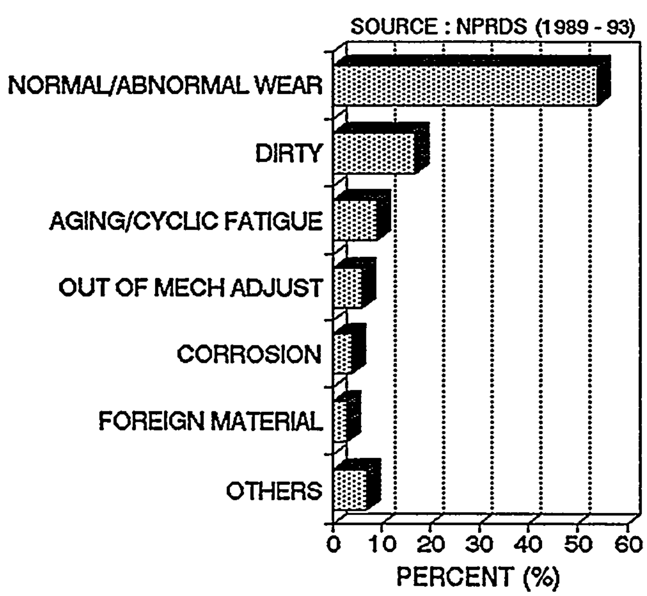


Figure 5.22 Failure causes for butterfly valves in Westinghouse PWR CI function

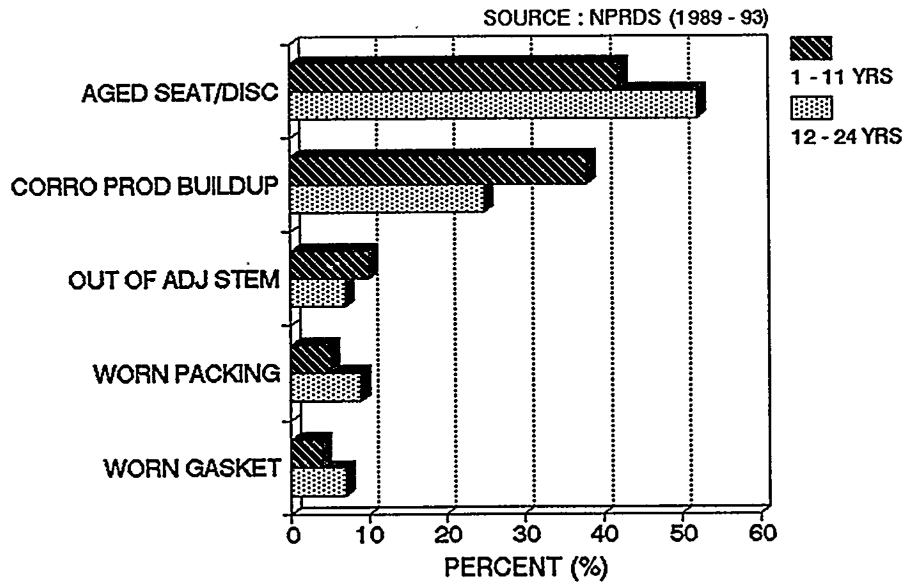


Figure 5.23 Proximate causes for butterfly valve failures in Westinghouse PWR CI function

5.1.2.1 Effects of valve operator failures on plant operation

During the period studied, one valve operator failure resulted in a unit off-line, and three resulted in reduced power operation. All of these valve operators belonged to the containment isolation system.

A burned-out motor of a steam-generator-blowdown containment isolation valve failed to open during a scheduled surveillance test, which caused a loss of the only flow path for steam generator blowdown. As required by plant Technical Specifications, the failure resulted in the unit being taken off-line while the valve was repaired. Breakdown of the insulation may have been the root cause.

Two failures resulted in reduced power operation. However, they occurred during the start-up process, and the effects were the same as those discussed above as failures that resulted in unit off-line. While heating up after a refuelling outage, and during the stroke-timing test of the containment isolation valves, the 36 inch containment purge valve failed to meet the stroke closure time of 2 seconds, and plant heat-up was limited to 200 degrees. This incident was reported as resulting in reduced power operation. Its cause was set point drift of the valve operator limit switch.

The second case also occurred with the unit in a refueling outage. A steam generator blowdown isolation valve would not open upon demand. Valve operation is needed for ascension to mode four, and the start up was delayed. The cause of the failure was due to degradation of the instrument-air regulating valve of the valve operator.

One incident was reported as "resulted in damage to other equipment." During a scheduled surveillance functional test of the nuclear sampling system pressurizer sample outside containment isolation valve, the motor stopped operating. The cause of the failure was a short to ground of the motor windings, which, in turn, made the valve operator over-torque and damage the valve.

Even though a few failures of CI valve operators resulted in either unit off-line or reduced power operation, none affected the safety of personnel or the public.

5.1.2.2 Effects of aging on failure frequency

When the normalized failure frequency (expressed as number of failures per unit-year), is plotted as a function of the age at failure, there is a continuous increase in the frequency (Figure 5.24). Evaluation of similar plots for AC electric and pneumatic valve operators, indicates that the failure frequency increase is mainly a result of aging of the pneumatic operators. Figure 5.25 shows that the failure frequency for the pneumatic operators increases dramatically after about 11 years, to its highest value at 18 years, when it is about 5 times higher than at 1 - 11 years.

On the other hand, AC electric motor operators did not show any significant aging effects (Figure 5.26). Aging effects on failure frequency are related to the causes of failure and the corrective action taken in response to earlier failures. For instance, if the major corrective method is replacing failed parts, it is expected that more failures will occur as the components get older because they usually are composed of several different parts and assemblies. However, if the whole component is replaced every time a failure occurs, then the failure frequency is expected to stay about the same. The effects of aging on failure frequency will be discussed later, using the results of failure data analysis.

5.1.2.3 AC electric motor operator

Failure Mode

For the AC electric valve operators, "failure to close" is the leading mode (35%), which is followed by "found during surveillance" (26%), "failure to open" (22%), and "failure to operate as required" (17%), (Figure 5.27).

Failure Causes

The NPRDS data analysis for the AC electric valve operators shows that the leading failure cause is "normal/abnormal wear" (28%), followed by "out of mechanical adjustment" (26%), "mechanical damage/binding" (8%), and "aging/cyclic fatigue" (8%). The electrical causes are "open circuit" (6%), "defective connection/loose parts" (4%), "short/grounded" (4%), "defective circuits" (3%), "out of calibration" (3%) and "burned/burned out" (2%) (Figure 5.28). The "other causes" include "dirty", "blocked/obstructed" and "abnormal stress". The results of the proximate cause analysis show in more detail which sub-components were responsible for the components' failures. The major proximate causes for the AC electric valve operators are the failures of torque switches (24%), limit switches (23%), and motors (7%). Problems with the valve stem such as out of adjustment and binding caused about 9% of the failures, and spring problems about 5%. Other minor proximate causes are "failed latching assembly", "worn shaft gear", "seal leak", "bad contacts", and "damaged gasket" (Figure 5.29).

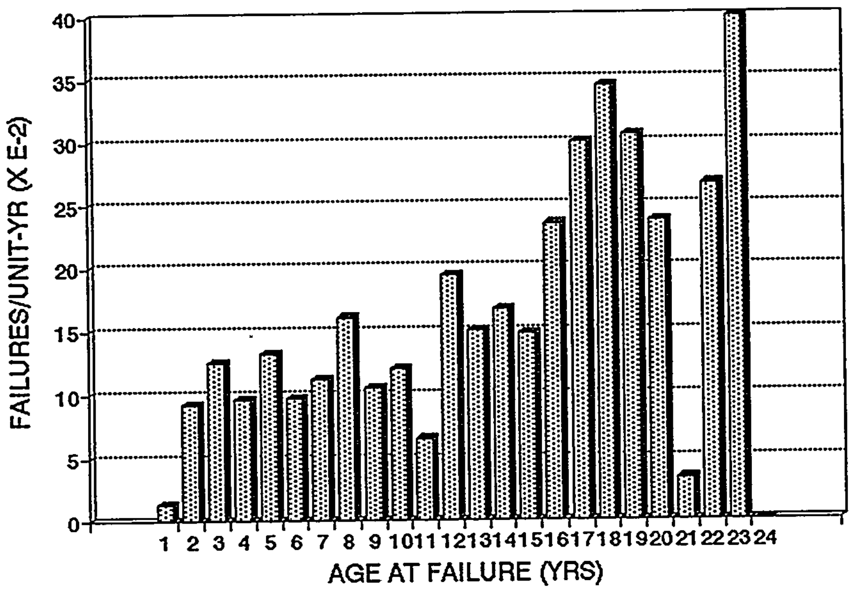


Figure 5.24 Normalized combined failure frequency for valve operators in Westinghouse CI functions

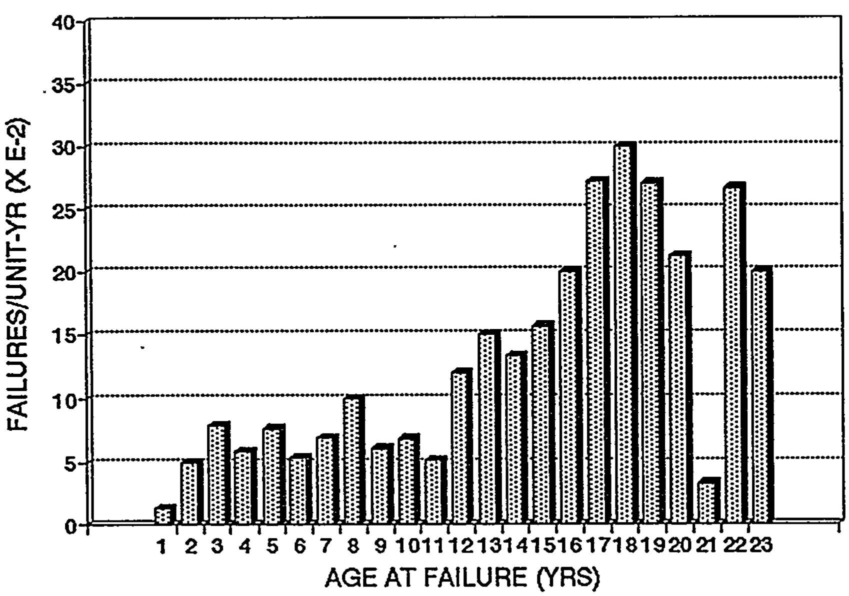


Figure 5.25 Normalized failure frequency for pneumatic valve operators in Westinghouse PWR CI functions

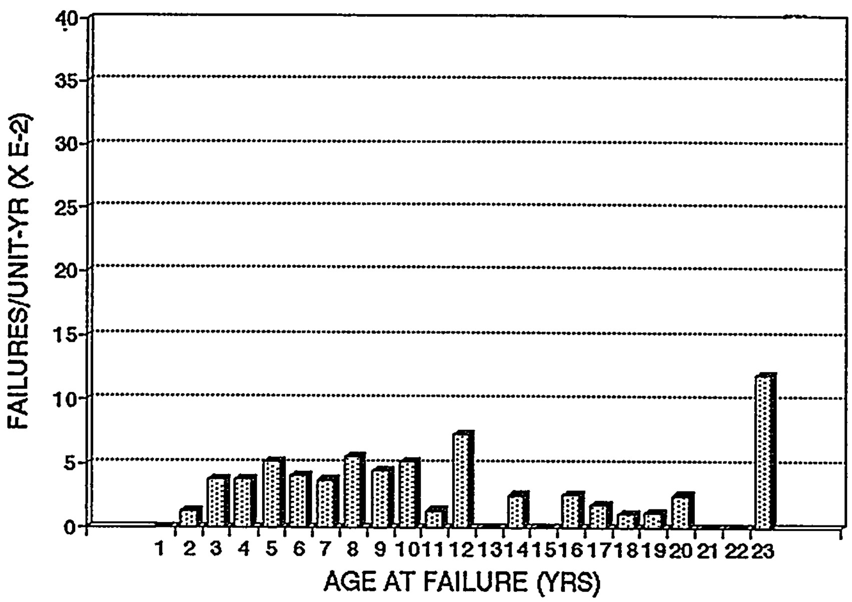


Figure 5.26 Normalized failure frequency for AC electric valve operators in Westinghouse PWR CI functions

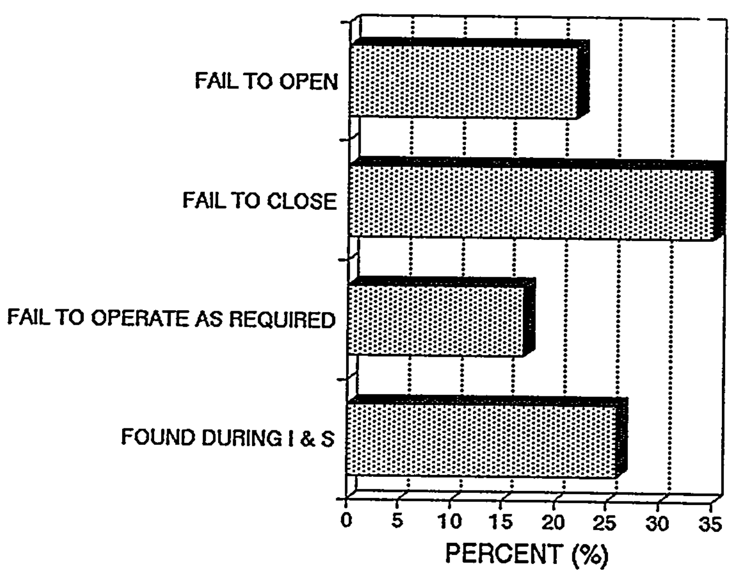


Figure 5.27 Failure modes for AC electric valve operators

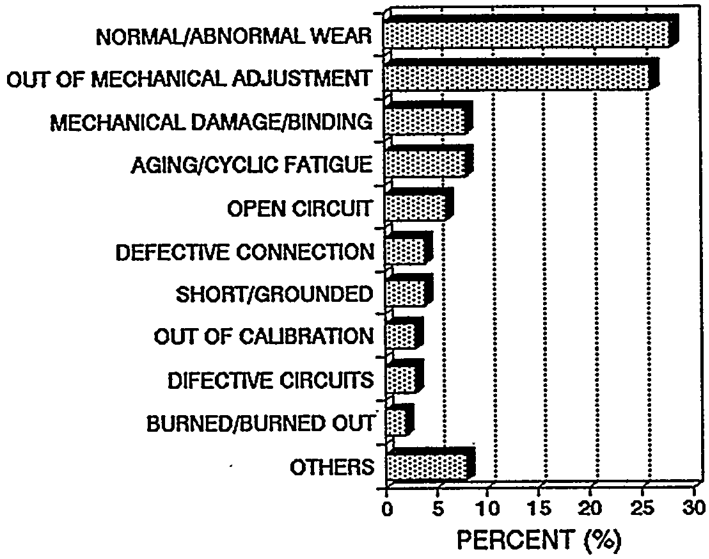


Figure 5.28 Failure causes for AC electric valve operators

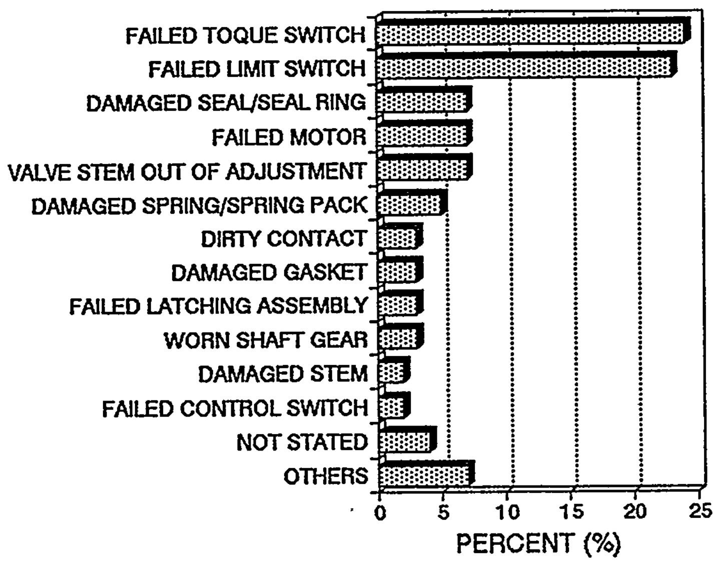


Figure 5.29 Proximate failure causes for AC electric valve operators

5.1.2.4 Pneumatic operators

Modes of Failure

For pneumatic valve operators, "failure to open" (35%) is the leading mode of failure, followed by "failure to close" (26%), "failure to operate as required" (25%), and "found during surveillance" (13%) (Figure 5.30). For the containment isolation function, "failure to close" is the only important mode, which makes the number of component failures that directly affect the CI function much less than those analyzed in this report. However, since aging characterization was the main concern and the valves may also have a safety function to open, all the failures were analyzed.

Causes of Failure

The leading cause of failure for pneumatic valve operators is "normal/abnormal wear" (33%), followed by "out of mechanical adjustment" (19%), "aging/cyclic fatigue" (16%), and "mechanical damage/binding" (6%). Other, less significant, causes are "connection defective/loose parts" (5%), "dirty" (4%), "open circuit" (3%), "out of calibration" (2%), and "defective circuit" (2%) (Figure 5.31). "Others" in this figure include "burned/burned out", "blocked/obstructed", "short/grounded", and "insulation breakdown."

Proximate cause analysis shows that the failures of diaphragm (18%), solenoid valve (16%), limit switch (15%), air regulator/line (13%) caused the majority of pneumatic valve operator failures. Other significant proximate causes are "valve stem out of adjustment/binding" (7%), "failed spring/spring pack"(5%), "failed seal ring" (4%), "failed solenoid coil", "lock nut/bolt problems" (2%), and "failed control switch" (2%) (Figure 5.32). "Others" in this figure include "dirt buildup", "damaged gasket", "failed relay electronics", and "worn bearing."

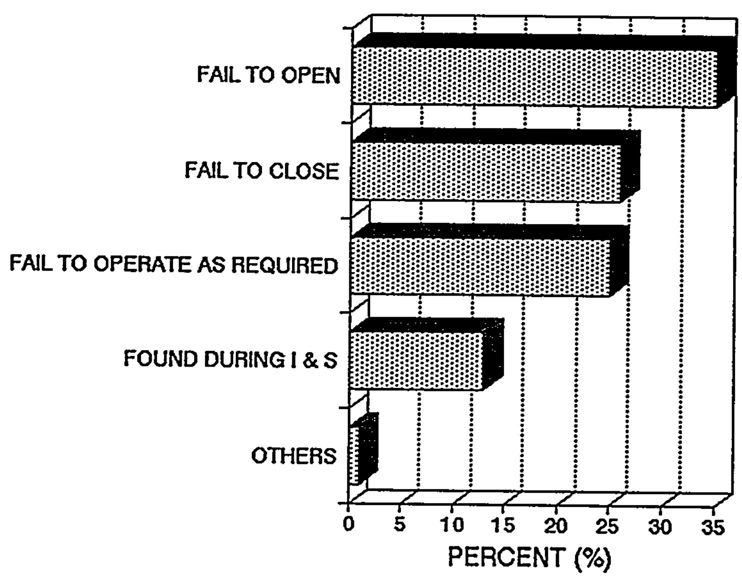


Figure 5.30 Failure modes for pneumatic valve operators

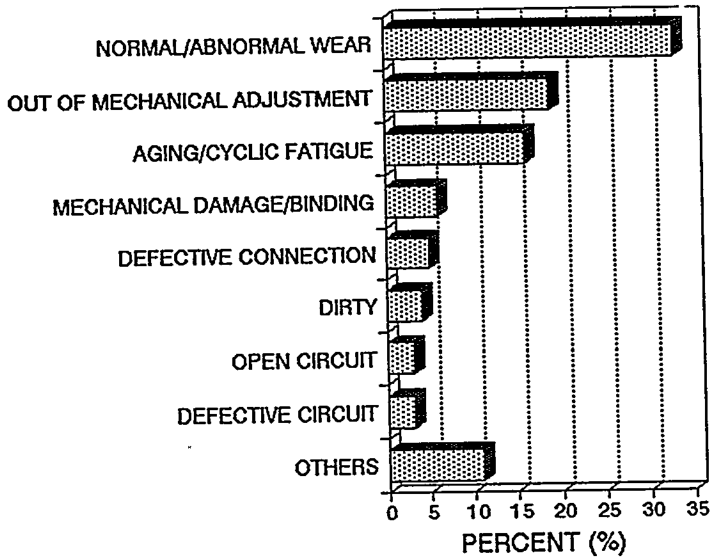


Figure 5.31 Failure causes for pneumatic valve operators in Westinghouse PWR CI functions

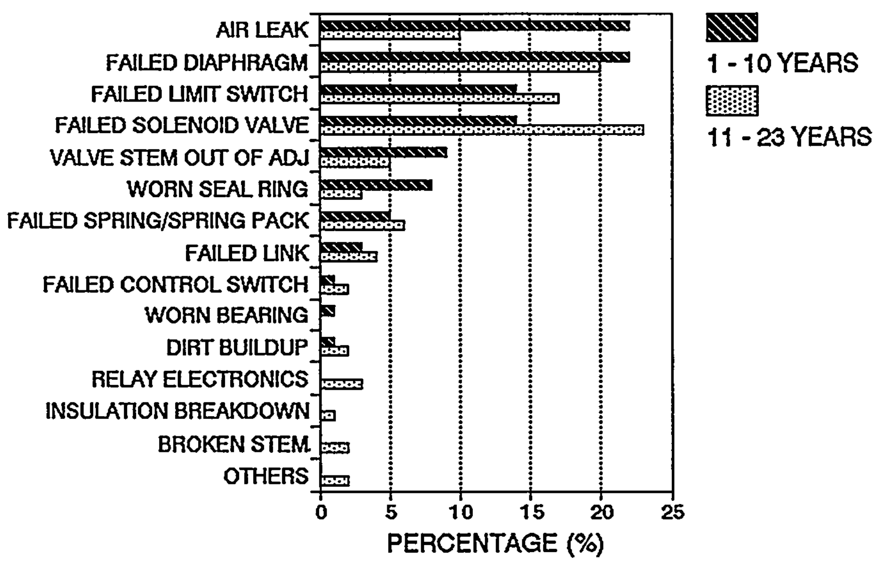


Figure 5.32 Proximate failure causes for pneumatic valve operators in Westinghouse PWR CI functions

To see the effects of aging on the major sub-components, the normalized failure frequencies for diaphragms, limit switches, solenoid valves, and air regulators/line are plotted in Figures 5.33 and 5.34. Solenoid valves show the dramatic effects of aging on the failure frequency. The failure frequency for the limit switches and the diaphragms also are very significantly affected by aging. For example, the failure frequency of the limit switches at 19 years is about 4 times higher than during earlier years. The failure frequency for diaphragms stays almost constant until 11 years, after which it increases rather rapidly, and at age 20, is almost 6 times higher. The effects of aging on failure frequency for the air regulators/lines (air leak) is not as dramatic as the above three sub-components, even though they also are significantly affected. The aging effects on these sub-components explains the steady increase in the failure frequency of the pneumatic valve operators shown in Figure 5.25.

5.2 Review of LERs for the PWR Containment Isolation Function

Though the information contained in the NPRDS database partly duplicate that found in the LER database, LERs have information on system failures which occurred during, and affected, plant operation. Normally, plant operators do not generate LERs for degradations and failures which were discovered during regularly scheduled maintenance or refueling outages. A detailed review of the information in both databases is necessary to a completely understand the cause and effects of aging.

For the six-year period (1988-1993), 488 LERs from 53 plants were written, documenting degradation or failures associated with the CI functions, the majority (316; 65%) were not related to aging (Figure 5.35). It is important to note that all of these were human-related failures, demonstrating a definite susceptibility to these types of causes. Many did not result in any valve malfunction, but did result in missed surveillance intervals, and incorrect procedures. However, the remainder resulted in spurious, unneeded valve actuations, which induce additional operating stresses to the components and the respective systems, and over time, may contribute to their aging. Twenty seven LERs (5.5%) documented age-related failures, while an additional 112 (23%) were potentially aging related. The remaining thirty three LERs had insufficient information to make a determination.

Though these human-related events are out of scope of this aging study, attention is called to them because they highlight the susceptibility of these components to this particular cause of failure. Over time, inadvertent actuations resulting from these occurrences may increase operating stresses both on the individual components, and the systems which perform the containment isolation function, and contribute to their aging.

As discussed in Section 2, the containment isolation function is performed by components which are part of various plant systems. Normally, these components are passive, and function only during periodic testing, or during an accident. Their failures typically do not render the remainder of the system inoperable. For example, failure of the isolation valves for the component cooling water will not affect the ability of the system to operate normally. However, often in the event of a CIV failure, the entire system, or a individual train, nonetheless, must be removed from service.

As with the NPRDS reviews and the reviews of the BWR LERs, which will be discussed in Section 6, only the aging and potential aging failures will be reviewed in depth. The CI function is primarily performed by valves and airlocks, the components which were reported failed most frequently on the LERs (Figure 5.36). Eighty percent of the LERs documented either valve or valve operator failures, followed by penetrations at 10% (primarily equipment and personnel airlocks).

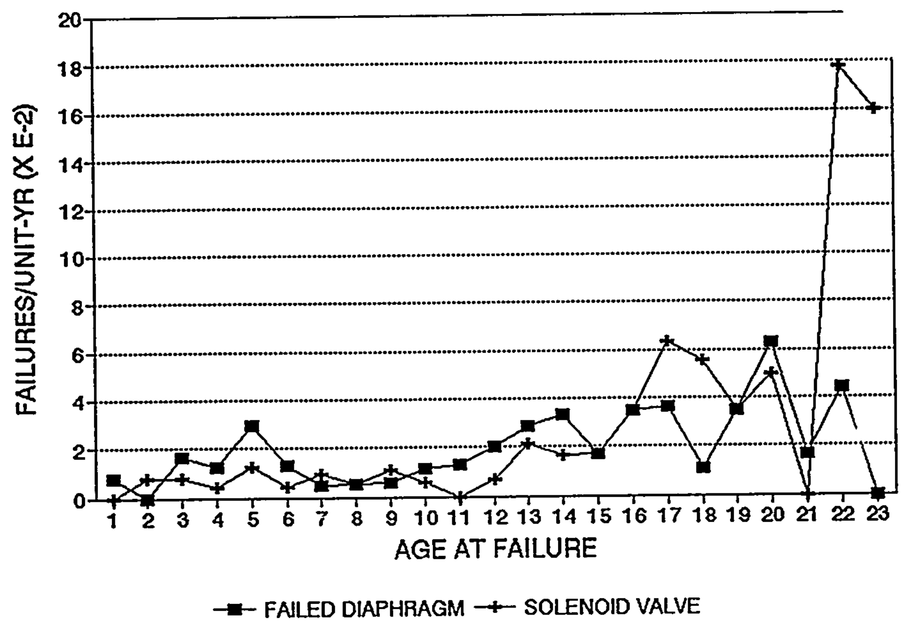


Figure 5.33 Normalized failure frequencies for diaphragms and solenoid valves in pneumatic valve operators

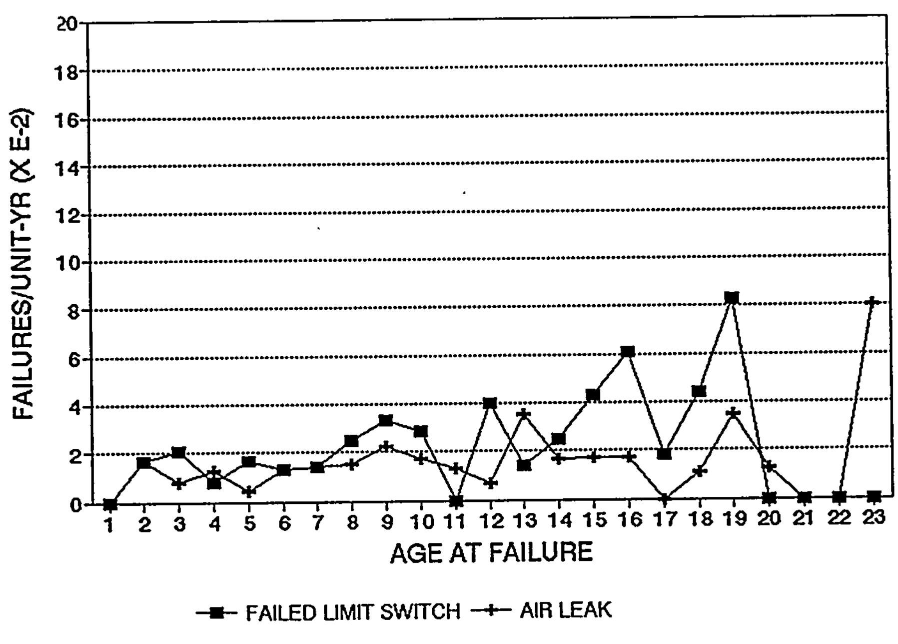


Figure 5.34 Normalized failure frequencies for limit switches and air regulators/lines in pneumatic valve operators

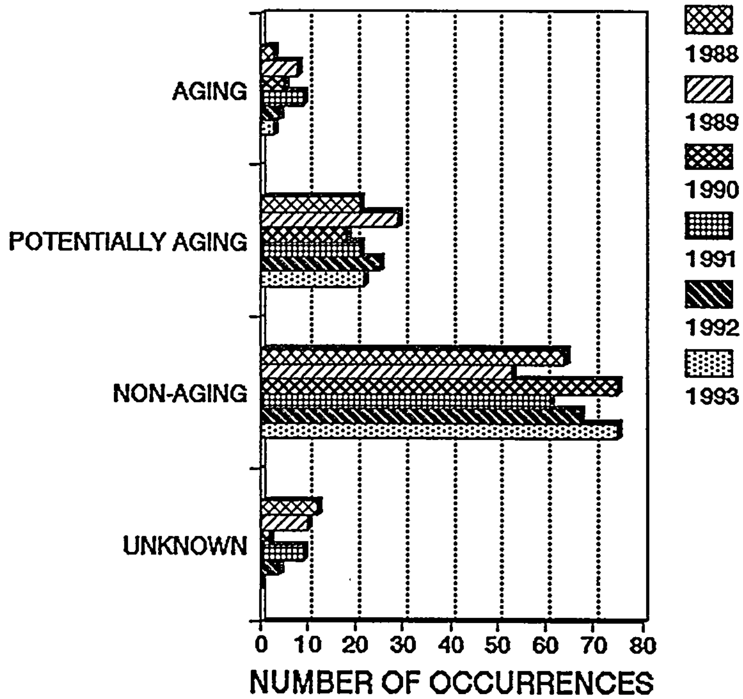
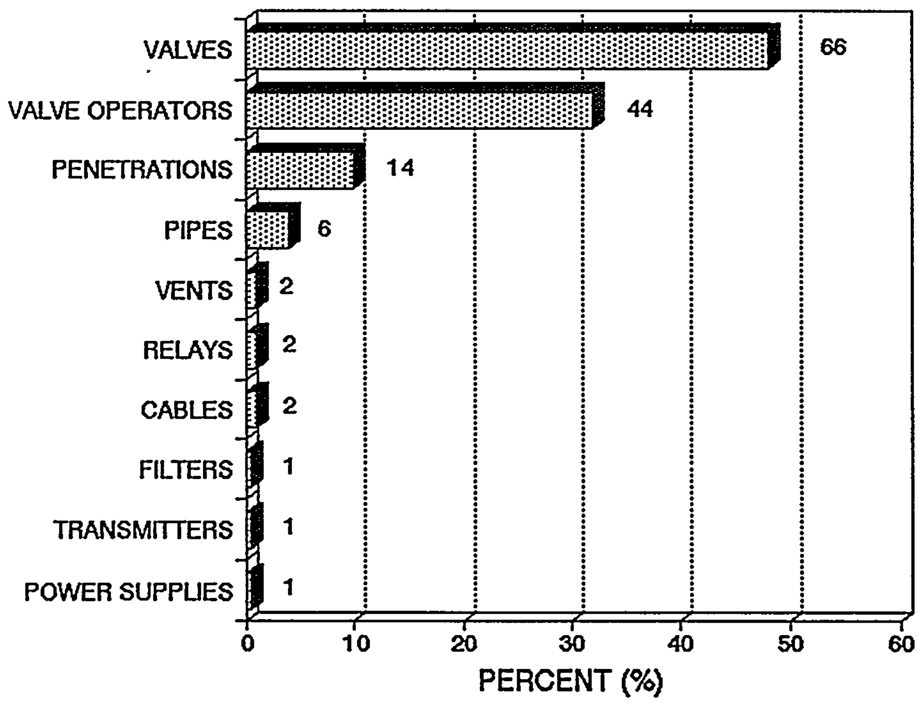


Figure 5.35 Percentage of PWR CI aging related LERs (1988-1993)



AGING & POTENTIALLY AGING

Figure 5.36 PWR CI components failed (1988-1993 LERs)

Table 5.4 shows the causes of failure for these LERs. As shown in this table, for all three types of components, the majority of the LERs did not have sufficient information to determine the cause; this lack is significant because it may be the result of either poor, or no root cause analysis. With the importance of a properly operating CI function, it is essential that these components are maintained properly, and are operable. An inadequate, or no root cause analysis of failure may allow degradation to remain uncorrected, which could result in further failure of components or systems. Of the causes of failure documented for valves, internal wear was the predominant failure cause, followed by corrosion and seal degradation. It is interesting that internal wear was the predominant cause, especially since these valves mostly are passive; this may point to inadequate design or application. For example, a check valve which is not properly sized for a particular flow rate may experience chatter, which over time cause it to fail due to mechanical wear. Corrosion is a difficult degradation mechanism to detect by functional testing, which is used to demonstrate the operability of these valves. Corrosion failures will only be detected when the degradation has progressed so far that the internal seat leaks, or when the valve is disassembled and inspected.

The failure causes for the valve operators are a function of the type of valve. Relay and switch failures occurred on MOVs, and solenoid and pneumatic failures on SOVs and AOVs, respectively. The particular valve operator chosen for a specific CIV may vary between plants and systems, and as a result, the population varies. Thus, the numbers of the valve operator failures shown in Table 5.4 do not indicate any inherent design-oriented weakness of any valve operator type. Similarly, no particular cause of failure was apparent for the reported penetration failures.

Table 5.4 Causes of PWR CI Component Failures (1988-1993 LERs)

Failure Cause	Number of Failures		
	Valves	Valve Operator	Penetrations
Not Specified	34	15	8
Worn Internals	15	2	-
Internal Corrosion	6	4	-
Seal Degradation	5	-	4
Air Supply Failure	-	1	-
Solenoid Failure	-	9	-
Relay Failure	-	3	-
Degraded Switch	-	6	1
Miscellaneous	7	2	1

From a review of the causes of failure, it was anticipated that leakage, both internal and external, would be the predominant failure mode, which was the case (Figure 5.37). Thus, internal leakage was the failure mode for approximately 70% of valves. Valve operator failures resulted in improper operation (immovable or improper operation), while all of the penetration failures resulted in loss of the pressure retaining boundary.

CI valves and airlocks generally are passive components, whose primary responsibility is to provide an adequate seal. Since degradation is unlikely to be detected during regular operation, it is essential that the periodic testing is adequate to detect degradation before the component fails. Containment isolation is ensured through a combination of Technical Specification and the In-Service testing programs. Therefore, it is not surprising that the majority of valve and penetration failures were detected by these tests (Figure 5.38). Other isolated, random failures were detected by normal and incidental observations by operating personnel (predominantly external leakages), and when incidental corrective maintenance was undertaken. Valve operator failures were detected by operational abnormalities, though it is suspected that many also were detected by stroke-time testing. Early degradation of the valve seats from corrosion may not be detectable by the normal seat-leakage tests. It is essential that plant operators remain cognizant of this failure mechanism, and investigate increasing trends in leakage rate which should be corrected before they exceed a defined limit (Technical Specification or ASME Section XI), and action is required.

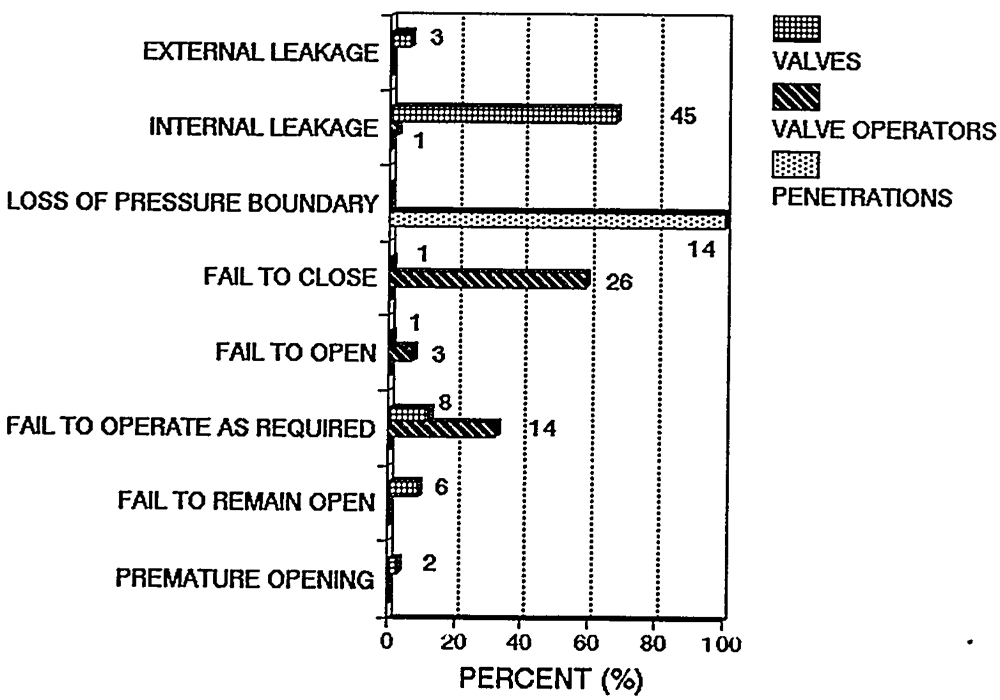


Figure 5.37 PWR CI component failure modes (1988-1993 LERs)

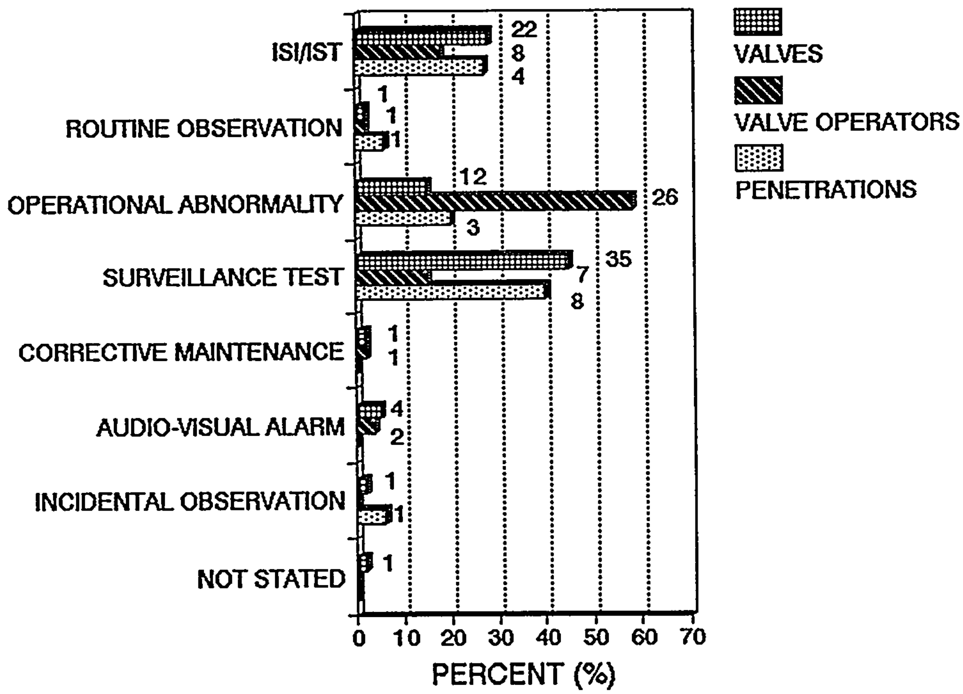


Figure 5.38 BWR CI failure detection methods (1988-1993 LERs)

The majority of the reported failures could be corrected by repairing or replacing the sub-components (Figure 5.39). Only 15% of the reported valve and valve operator failures were serious enough to require replacement of the whole component.

The most common effect on the system function resulting from failures of the CI valves and penetrations was its degraded system operation, with only 14 failures (21%) causing the loss of system function (Figure 5.40). However, 50% of the valve operator failures resulted in loss of system function; examples were check valves which failed closed, spurious actuations of vent valves, and excessive inter-system leakage. Primarily due to the redundancy provided by the CI components, the subsequent effects on the plant were not significant (Figure 5.41). Nevertheless, approximately 20% of these failures resulted in reactor trips, again due to spurious actuation, excessive leakage, or loss of instrumentation and control signals. These events also caused the actuation of and increased stresses on other safety-related components.

5.3 Summary of PWR Operating Experience

There were 2,042 failures of the components which provide the containment isolation (CI) for Westinghouse PWRs during the 5 years, 1989 - 1993, that were reported to NPRDS; about 76% were aging-related failures. Most of them involved valves (965;47%) or valve operators (668;33%), which together accounted for more than 80% of all the component failures in the CI function.

About 74% of the valve failures, and about 73% of the valve operator failures were aging-related. The globe valve is the major type used for CI functions in the Westinghouse PWRs, and accounted for

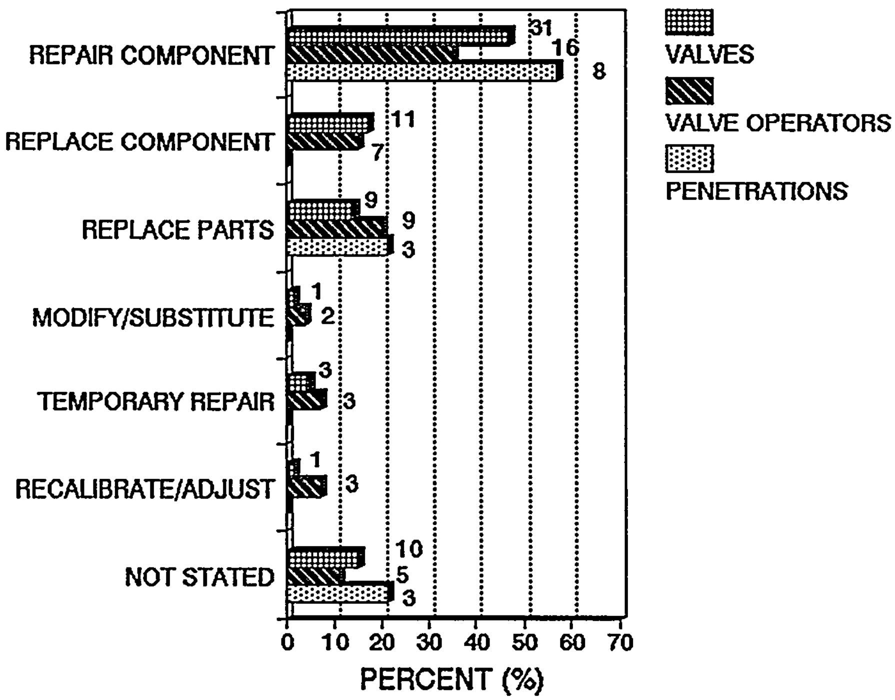


Figure 5.39 PWR CI component corrective actions (1988-1993 LERs)

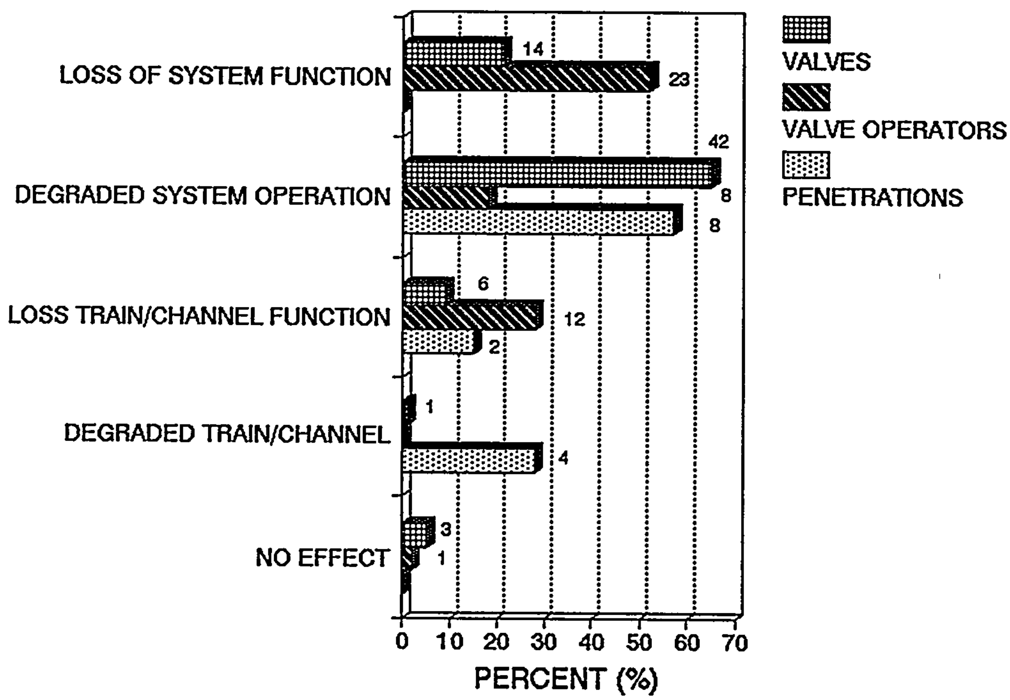


Figure 5.40 PWR CI system effects (1988-1993 LERs)

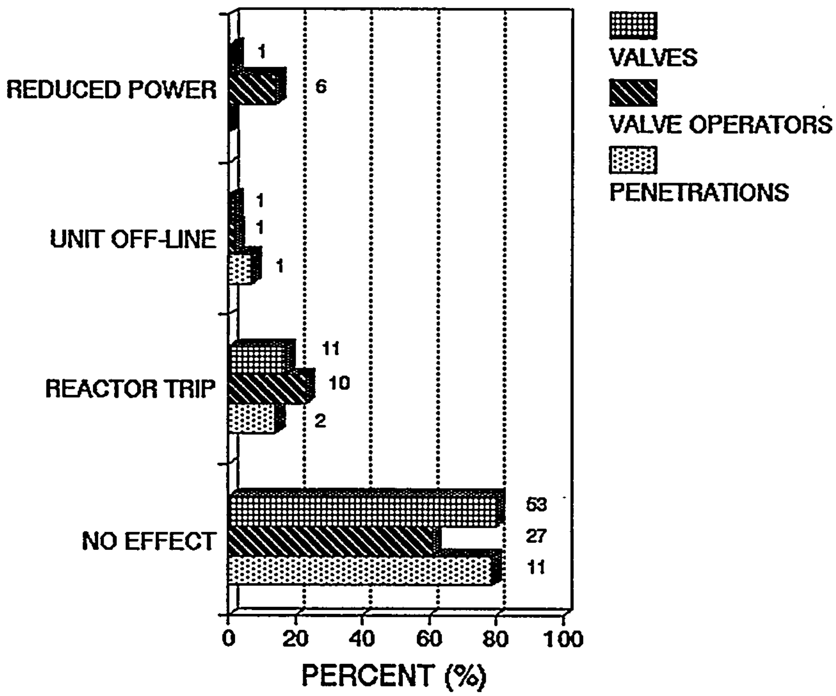


Figure 5.41 PWR CI plant effects (1988-1993 LERs)

41% of all the aging-related valve failures in the CI functions. The other types are check valves (18%), gate valves (15%), and butterfly valves (14%). The major types of valve operator are pneumatic operator and AC electric motor operator, which accounted for about 74% and 22%, respectively, of the aging-related failures.

The failure-frequency curve for all valves shows a high peak at 3 years, which decreases slowly until 11 years, after which it increases rapidly. Globe valves and gate valves individually showed similar trend as that for all valves combined. On the other hand, for check valves and butterfly valves, the failure frequency did not change much except to peak at the age of 23 for the former and at 18 years for the latter.

The failure frequency curve for globe valves shows a high peak at 3 years, after which it decreases to a low value until 11 years. Starting from 12 years of age, the frequency again increases reaching a high peak between 16 - 18 years. The major failure modes for globe valves are internal leaks (46%), external leaks (25%), and failure to close (13%). The major causes are aged seats/discs (37%), worn packing (19%), and corrosion product/dirt buildup (18%). As the globe valves get older, the fractions of the failures caused by aged seats/discs and worn packing increase, which, respectively, increase the fractions of internal and external leaks. After about 10 years, there is a large increase of failures caused by worn gaskets, which also cause external leaks. The first peak at 3 years in the frequency curve is mainly due to the failures caused by the buildup of corrosion products/dirt and by worn packing. The second peak at 12 years is caused by worn seats/discs, and the peak at 16 - 18 years is mainly caused by worn packing.

There are only two major failure modes for check valves, internal leaks (69%) and failure to close (21%); they were not affected by aging. Aged (worn/corroded/damaged) seats/discs and corrosion products/dirt buildup are the main causes of check valve failures, which also were not affected significantly by aging. There are two peaks, at 5 years and at 16 years in the failure frequency curve; the first one was caused by combination of corrosion products/dirt buildup and worn seats/discs, and the second one by corrosion products/dirt buildup.

The failure frequency curve for gate valves peaks at 3 years and 17 years, in both cases caused by worn seats/discs and worn packing. The main failure modes are internal and external leaks, accounting for about 77% of the failures. As the valves get older, higher percentages of the failures are due to internal leaks (44%) rather than external leaks (30%), and their causes are worn packing, aged seats/discs, and corrosion product/dirt buildup. The failure frequency curve for butterfly valves does not significantly peak until 18 years, when the failures increase rapidly. This peak is due to increased failures caused by corroded/worn seats/discs. The butterfly valves failed mainly from internal leaks, which accounted for about 65% of the failures. The major proximate causes are corroded/worn/damaged seats/discs (44%) and corrosion product/dirt build-up (30%). This shows that most of the failures are caused by corrosion and aging of seat and disc.

During the period studied, one valve operator failure resulted in a unit off-line, and three that resulted in reduced power operation. However, none of them affected the safety of personnel or the public due to the redundancy in the CI function.

The types of valve operators most often reported in NPRDS as failing are pneumatic (74%) and AC electric motor (22%) operators, and about 73% of these failures are aging-related. The failure frequency of the valve operators continuously increases as they get older, in contrast to the effects of aging on failure frequency for the valves. Because most of the data on valve operators are from the pneumatic operators (74%), the combined frequency curve is dominated by them.

Failure to open (35%) is the leading failure mode of pneumatic valve operators, followed by failure to close (26%), and failure to operate as required (25%). For the containment isolation function, "failure to close" is the only important mode, which makes the number of component failures that directly affect the CI function much less than those analyzed in this report. The failure frequency of pneumatic valve operators increase continuously but slowly for the first 15 years, then rises rapidly from 16 years. The major causes for failures are the failures of diaphragm (18%), solenoid valve (16%), limit switch (15%), and air regulator/line (13%). The effects of aging on these proximate causes all are similar; the numbers of failures caused by these show slow increases during the first 14 - 15 years, then followed by rapid increases.

The failure frequency of the AC electric valve operators did not change much over time. The major failure mode is fail to close, accounting about 35% of the total. The major proximate causes for the AC electric valve operators are the failures of torque switches (24%), limit switches (23%), and motors (7%).

During the six years, 1988 - 1993, there were 139 LERs caused by aging-related failures of components in the Westinghouse containment isolation functions; the failures of valves (47%), valve operators (32%), penetrations (10%), and pipes (4%) caused the LERs. The percentages for failures of valves and valve operators are very close to those reported to NPRDS, which are 49% and 34%, respectively. The causes, identified from LERs, are worn seat/disc, corrosion, and seal/gasket degradation, which agree with the results obtained from the NPRDS data analysis. For valve operators, the identified causes are failed solenoid valves/coils and failed torque/limit switches, also agreeing with the NPRDS data analysis. The major failure mode for the valves is internal leak, which accounted for almost 70% of the LERs. For valve operators, failure to close is the major mode, accounting for about 60% of LERs; this value differs from the NPRDS data analysis because most of the component failures including those that do not affect the plant operation are reported to NPRDS, while only the failures that affect the plant operation are reported to LERs.

6. BWR OPERATING EXPERIENCE

6.1 NPRDS Data Analysis

The following systems contribute to the containment isolation function of the Boiling Water Reactors (BWRs):

Main Steam System
Reactor Core Isolation Cooling System (RCIC)
Feedwater System
Nuclear Steam Supply Shutoff System
Standby Gas Treatment System
High Pressure Coolant Injection System (HPCI)
Essential Service Water System
Reactor Building Closed Cooling Water System
Residual Heat Removal/Low Pressure Coolant Injection System
Low Pressure Core Spray System
Combustible Gas Control System

Table 6.1 shows the component failures that were related to the containment isolation function. As discussed earlier, several different systems contribute to this function. However, most component failures (72%) are from the nuclear steam supply shutoff system because it contains many bistables/switches, radiation detectors/transmitters, and relays in addition to valves and valve operators, and their failures are reported to nuclear steam supply shutoff (NSSS) system. Other systems, listed above, consist mainly of valves and valve operators.

Approximately 59% of the aging-related valve failures, and about 66% of the aging-related valve operator failures were from the NSSS system. On the other hand, much larger portions of bistables/switches and radiation detectors/transmitters, 85% and 88%, respectively, were from the NSSS system. Thus, for analyses of the aging effects on valves and valve operators, all the failures from different systems were combined, while for bistables/switches and radiation detectors/transmitters, only failures from the NSSS system were analyzed.

6.1.1 Valves

When all failures of the valves related to the containment isolation function are normalized by the number of the plants, the failure frequency as a function of "age at failure" can be plotted (Figure 6.1). The initial peak in failure frequency occurs around 6 years, after which it decreases until 11 years of age. Starting at 12 years, the frequency then increases rapidly. When these valve failures are analyzed for different types of valves, gate valves, check valves, and globe valves are the ones that have the most failures in the containment isolation function. Because they have different designs and sub-components, the aging effects on failure frequency probably differ for each type. Thus, to gain insights on the reasons for the shape of the frequency curve in Figure 6.1, and to better characterize the aging of the valves, the aging-related failure data are analyzed separately.

As shown in Figure 6.2, 41% of the failures in the containment isolation function are those of gate valves, followed by globe valves (27%), and check valves (23%).

Table 6.1 Containment Isolation Failures Contributed by Different Systems

Systems	Components						Aging	Non-Aging	Total
	Valve	Valve Operator	Circuit Breaker	Bistable/ Switch	Rad Detector/ Transmitter	Relay			
Main Steam	23/2	3/1	2/2	7/5	8/1	0/1	43	12	55
RCIC	32/3	3/5	2/1	19/7	0/1		56	17	73
Feedwater	26/7	3/1	1/0		1/0		31	8	39
Nuclear Steam Supply Shutoff	263/39	92/49	19/6	316/44	100/59	68/13	858	210	1068
Standby Gas Treatment	4/0	2/1		1/0		1/0	8	1	9
HPCI	15/8		1/0	17/7	1/0		34	15	49
Essential Service Water	12/1	3/1					15	2	17
Reactor Building Closed Cooling Water	25/2	8/4	2/1				35	7	42
Residual Heat Removal/ Low Pressure Coolant Injection	22/8	12/4	7/0	11/2			52	14	66
Low Pressure Core Spray	16/3	12/1	6/4				34	8	42
Combustible Gas Control	10/0	1/0			4/1	0/1	15	2	17
Total	448/73	139/67	40/14	371/65	114/62	69/15	1181	296	1477

Note: The numbers before the "/" are those of aging-related, and the numbers after "/" are those of non-aging failures.

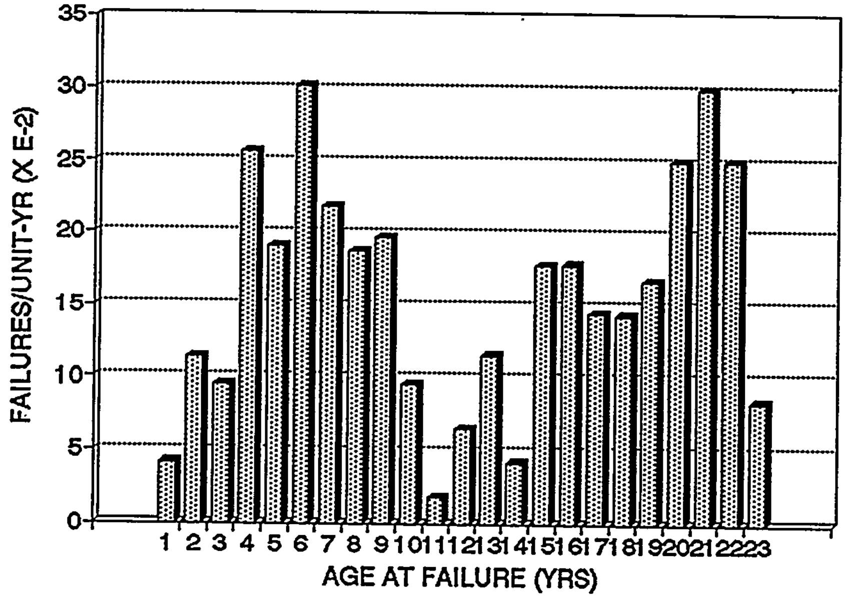
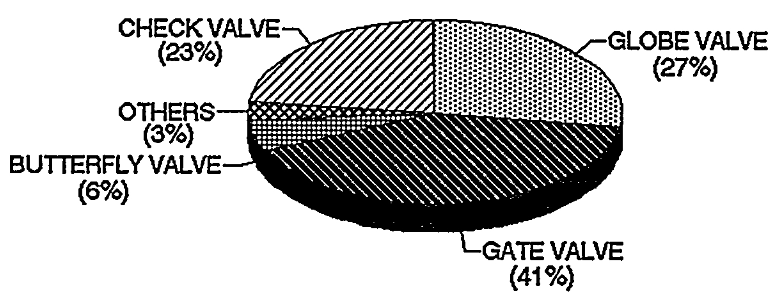


Figure 6.1 Normalized failure frequency for valves as a function of age at failure



SOURCE : NPRDS (1988 - 93)

Figure 6.2 Different valve types in the containment isolation function

6.1.1.1 Effects of BWR CI Function Valve Failures on Plant Operation

Five CI function valve failures resulted in "unit off-line" during the 6 years studied; three were external steam leaks due to packing failures, and the fourth was an external leak due to a worn seal ring. The last one was an internal leak of a check valve caused by a worn disc and seat.

Three of these failures occurred when the plants were at power, so the reactor was shutdown for repairs. The other two occurred during the start-up process, and caused delays.

One example of an external leak failure is that of a RCIC steam supply isolation motor-operated valve, which blew its packing, and more than 5 gallons per minute of reactor coolant leaked into the containment. The leak could not be stopped by tightening the packing, and unit was taken into cold shutdown for repairs. The packing failure was due to aging, and the valve was repacked.

Another example of an external leak is the following. With reactor at 100% power, a feedwater system outside isolation valve leaked steam at its bonnet into a steam line tunnel, which activated a high temperature alarm at the control room. The unit had to go off-line to repair the valve. A worn pressure seal ring (gasket) caused the steam leak. The pressure ring is a silver-plated carbon steel piece, which was replaced.

An example of an internal leak is a failure of a check valve. With the plant in a refueling outage, a feedwater containment outboard isolation check valve failed LLRT, and the start-up was delayed until the valve was repaired. The cause of the problem was a worn disc and seat, and they were lapped until the sealing was satisfactory.

One valve failure resulted in reduced power operation during the 6 years. The solenoid valve for the main steam isolation valve leakage control isolation valve was leaking about a pint of condensed steam per minute from the drain line, which constituted a breach of the integrity of the primary containment.

All the valve failures described above were leaks, either internal or external, which did not affect the safety of the plant personnel nor the public. Also, they were detected by surveillance tests or detected immediately by the control room personnel. However, at the same time, all of them resulted in reactor shutdown or reduced power operation, which affected the cost of operation. If a proper aging management can prevent and reduce some of these failures, significant plant benefits will result.

6.1.1.2 Gate Valves

Failure Frequency

When the failures of gate valves are normalized to show the failure frequency per unit-year as a function of age at failure, the curve resembles a typical bathtub curve (Figure 6.3). Comparing this curve with that for gate valves in PWRs (Figure 5.6), they appear very similar except that the frequency for BWRs seem higher than that for PWRs. The failure modes and causes were analyzed for two different periods, 1-10 years, and 11-23 years, to identify any effects of aging on them.

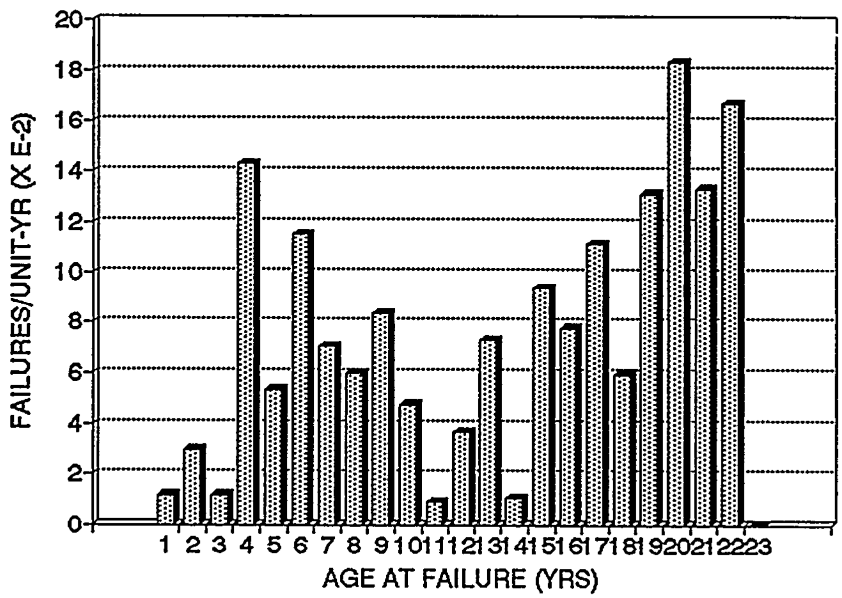


Figure 6.3 Normalized failure frequency of gate valves as a function of age at failure

Failure Mode

We analyzed the failure modes for the gate valves for two age periods, based on the shape of the frequency curve in Figure 6.3; the results show the relative proportion of each failure mode (Figure 6.4). Leakage was the predominant failure mode (>70%), especially internal leaks, which accounted for more than half of the failures. Furthermore, the failure mode did not change much over time, except for a slight decrease in the percentage of the "fail to close" mode. These effects will be discussed with the information on failure causes in the following section.

Failure Cause

Approximately 55% of the failures were due to "normal/abnormal wear," and about 15% due to "dirty internals" (Figure 6.5). "Corrosion" (7%), "aging/cyclic fatigue" (5%), "mechanical damage/binding" (5%), and "out of mechanical adjustment" (5%) also caused a significant number of failures. The analysis for proximate causes provided more detailed information, showing that as the gate valve gets older, aging of the wedge causes a significant portion of their failures (Figure 6.6). During the earlier years (1-10 years), only about 3% of the failures were due to worn/damaged wedges, while it increased to 23% during later years (11-23 years). There are two types of gate valve discs, parallel and wedge types. However, the terms disc and wedges are used interchangeably. When NPRDS narratives state that a worn seat/disc is the failure cause, we differentiated it from a worn wedge. This is because most failures are due to a worn seat rather than a worn disc, when a worn seat and disc are

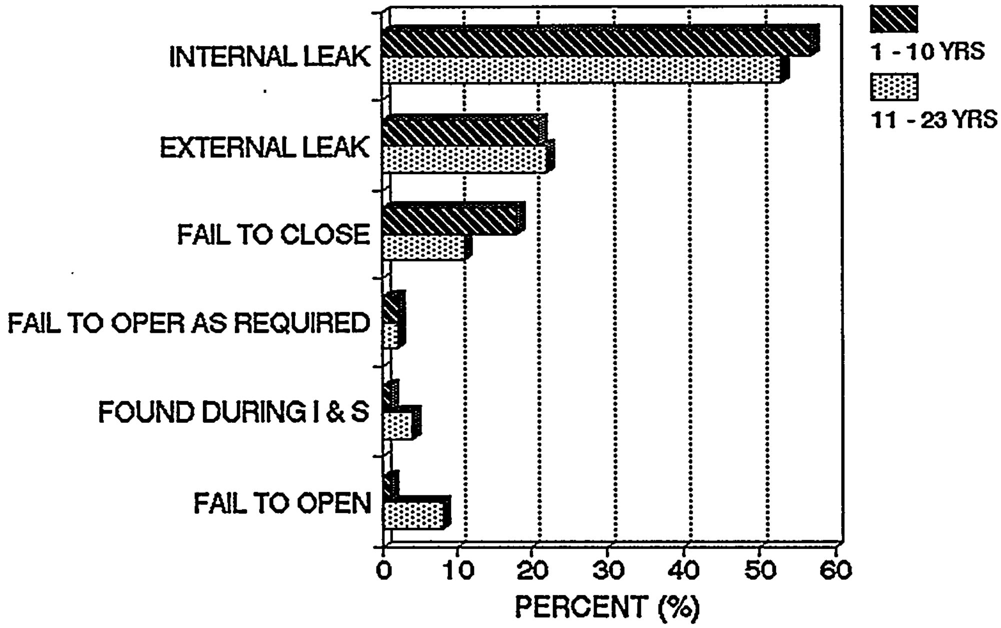


Figure 6.4 Failure modes for BWR gate valves

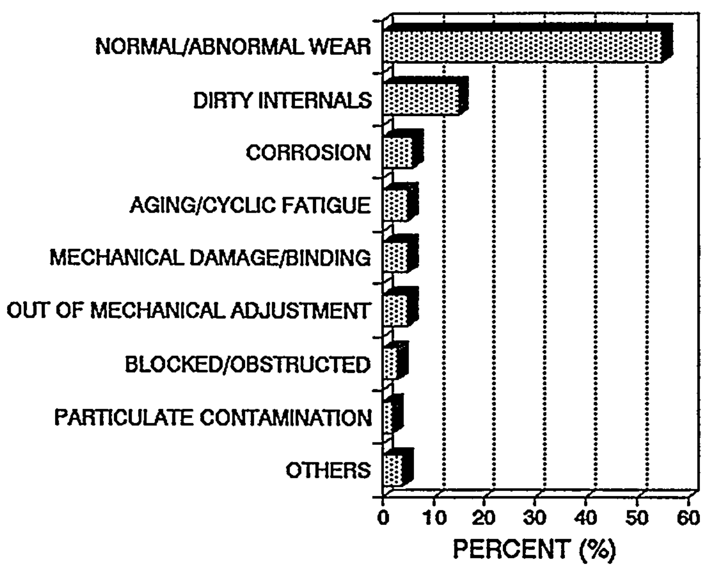


Figure 6.5 Failure causes for BWR containment isolation gate valves

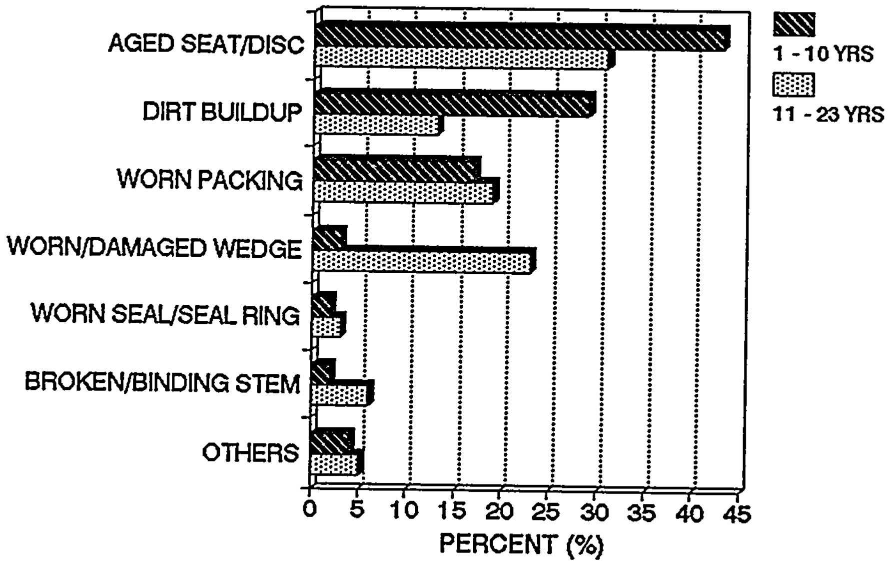


Figure 6.6 Proximate causes for failures of BWR containment isolation gate valves

stated as the cause. However, when the narrative states that a worn wedge is the cause, it means an aged (worn/corroded/damaged) disc caused the failure. Thus, it may be concluded that seats usually start to age before discs.

Figure 6.6 also shows that aged seat/disc, dirt/corrosion product buildup, and worn packing are the major proximate causes. "Worn packing," "worn seal/seal ring," and "broken/binding stem" caused a larger portion of the failures during the later years compared to the earlier years, as expected. To see the effects of aging on three major proximate causes, they were normalized and plotted as a function of age at failure (Figure 6.7); there was a high peak at 4 years. However, this peak was caused by multiple failures at one plant (seven out of the nine). The failures caused by dirt/corrosion product peaks at around 6 years, after which they stay low until they begin to increase at 16 years. This finding explains the decrease of the "fail to close" mode discussed earlier, since most of these failures are caused by buildup of dirt/corrosion products. This figure also shows that failures caused by worn seat/disc peak at 6 - 8 years, after which they stay low until they begin to increase at 17 years. The wear and corrosion of seat and discs/wedges are the main causes for the internal leaks of gate valves. The failures caused by worn packing stay almost constant for the first 15 - 16 years, after which they increase; this explains the slight increase of "external leak" failures during the 11 - 23 years.

The difference between Figure 6.7 and Figure 5.20 for PWRs is noticeable. For PWR gate valves, the failures caused by dirt/corrosion product buildup, worn seat/disc, and worn packing generally increase as the valves get older without showing any significant peaks. These different aging effects may be due to different fluid chemistry and temperatures, and possibly to different materials used by various manufacturers.

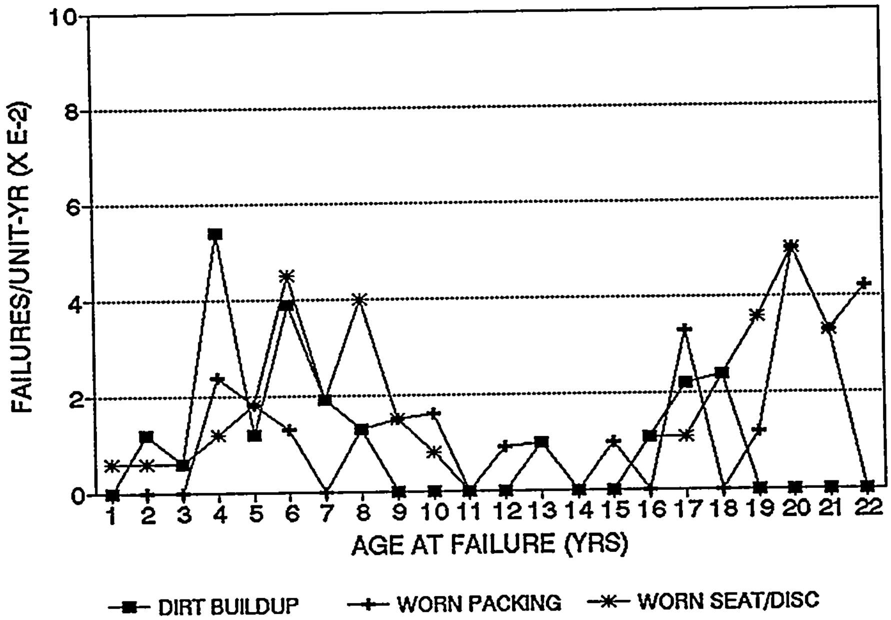


Figure 6.7 Effects of aging on proximate causes for failures of BWR CI gate valves

6.1.1.3 Globe Valves

Frequency of Failures

Figure 6.8 shows the normalized failure frequency for the globe valves in the containment isolation function. No dramatic changes are seen, except that the failure frequency continuously increases until 6 years of age. Between 11 and 13, it stays very low, after which it increases gradually. The effects of aging on modes and causes of failure again were studied by analyzing data from 1-10 years and then from 11-23 years.

Failure Mode

During the early years, about 73% of the failures of globe valves reflect leaks, internal (41%) and external (32%) (Figure 6.9). However, in the later years, about 35% are "fail to close." This large increase in the latter mode is coincident with the large increase in the "corrosion product/dirt buildup" (Figure 6.11). The NPRDS Manual [Ref. 36] states that leakage of the working fluid past the valve seat when fully closed should be reported as "internal leakage," and leakage through the valve because it has not closed completely is classified as "failure to close." It also states that the examples of "failure to close" include the failure to close due to corrosion, internal binding, missing or damaged parts, travel stop-nuts being out of adjustment, and excessive dirt or trash buildup. Thus, this change of predominant mode probably indicates that dirt/corrosion product buildup becomes an important cause of failure as the valves age. In fact, this was confirmed in our analysis of the causes of failures, which is presented below.

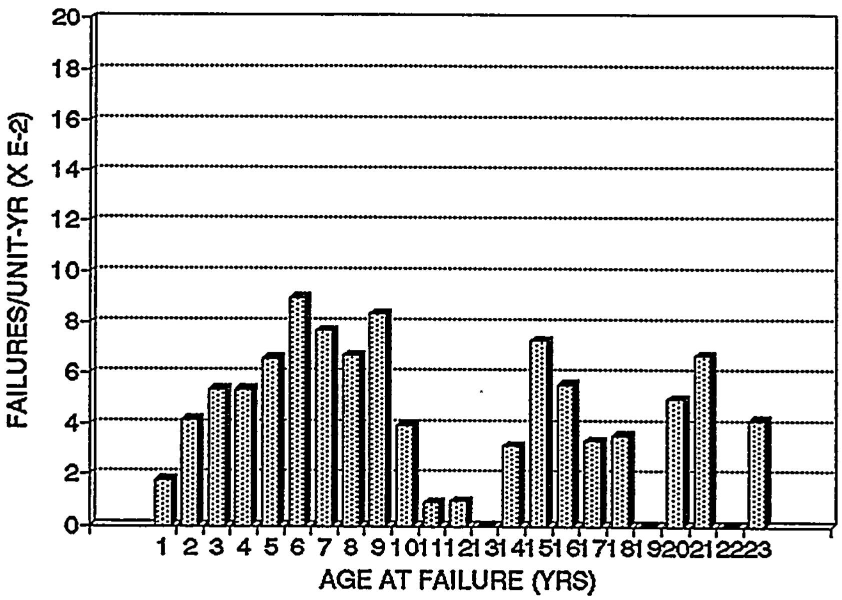


Figure 6.8 Normalized containment isolation globe valve failures as a function of age at failure

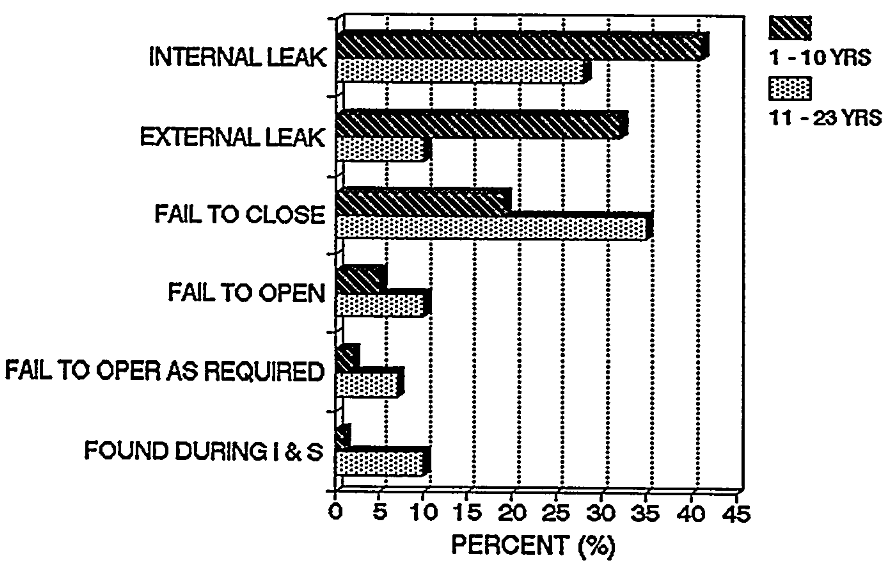


Figure 6.9 Failure modes for containment isolation globe valves

Failure Cause

The major causes for the globe valve failures are "normal/abnormal wear" (50%), "dirty" (15%), "corrosion" (8%), and "aging/cyclic fatigue" (7%) (Figure 6.10). "Mechanical damage/binding" (6%) and "particulate contamination" (4%) also caused globe valve to fail. However, as before, these NPRDS failure causes do not provide insights on aging effects. The proximate causes analysis developed at BNL provides more useful information, and can explain the aging effects on the failure modes. Figure 6.11 shows that the portion for the "aged seat/disc" was lower during the 11-23 years than during the earlier years; this explains the decrease of the "internal leak" in Figure 6.9, because most of them are caused by corroded, damaged, worn, seat or disc (aged seat/disc). The large increase of the corrosion product/dirt buildup also explains the increase in the "fail to close" mode.

When the numbers of failures caused by three major proximate causes, dirt/corrosion product buildup, worn seat/disc, and worn packing, are normalized and plotted as a function of age at failure, we obtained more detailed information on the aging effects (Figure 6.12). The most interesting trend is shown by the failures caused by buildup of dirt/corrosion products. It suggests, plausibly, that there may be a 7-year cycle for the buildup. On the other hand, failure frequency due to packing degradation stays almost constant until 7 years, after which it begins to increase. After 10 years, fewer packing failures are reported, probably because packing is a regular maintenance item, and most are replaced around 10 years. The aging trend for seat/disc is similar to that for packing, with peaks between 6 and 9 years, after which failure frequency becomes very low. Comparing this figure with that for PWRs (Figure 5.12), the most noticeable difference is the failures caused by corrosion product/dirt buildup, which causes a higher percentage of failures in BWRs. As mentioned earlier, this may indicate that corrosion is a bigger problem for BWRs than for PWRs.

For globe valves, about 57% of the failures were corrected by replacing the parts, and 30% by repairing components or parts; only 6% of the failures required replacing. Thus, the decreased failure frequency of the seats/discs may be because they wear less, either because of better replacement parts or the lapped surfaces match better than the new surfaces.

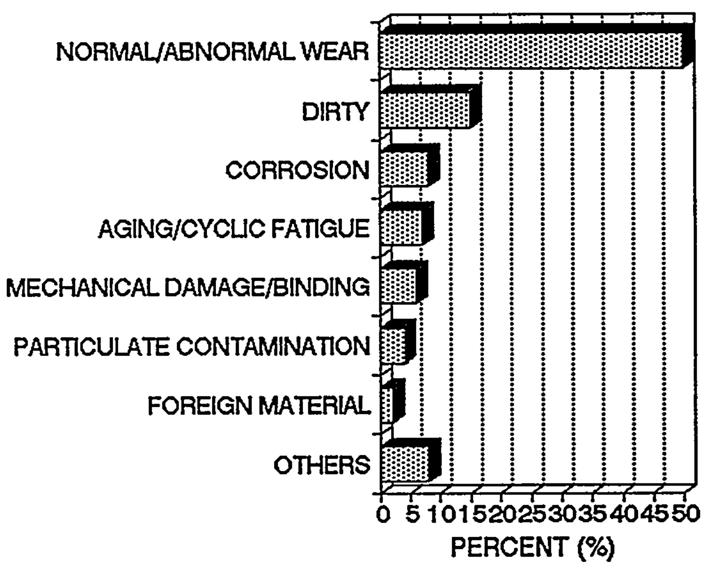


Figure 6.10 Failure causes for containment isolation globe valves

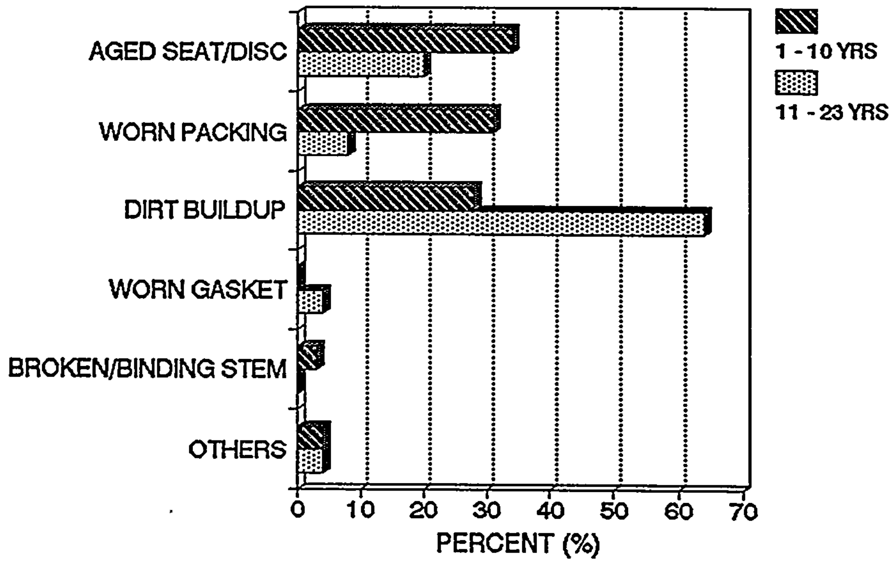


Figure 6.11 Proximate causes for containment isolation globe valves

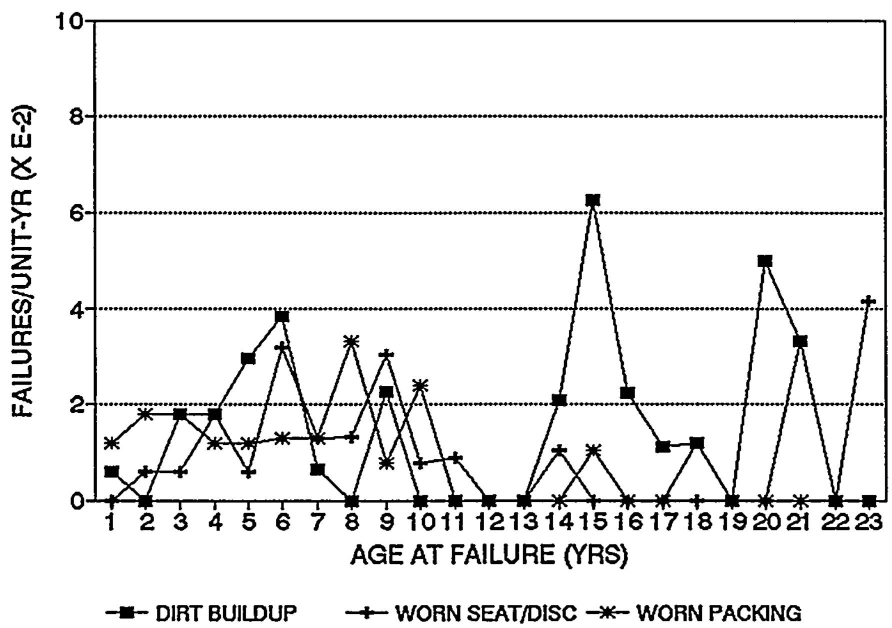


Figure 6.12 Normalized failure frequency for seat/disc, packing and dirt buildup as a function of age at failure

6.1.1.3 Check Valves

Failure Frequency

The failure frequency curve for the check valves in the containment isolation function is similar to that for the globe valves with a peak at 6 years (Figure 6.13). There is a gradual increase of the failure frequency during 10-21 years. As were the cases for the gate and globe valves, the failure data were divided into two age groups, 1-10 years and 11-23 years, to analyze aging effects.

Failure Mode

Figure 6.14 shows that the major failure modes for the check valves during the earlier years are "internal leak" (49%), "fail to close" (27%), and "external leak" (18%). As the check valves age, "internal leak" increased to 60%, while "fail to close" decreased to 8%; "external leak" stayed about the same. These effects of aging on failure modes are somewhat different from those for gate and globe valves, which will be discussed later again using the information on the proximate failure causes.

Failure Cause

The major failure causes for the check valves are "normal/abnormal wear" (57%), "dirty internals" (18%), "corrosion" (11%), and "mechanical damage/binding" (8%) (Figure 6.15), similar to those for gate and globe valves. The proximate cause analysis shows that, during the earlier ages, 1-10 years, about 74% of the failures of check valves were due to "corrosion product/dirt buildup" (41%) and "aged

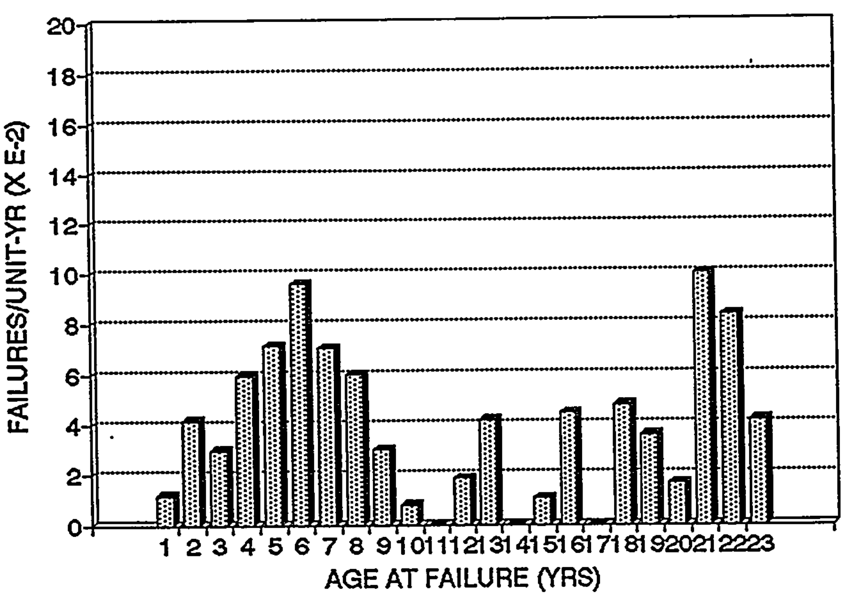


Figure 6.13 Normalized failure frequency for containment isolation check valves

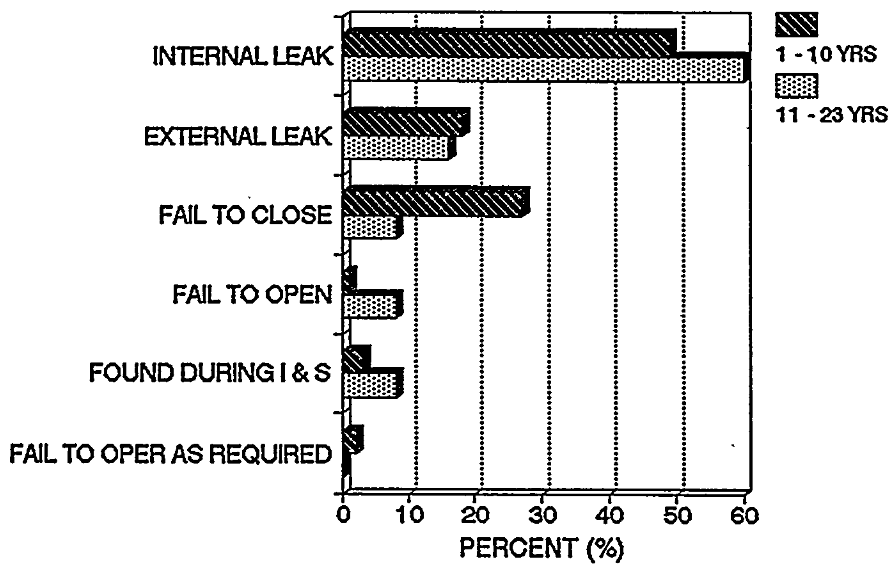


Figure 6.14 Failure modes for containment isolation check valves

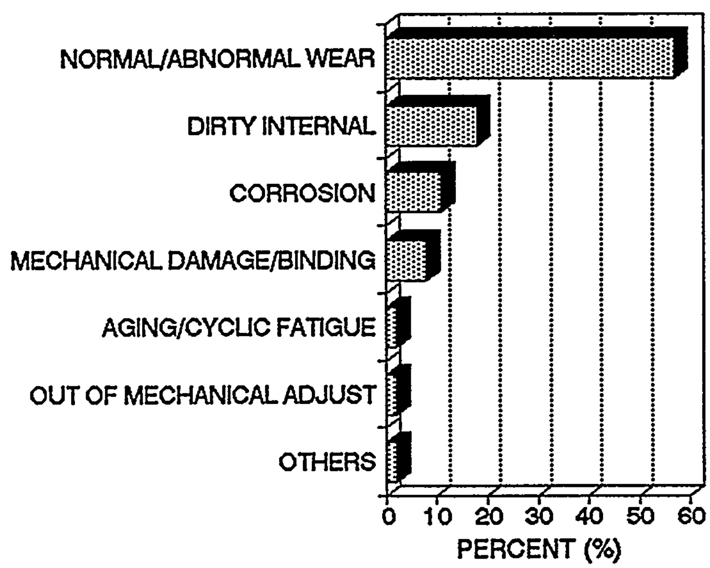


Figure 6.15 Failure causes for containment isolation check valves

seat/disc (corroded, damaged, worn, seat or disc)" (33%) (Figure 6.16). In the later years, aged seat/disc caused most of the failures (54%), increasing the number of internal leaks (Figure 6.14). On the other hand, as Figure 6.16 shows, during the later years, "corrosion product/dirt buildup" caused fewer failures (18%), and hence, a smaller percentage of "fail to close" failures. This result is similar to that for gate valves, but differs from that for globe valves. To understand this better, the normalized number of failures caused by corrosion product buildup and worn seats/discs are plotted as a function of age at failure in Figure 6.17. The number of failures due to corrosion product/dirt buildup increases continuously until 8 years, after which it stays very low; again, this trend is very similar to that for gate valves. The trend also is similar in the first six years to that for PWR check valves, but the number of failures due to buildup increased slowly as the valves got older. Other proximate causes, "worn packing," "worn gasket," and "broken/binding stem" caused higher percentages of failures in the later years than the earlier years, as expected.

6.1.2 Valve Operators

There were 205 failures of valve operators in the containment isolation function of BWRs during the period studied, of which 66% were aging-related. Two major types, AC electric and pneumatic, contributed to about 83% of the reported failures (Figure 6.18), and only these failures were analyzed. The numbers of reported, aging-related failures for the other types are too few for meaningful results. For the PWR containment isolation functions, the same two types contributed to about 96% of the valve operator failures (see page 5.18).

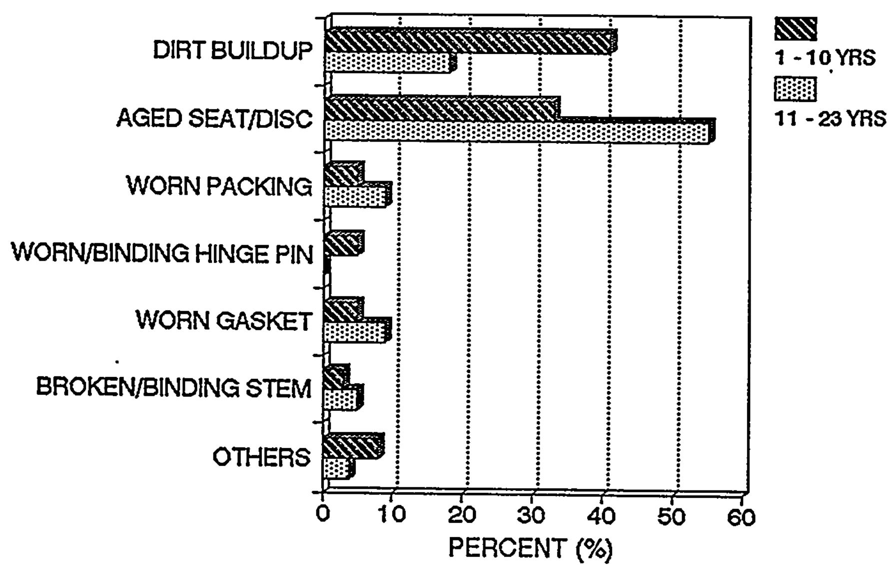


Figure 6.16 Proximate causes for containment isolation check valves

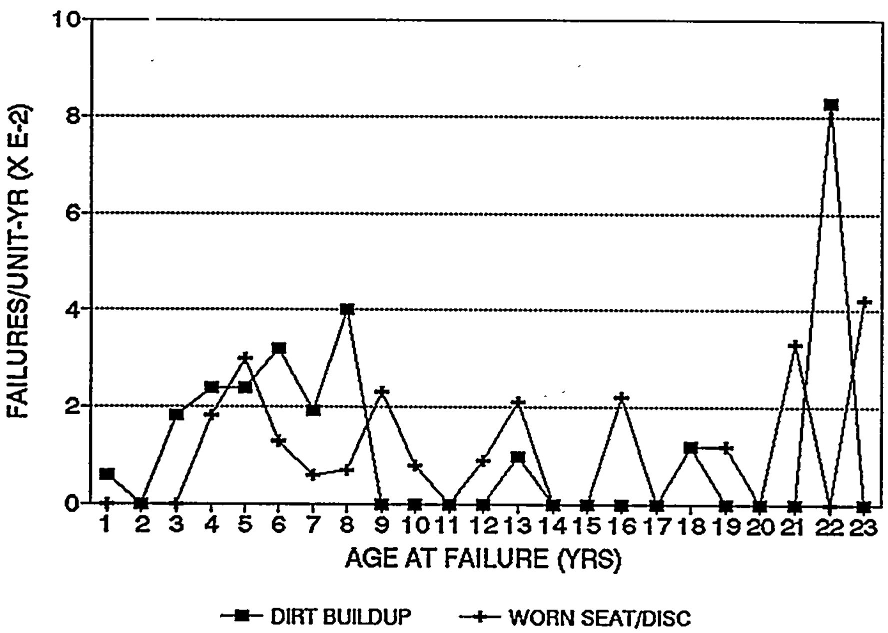


Figure 6.17 Normalized numbers of failures caused by corrosion product/dirt buildup and by worn seat/disc

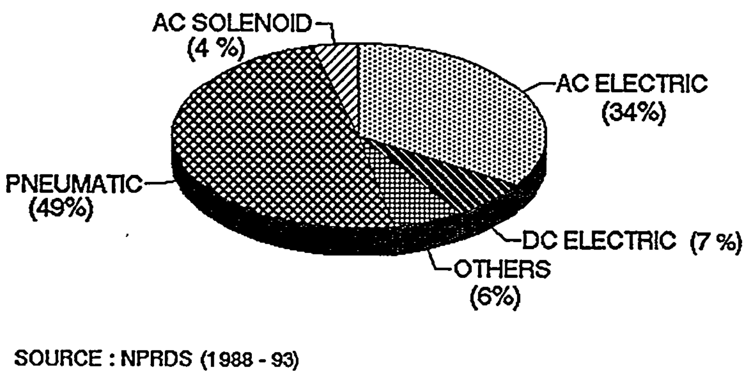


Figure 6.18 Different types of valve operators that failed (source: NPRDS)

Figure 6.19 gives the combined failure frequency of these two types of valve operators. Comparing this frequency curve to that for valves (Figure 6.1) shows that the failure frequency for valve operators is about one third of that for the valves; this is partly because not all valves have operators. The failure frequency curve for valve operators peaks at 6 years, and there is a higher peak at 16-17 years. The type of valve operator causing these peaks is discussed in the following sections.

6.1.2.1 Effects of Failures of CI Function Valve Operators on Plant Operation

No failure of a valve operator had any significant effects on plant operation, except one caused by a human error. While shutdown for refueling, it was found that a bracket for the air operator was cracked, and start-up was delayed for repair, which was reported as unit off-line. The cracking was due to previous unsatisfactory maintenance.

6.1.2.2 AC Electric Valve Operators

Failure Frequency

There were 45 aging-related failures of AC electric valve operators during the period studied. When these are normalized to show failures per unit per year, the curve peaks at 6 years, after which it resembles a typical bathtub curve (Figure 6.20). Clearly, this peak at 6 years is the main contribution to the 6-year peak in Figure 6.19, the combined frequency curve for the valve operators. Differences are apparent in comparing this CI function valve operator curve for BWRs with that of PWRs; in the latter, the frequency stayed almost constant for the first twelve years or so, while BWR frequency peaked at age 6, as discussed earlier. This difference will be discussed later when comparing the modes and causes of failures.

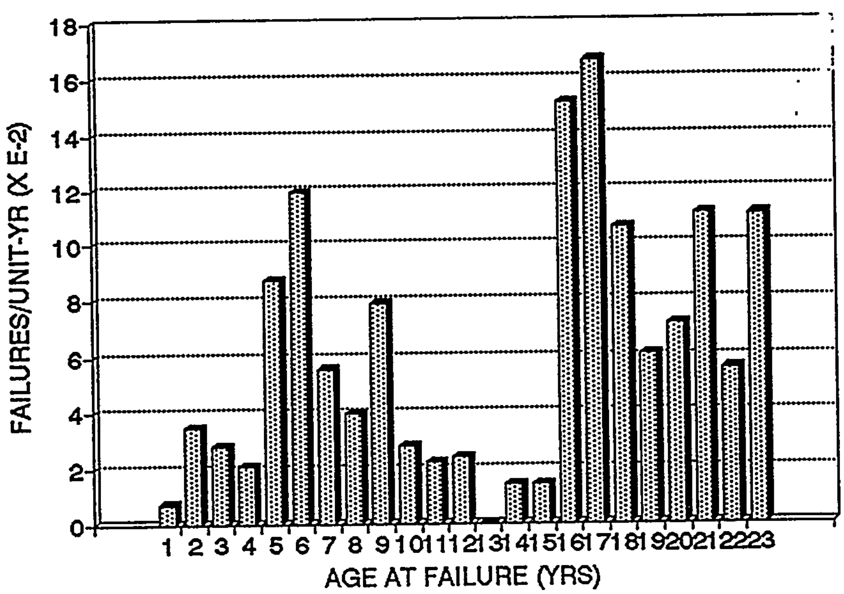


Figure 6.19 Normalized failure frequency of the containment isolation valve operators as a function of age at failure

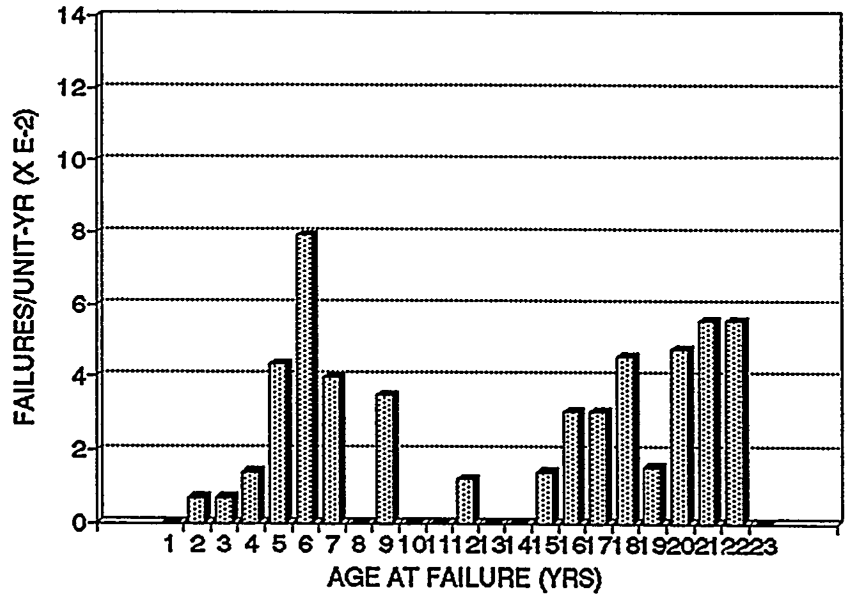


Figure 6.20 Normalized failure frequency of the containment isolation AC electric valve operator as a function of age at failure

Failure Mode

The major failure modes for the AC electric valve operators are "fail to close" (38%) and "fail to open" (38%); other modes are "fail to operate as required" (16%) and "found during I&S" (7%) (Figure 6.21). Comparing this result to the PWR data, the significant difference is in the "found during I&S" mode. For PWRs, about 26% of the failures belong to this mode, while BWR data showed only 7%. Also, the "fail to open" mode for PWRs was 22%, while that for BWR was 38%.

Failure Cause

There are no predominant failure causes for the AC electric valve operators in the BWR CI functions (Figure 6.22). The major ones are "out of mechanical adjustment" (20%), "normal/abnormal wear" (14%), "aging/cyclic fatigue" (9%), and "abnormal stress" (7%). Other significant causes are "insulation breakdown," "short/grounded," "circuit defective," "bad contacts," "lubrication problem," "dirty," "open circuit," and "corrosion."

However, the proximate cause analysis gives more useful information on which sub-components or parts fail most often, and cause component failures. Torque switches caused about 29% of the valve operator failures, and about 13% of the AC electric valve operators were due to failed motors. Figure 6.23 shows that the motor failures caused more of the failures after about 10 years, while failed torque switches were responsible for about 35% of the failures during the earlier years, 1-10 years; this later decreased to about 25% (11-23 years). Comparing these results with those from the PWR valve operators (Figure 5.29), the main differences are the limit switches and the motors. For the PWR AC electric valve operators, about 23% of the failures were due to the failed limit switches, but only 11% for BWRs. However, for PWRs, only 7% of the failures of AC electric valve operators were due to failed motors, but 13% for BWRs. These differences may be due to different pressure drops across the valves between BWRs and PWRs.

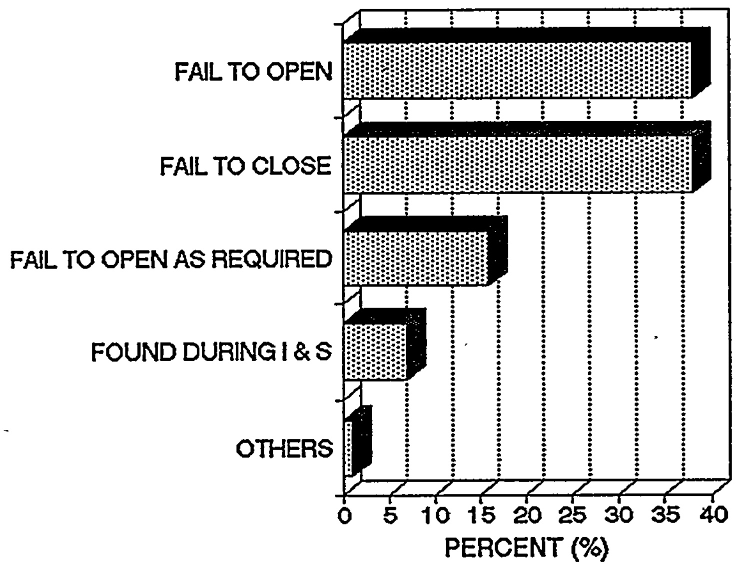


Figure 6.21 Failure modes for AC electric valve operators in the BWR CI function

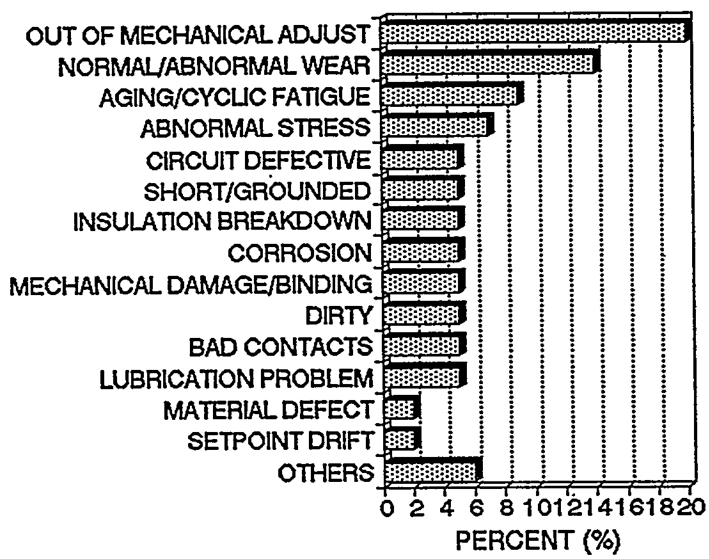


Figure 6.22 Failure causes for the AC electric valve operators in the BWR CI function

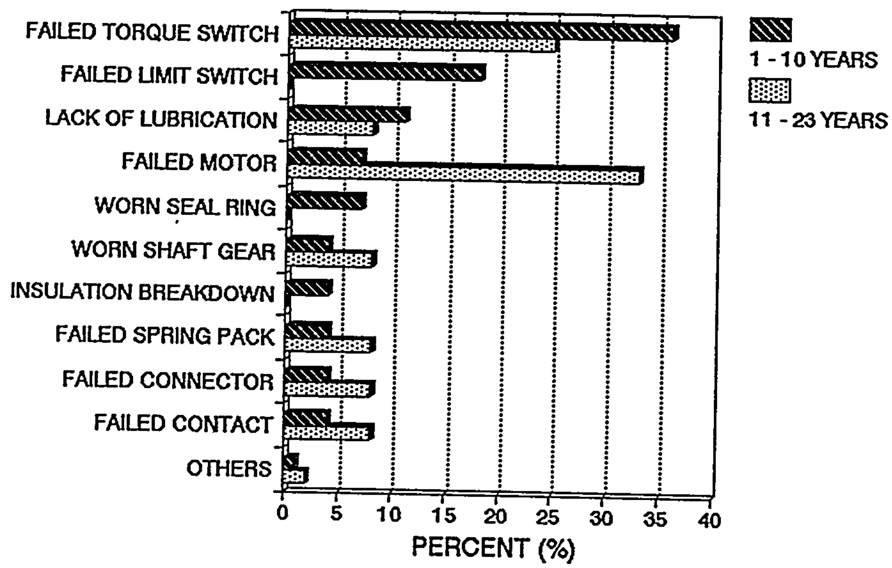


Figure 6.23 Proximate causes for AC electric valve operators in the BWR CI function

6.1.2.2 Pneumatic Valve Operators

Failure Frequency

Figure 6.24 gives the normalized failure frequency for the pneumatic valve operators in the BWR CI function. There is a dramatic increase in the failure frequency at 16 and 17 years, which is probably responsible for the peak at 16-17 years in Figure 6.19. When this curve is compared with that for the PWRs (Figure 5.25), a similar general trend is seen, except that there is a dip during 12 - 15 years for the BWRs. The causes for the difference are discussed in the following section.

Failure Mode

The major failure modes for the pneumatic valve operators are "fail to close" (40%), "fail to operate as required" (25%), and "fail to open" (19%) (Figure 6.25). This result differs from those from PWR pneumatic valve operators, where "fail to open" accounted for about 35% of the failures, and "fail to close" for about 26% (Figure 5.30).

Failure Causes

Figure 6.26 shows that "normal/abnormal wear" (27%), "mechanical damage/binding" (17%), "out of mechanical adjustment" (17%), "aging/cyclic fatigue" (12%), and "lubrication problem" (9%) are the major causes of failure for pneumatic valve operators in the BWR CI functions. However, the proximate cause analysis revealed that approximately 57% of them are due to problems with the solenoid valves, including failed solenoid coils. As shown in Figure 6.27, failed solenoid valves remain the main cause for the failures of pneumatic valves over years. Again, this finding differs from that for the PWR pneumatic valve operators, where failed diaphragms, failed limit switches, and air leaks due to failed air

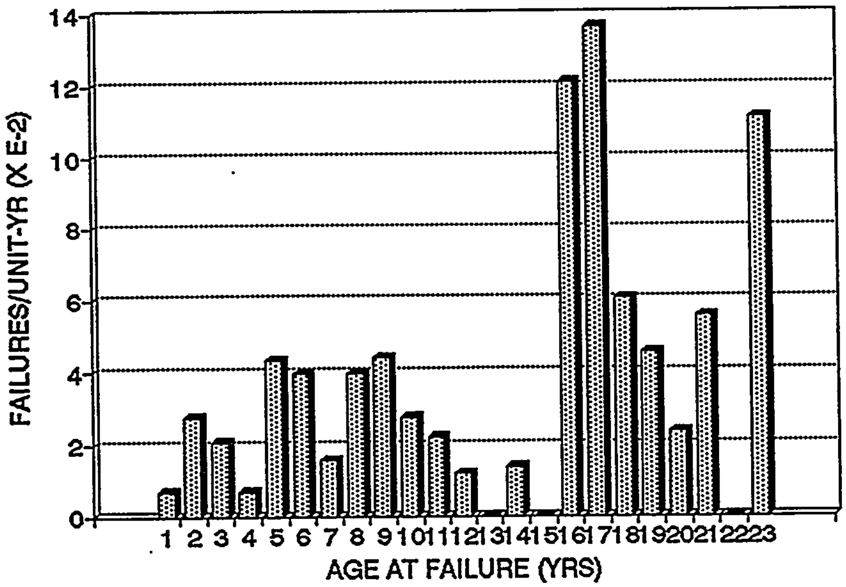


Figure 6.24 Normalized failure frequency of CI function pneumatic valve operators as a function of age at failure

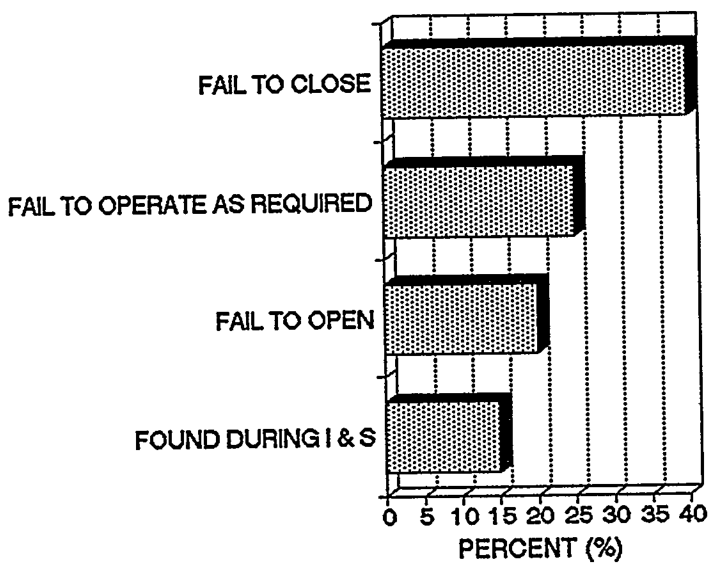


Figure 6.25 Failure modes for pneumatic valve operators in the BWR CI function

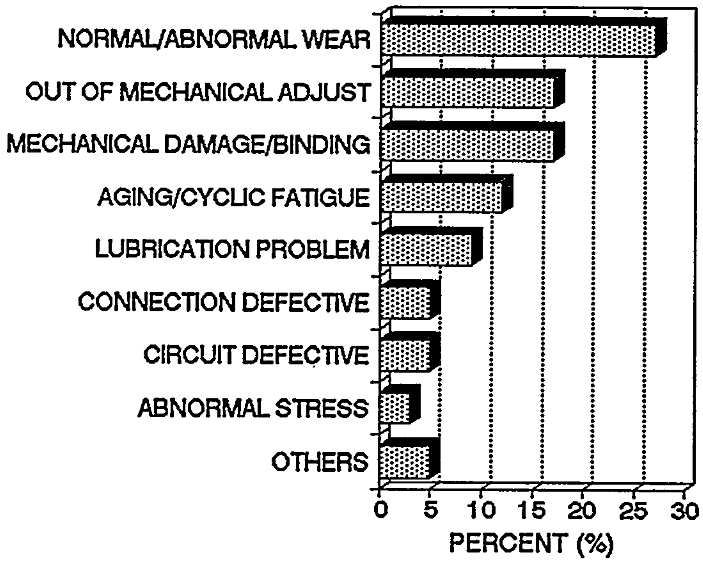


Figure 6.26 Failure causes for pneumatic valve operators in the BWR CI functions

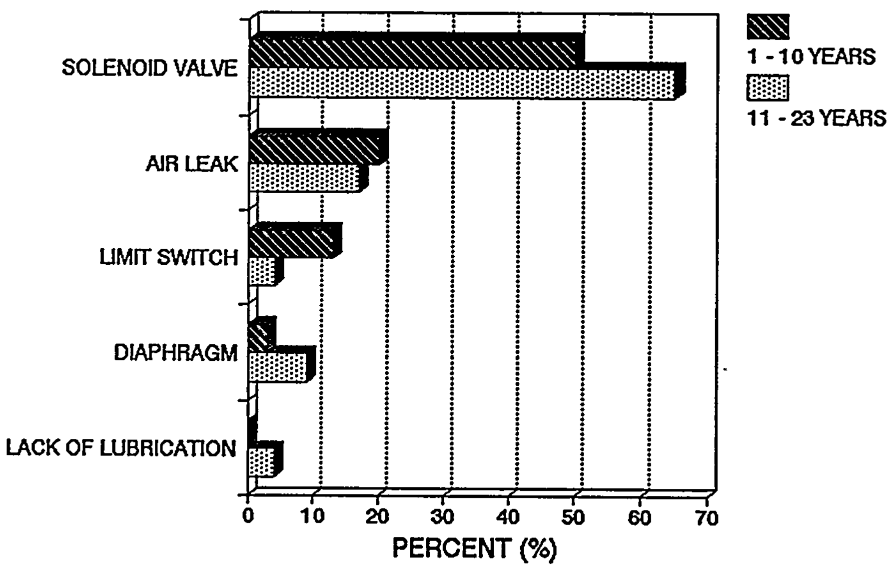


Figure 6.27 Proximate causes for pneumatic valve operators in the BWR CI functions

regulators and air lines generated as many failures as solenoid valves (Figure 5.32). Plotting the failures of the solenoid valves for both BWRs and PWRs together (Figure 6.28) shows that the failure frequencies track very well.

6.1.3 Bistables and Switches

There were 436 bistable and switch failures reported to NPRDS related to the containment isolation function; of these, 85% were aging-related. As discussed before, 83% of the bistable/switch failures were from the nuclear steam supply shutoff system; only these data were analyzed.

6.1.3.1 Bistables

Frequency of Failure

Figure 6.29 shows the normalized failure frequency for the bistables as a function of age at failure. Extra consideration was given to interpreting this curve. Previous NPAR research at BNL demonstrated that the same, or similar, models of the bistables/trip modules should be grouped together for aging research [Ref. 21]. Because electronic technology has rapidly advanced and changed over the last 30 years, and commercial nuclear power plants are of different vintages, the bistables of different groups of plants have varying levels of sophistication.

From the mid-1960s to mid-1970s, semiconductor technology replaced the older technologies. Transistors and diodes compacted the size of electronic modules and considerably enhanced the reliability of these modules. In mid-1970s, integrated circuit (IC) technology was used designing bistable cards, making previous technology obsolete. Since mid-1980s, microprocessor technology has been replacing the existing technology, and the state-of-the-art trip modules are all digital systems, monitored and controlled by computers. Thus, this plot does not show the effects of aging on failure frequency because

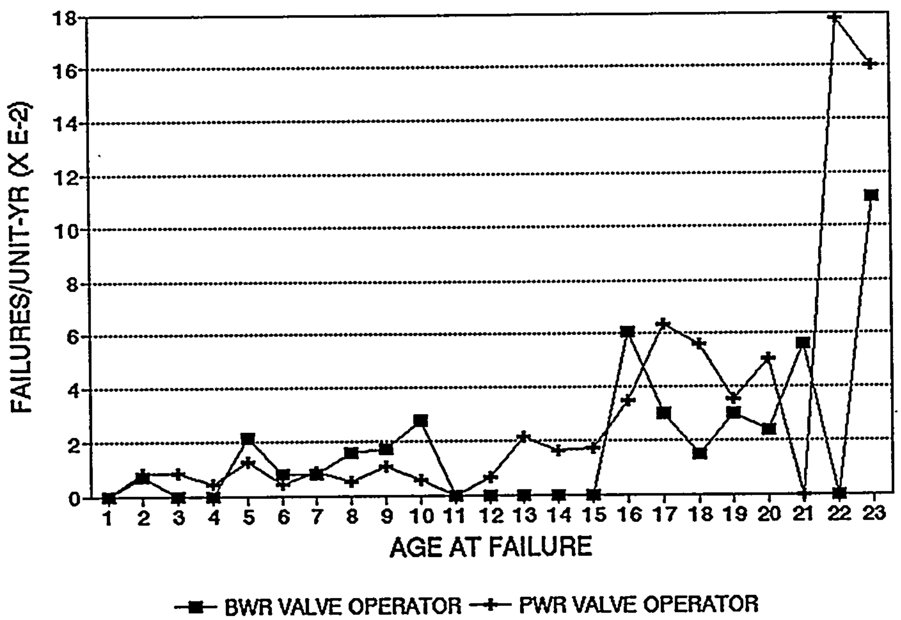


Figure 6.28 Normalized failure frequencies for solenoid valves in the pneumatic valve operators in the CI functions of BWRs and PWRs

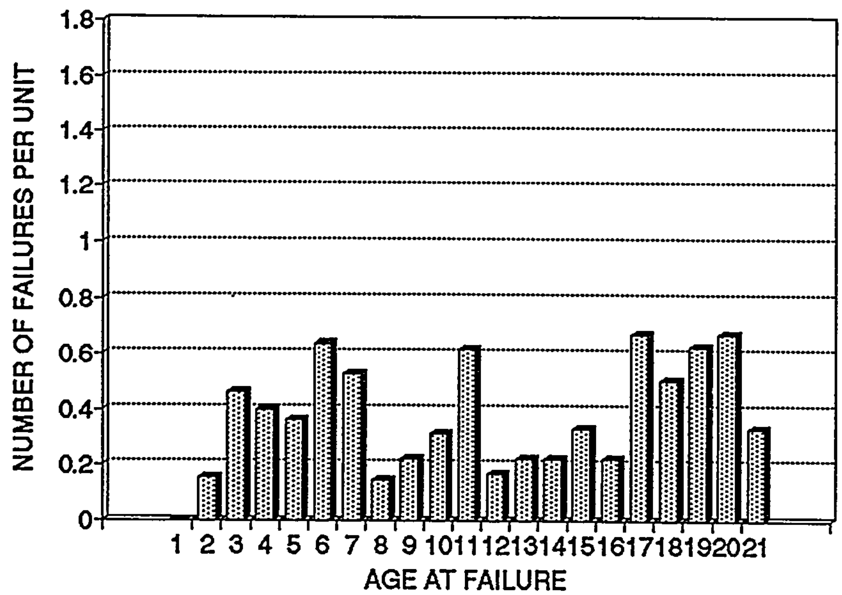


Figure 6.29 Normalized failure frequency of bistables in the BWR NSSS system as a function of age at failure

the bistables that provided the data are of several different vintages. However, this plot shows the general failure frequency for bistables, which contributed significantly to component failures in the BWR CI functions.

Failure Mode

The major failure modes for the bistables are "fail to change state upon demand" (21%), "low actuation" (16%), "spurious actuation" (15%), "high actuation" (13%), and "erratic actuation" (14%) (Figure 6.30).

Failure Cause

Circuit problems due to aging caused the most failures of the bistables (37%), followed by "aging/cyclic fatigue" (15%), "out of calibration" (14%), "setpoint drift" (11%), and "normal/abnormal wear" (10%). Other causes are "defective connection" (3%), "open circuit" (3%), "burned/burned out" (2%), and "abnormal stress" (2%) (Figure 6.31). The proximate cause analysis did not give any significant information because most of the reports did not state the proximate or root causes; it is a common practice to replace the whole card, and send it out for service. Again, we emphasize that the proximate or root causes are component-model specific, especially for electronic components, and a detailed study on several models is available [Ref. 21].

6.1.3.2 Switches

There were 197 aging-related failures of switches in the nuclear steam supply shutoff system, of which 36% were pressure switches, and 39% were temperature switches. Level switches contributed about 5% of the failures, and flow switches, 4%. In this section, only the first two are discussed.

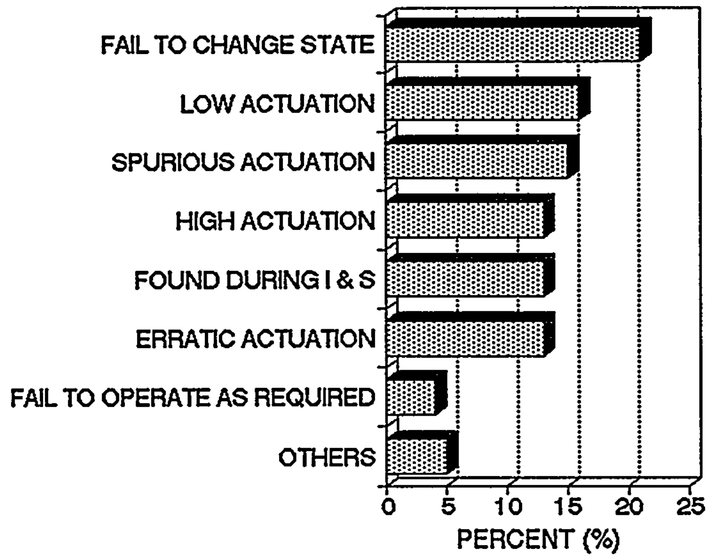


Figure 6.30 Failure modes for bistables in the BWR NSSS system

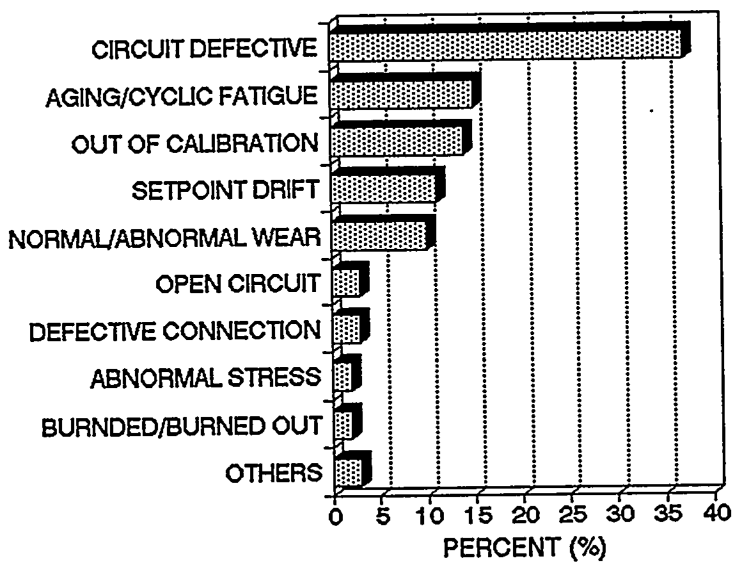


Figure 6.31 Failure causes for bistables in the BWR NSSS system

Effects of Failures of Switches on Plant Operation

Three switch failures resulted in reduced power operation; one was caused by the failure of a level switch, and the other two were caused by temperature switches.

In the first event, while the unit was at full power, a level switch for monitoring reactor low water level was out of calibration, resulting in a Technical Specification violation, so that power operation was reduced to repair it. In the second event, with the plant at full power, the differential temperature switch of the reactor water cleanup (RWCU) heat exchanger tripped high, indicating an open thermocouple, causing a RWCU isolation (Group 4 of containment and reactor vessel isolation control system). The failure placed the plant in a 2-hour limiting condition for operation (LCO) and forced the plant operators to reduce the power level. The thermocouple leads had broken off at the control room terminal board. The last incident may not be an aging-related failure, but it is worth noting because of its effects on plant operation. With the plant at 89% power, the temperature switch in the reactor building pipe chase failed high, causing the high-temperature annunciator in the reactor building general area to alarm. This switch provides a division one Group 5, 6, and 10 valve isolation. This also resulted in high area-temperature signals for the RCIC and RWCU systems. It is believed that the root cause was a deficiency in manufacturing design. The failure mode was to drift in and out of the alarm state.

One other switch failure resulted in unit off-line. While the plant was at 75% power, a routine round discovered a level switch for the reactor's low water level was out of calibration, which could have failed, giving a Group 1 isolation. This finding resulted in the unit being taken off-line for repair.

These descriptions show that switches are one of the components whose failures can affect the plant's operation significantly. Even though a failure of a single switch in the CI function cannot initiate containment isolation, it may result in entering a limiting condition for operation. Thus, bistables and switches should have a high priority in an aging management plan.

Failure Frequency

When the failures of pressure switches and temperature switches are combined and normalized, the frequency curve peaks at 6 years, after which the curve resembles a bathtub curve with increased failure frequency at older ages, 18-19 years (Figure 6.32). Plotting the failures of these two switches separately (Figure 6.33) demonstrates that pressure switches are affected by aging more than the temperature switches. The failure frequency for the pressure switches stay low until 16 years, after which it increases rather fast. The effects of aging on the failure frequency of temperature switches are less significant.

Failure Mode

The major failure modes for the switches are "high actuation" (30%), "fail to change state upon demand" (21%), "low actuation" (20%), and "erratic actuation" (11%) (Figure 6.34). "Spurious actuation," "fail to operate as required," and "found during I&S" each contributed 5% of the failure modes.

Failure Cause

The major causes of failure for the switches are "defective circuit" (28%), "setpoint drift" (28%), "aging/cyclic fatigue" (13%), and "normal/abnormal wear" (10%). Other significant causes are "out of

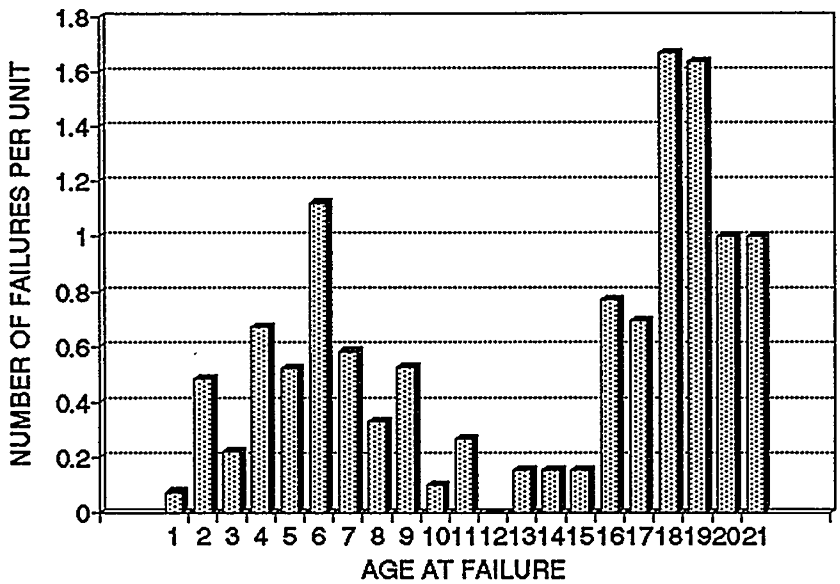


Figure 6.32 Normalized combined failure frequency for pressure and temperature switches in the BWR NSSS system as function of age at failure

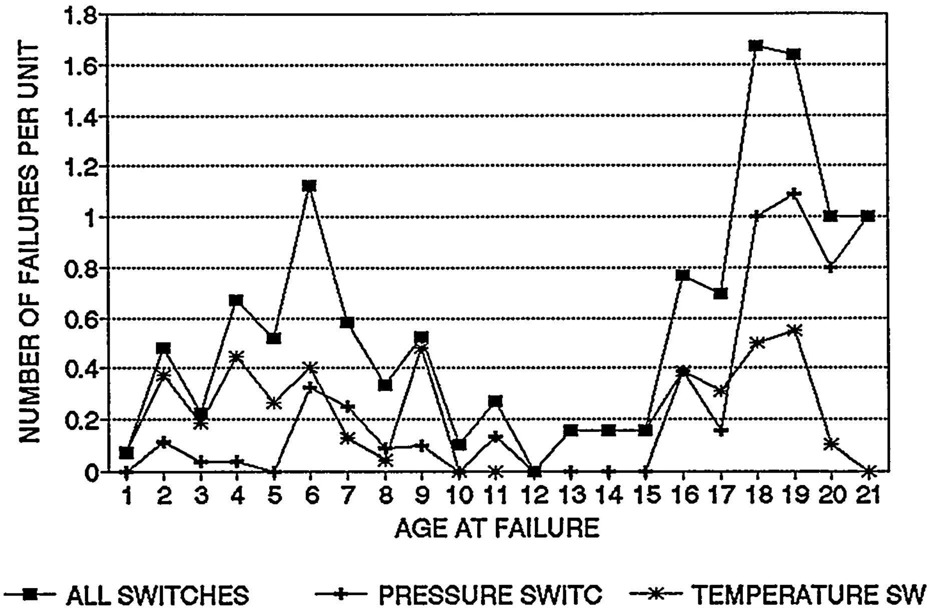


Figure 6.33 Normalized failure frequencies for pressure and temperature switches in the BWR NSSS system

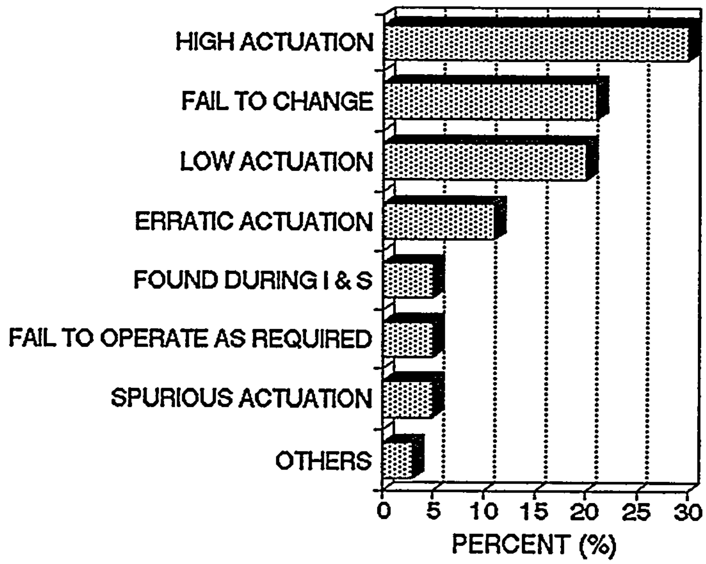


Figure 6.34 Failure modes for switches in the BWR NSSS system

calibration" (8%), "abnormal stress" (3%), and "dirty" (2%) (Figure 6.35). "Others" in this figure include "burned/burned out," "mechanical damage/binding," and "defective connection."

6.1.4 Radiation Detectors/Transmitters

There were 159 reported failures of radiation detectors and transmitters in the NSSS system, of which 63% were aging-related. Approximately 47% of these were failures of radiation detectors and 23% of pressure transmitters. The other failures were those of transmitters monitoring level (15%) and flow (14%). The failure frequency for the radiation detectors was normalized, and Figure 6.36 shows the frequency as a function of age at failure; the frequency is almost directly proportional to its age. Approximately 45% of the failed radiation detectors had Geiger-Muller elements for radiation detection. We did not conduct a proximate cause analysis for radiation detectors and transmitters because the NPRDS narratives had insufficient information on each type.

During the period studied, two transmitter failures resulted in one reactor trip and one reduced power operation. During the same period, one radiation detector failure resulted in a unit-off line.

With the unit at 100% power, a main steam line (MSL) high flow signal from channel B caused a full containment isolation. It is suspected that a spurious initiation of MSL high flow instruments or possible erratic output from flow transmitter was the possible cause.

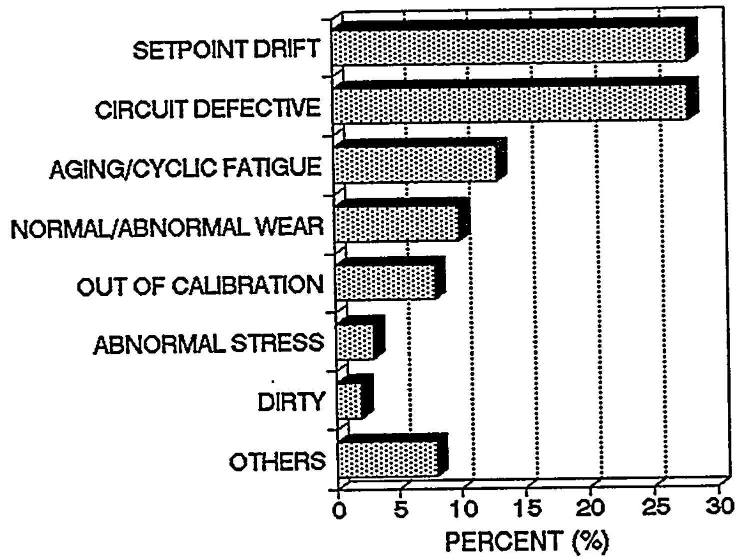


Figure 6.35 Failure causes for switches in the BWR NSSS system

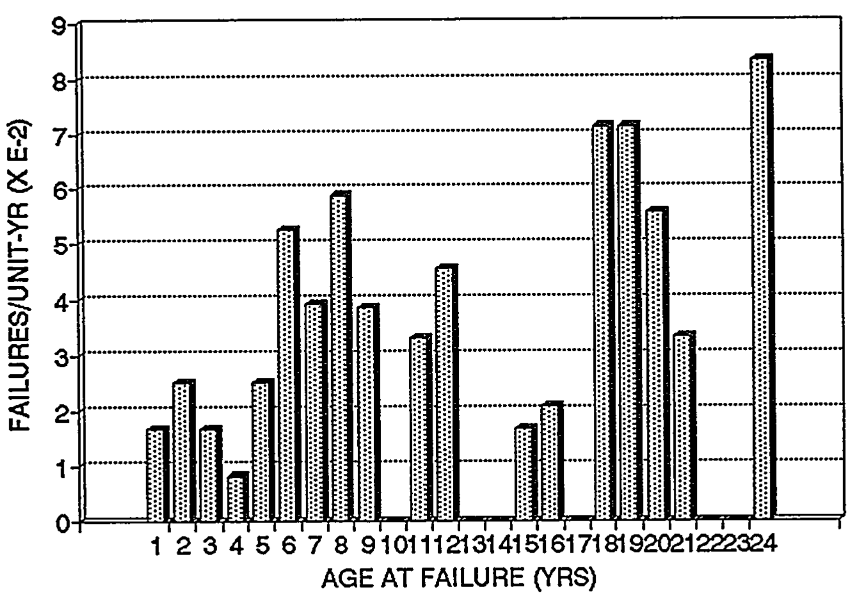


Figure 6.36 Normalized failure frequency for radiation detectors in the BWR NSSS system

In the second incident, with the unit at full power, the control room received an alarm on the main steam line flow differential pressure transmitter, indicating there was low steam pressure in the pipe. This resulted in a train loss, a degraded system, and reduced power. Failure of the differential pressure (DP) switch caused the alarm; its root cause was unknown.

While the plant was in the process of starting up, the radiation detector in the refueling floor vent duct spiked and caused a balance of plant isolation (Group 5). This failure put the plant in a limited condition for operation, and delayed the unit start-up. The failure was believed to be caused by dirt and paint chips in the detector.

As is the case of bistables and switches, these descriptions show that radiation detectors and transmitters also are the components whose failures can affect plant operation significantly. Thus, they also should have a high priority in an aging management plan.

6.1.5 Penetrations

There were 134 reported failures of penetrations, of which 78% were aging-related; all were those of penetrations for personnel access. As shown in Figure 6.37, most of the failures occurred before 12 years.

Effects of Penetration Failures on Plant Operation

No reported aging-related failures of penetration had any significant effect on the plant. However, two human error-related failures that resulted in unit off-line are described below to show the possible effects of failure of a penetration on plant operation.

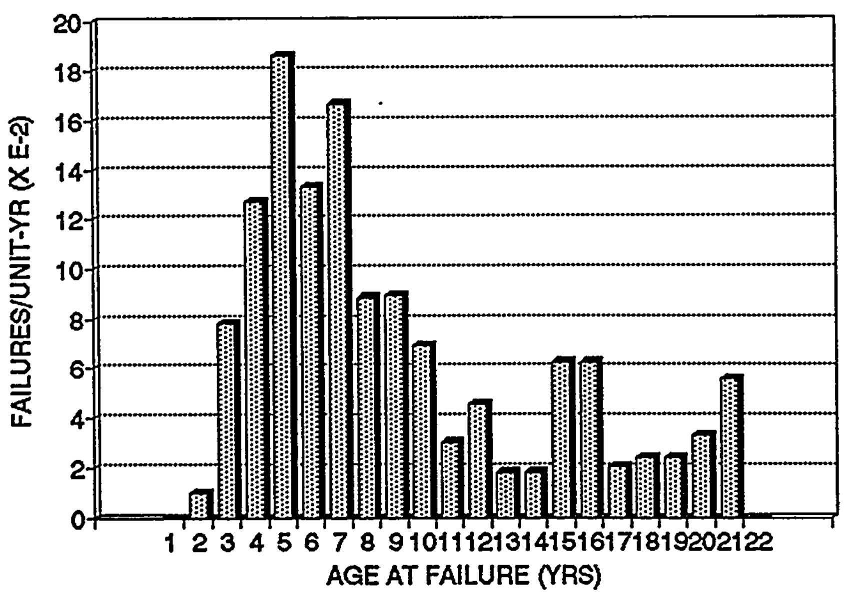


Figure 6.37 Normalized failure frequency for penetrations in the BWR CI functions

While at 21% power, during an LLRT on the primary containment's personnel access hatch, a sudden depressurization of the test pressure was experienced. The reactor was subsequently shutdown and the primary containment de-inerted to inspect the gasket seal of this airlock. A six-inch longitudinal tear in the inner door seal was found to be the cause; previous maintenance was thought to be responsible for the problem.

In the second case, during preparations for reactor start-up following a forced outage, inspection revealed that the weld of the nipple for the pressure warning device had broken, and the containment isolation function was lost; the start-up was delayed until the hatch was repaired. It is believed that the holding screw was over-torqued causing the weld to break.

Failure Mode

There are only two failure modes for penetrations, "loss of pressure boundary function" and "found during testing, surveillance, inspection, or maintenance." NPRDS describes the former as the penetration losing it ability to isolate the containment from the external environment. For personnel access hatches, an air lock that fails to seal belongs to this category. An airlock that fails to open yet maintains the containment's integrity, a malfunctioning airlock door interlock that does not cause the door to leak, and defects identified that do not cause the penetration to leak are reported as "found during testing, surveillance, inspection, and maintenance (TSI&M)." Incipient failures also are included in this category. Forty seven failures reported in the first category, and 58 in the second.

Failure Cause

The major failure causes for the penetrations are "normal/abnormal wear" (46%), "mechanical damage/binding" (19%), "aging/cyclic fatigue" (10%), and "out of mechanical adjustment" (10%) (Figure 6.38). The proximate cause analysis shows that seal failures (ruptured/damaged/worn) accounted for about 31% of the airlock failures, and failed ball valves including stem problems caused about 20% (Figure 6.39). Other minor proximate causes are "failed linkage" (8%), "failed solenoid valve" (6%), "worn bearing" (5%), "failed snap ring" (4%), "broken cable" (4%), "dirt accumulation" (4%), "failed cam" (3%), "worn gasket" (3%), and "failed handwheel" (2%). "Others" in this figure include "failed helm joint," "failed flex hose," and "failed locknut/bolts."

6.2 Review of LERs for the BWR Containment Isolation Functions

Licensee Event Reports (LERs) document failures which primarily affect plant operation, and may not always be included in the NPRDS database. Therefore, to completely understand aging effects on CI functions, it is important to review the LERs in addition to the NPRDS database.

For the six years from 1988 to 1993, 574 LERs were written which documented operational problems and component failures for the Containment Isolation Function at 36 BWRs. Each LER narrative was reviewed to determine if the failure was aging-related; the vast majority (68%) were not (Figure 6.40). An additional 12% had insufficient information to determine if the problem was aging-related or not, and, for this study, were classified as unknown. The remaining 19% of the LERs were classified as aging (27 LERs) or potentially aging (82 LERs). Therefore, for this aging assessment, only the 109 LERs categorized as aging-related failures were considered. Though human-related failures are not aging-related, it is important to note that over 90% of the LERs classified as non-aging fell into this

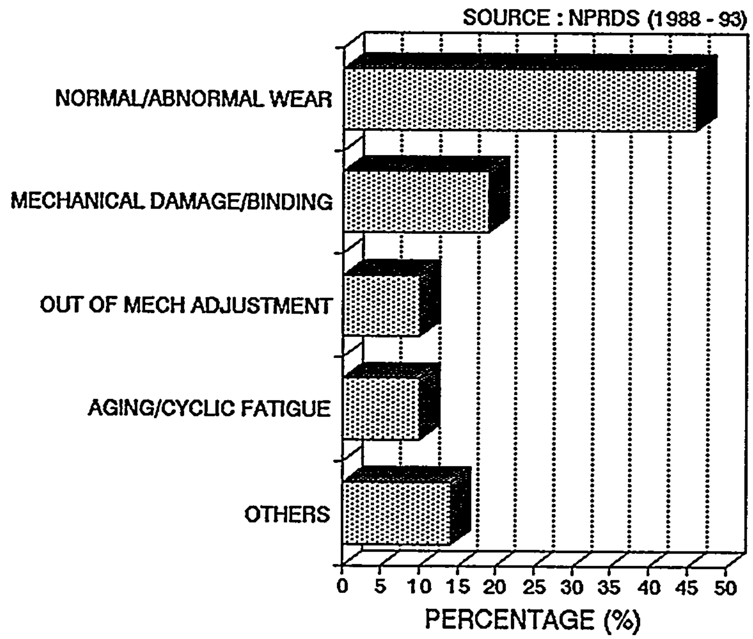


Figure 6.38 Failure causes for BWR personnel access penetrations

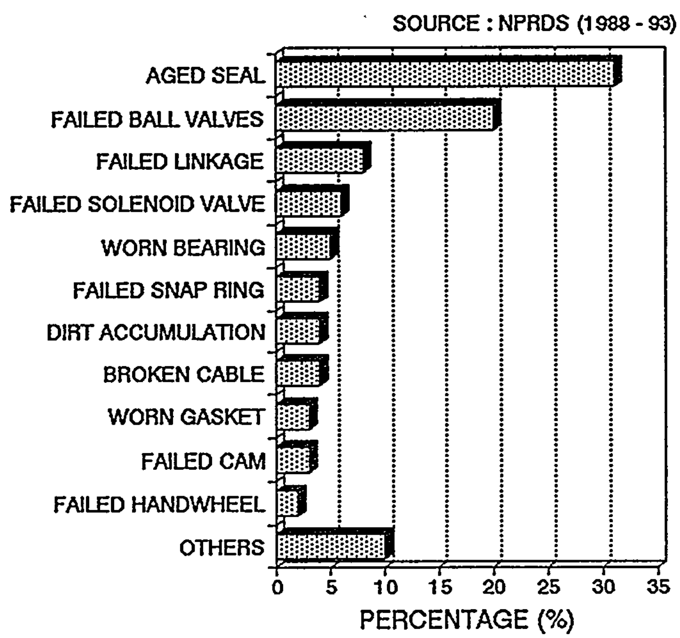


Figure 6.39 Proximate causes for BWR personnel access penetrations

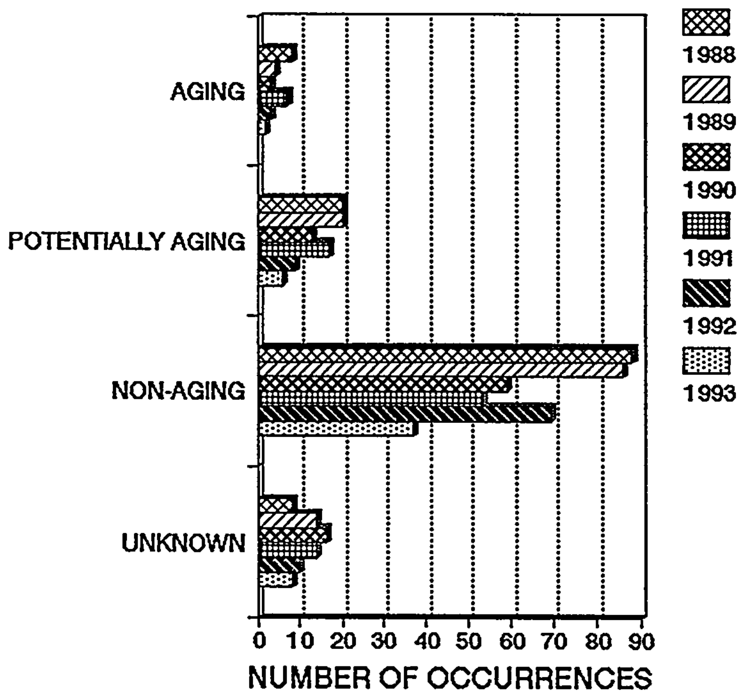


Figure 6.40 Percentage of BWR CI function aging related LERs (1988-1993)

category. This highlights the susceptibility of the CI function to failures caused by human errors, and furthermore, they may accelerate aging degradation. For example, spurious, erroneous actuations of components may contribute to the aging of these components over time.

Nine different components were reported as CI function components, but other than valves, relays, and penetrations, each of the remaining six components were documented on 6 or fewer LERs. Hence, though aging-related, these infrequent failures were considered as isolated occurrences, and we did not evaluate them further. Because the CI function is primarily comprised of valves, it was not surprising that they were the most frequently failed components (Figure 6.41). Seventy one (65%) LERs documenting valve failures were written for this period, followed by relays and penetrations.

The specific causes of failures for valves, relays, and penetrations are shown in Figures 6.42 and 6.43. For both valves and penetrations, our review of the LERs found that a significant fraction contained none or insufficient information to determine the root cause of failure. While this may seem insignificant for an individual LER, when it involves the majority of a system's LERs, it highlights the importance of a detailed root cause analysis of failures for components. Of the identifiable causes, packing wear and internal degradation (corrosion and mechanical wear) were the most common ones for the valves. For the penetrations, degradation and failures of the seals were the primary causes. Failures in the electronic control circuits, contact failures and coil failures were the major causes for the relay failures.

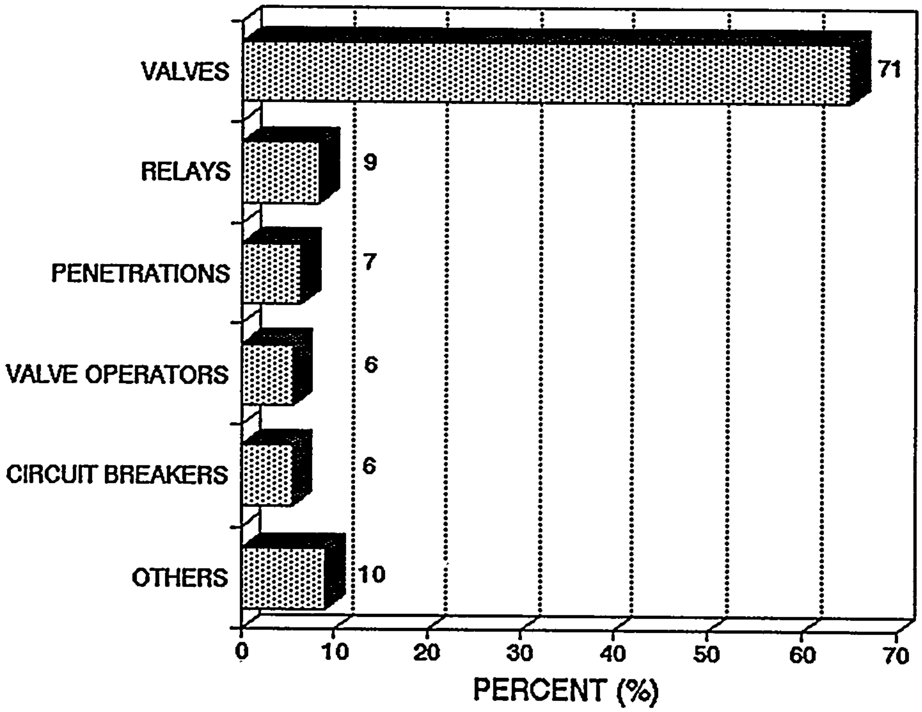


Figure 6.41 Failed components in BWR CI function(1988-1993 LERs)

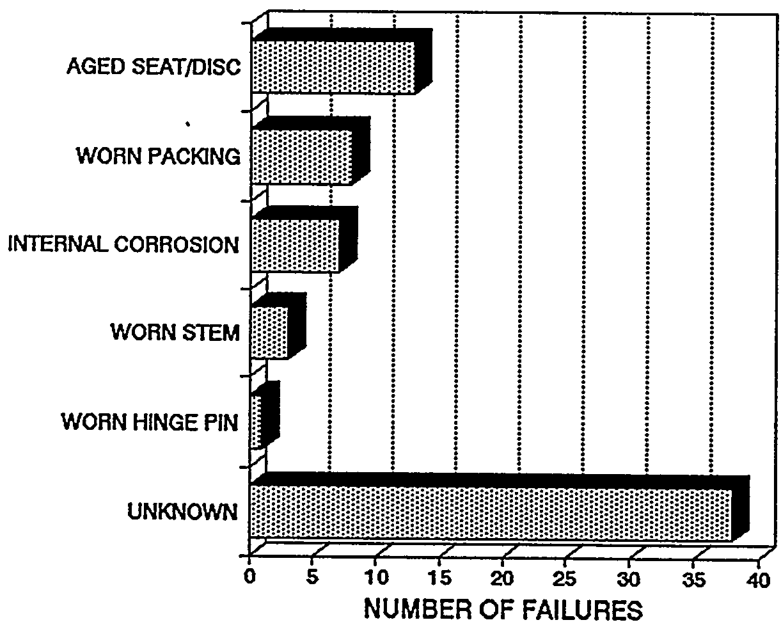


Figure 6.42 BWR CI valve failure causes (1988-1993 LERs)

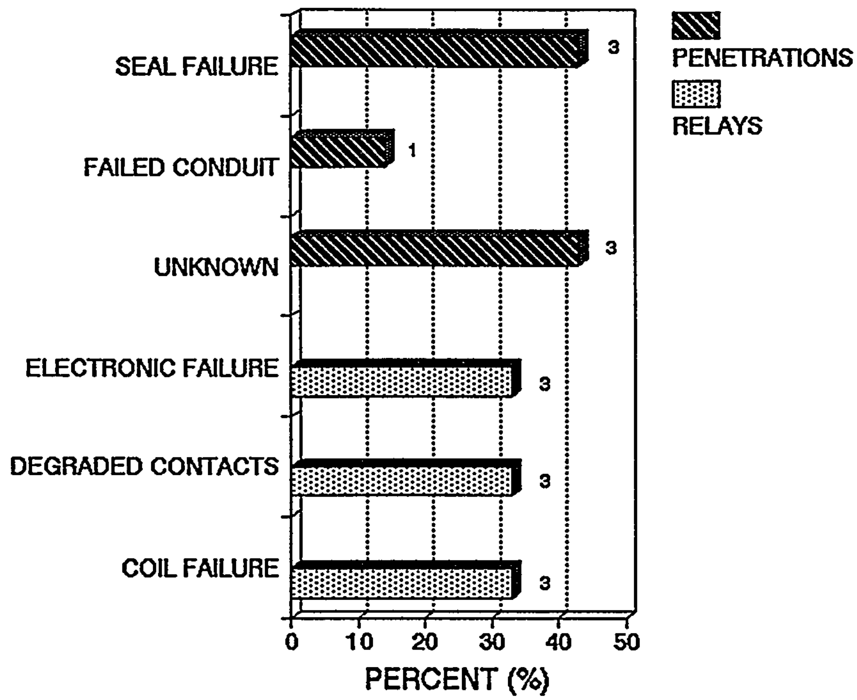


Figure 6.43 BWR CI penetrations and relay failure causes (1988-1993 LERs)

Figure 6.44 shows that the majority of these component failures were detected during testing. Plant testing may consist of Technical Specification, Inservice, or other types of tests. Since this system is primarily a standby system, these components are not used during normal operation, and are exercised during the scheduled tests (e.g., quarterly, refueling, cold shutdowns, or semi-annual). Valve and airlock seal leakages also were detected during Appendix J leak rate testing, performed once every 2 years. These tests also revealed relay failures, but they were found to have a audio-visual alarm associated with them if the component was inoperable (i.e., valve failure to close). Because most of the reported failures were detected during these surveillance-type tests, we can make some conclusions about the effectiveness of these programs.

The primary failure modes for the valves were internal and external leaks (Figure 6.45). The main failure modes for the relays and the penetrations are failure to operate as required and loss of pressure boundary, respectively. Operational failures (e.g., failure to operate upon demand) were much less frequent; however, this does not indicate that leaks are not significant. The primary purpose of CI components is to prevent leakage outside the containment in the event of an accident. Leakage which exceeds Technical Specification limits is a significant failure mode. Most of the reported failures were correctable by repairing the component (Figure 6.46), except the failed relays, which were replaced.

Containment isolation valves typically are installed in series, so that one failure cannot lead to a potential release. In these instances, failure of one of the two valves would represent no significant effects on the plant (Figure 6.47). However, on a system level (Figure 6.48), this would, as a minimum, result in a loss of redundancy. In locations where the safety analysis assumed operational series valves, a failure of one would result in a degraded or loss of system function.

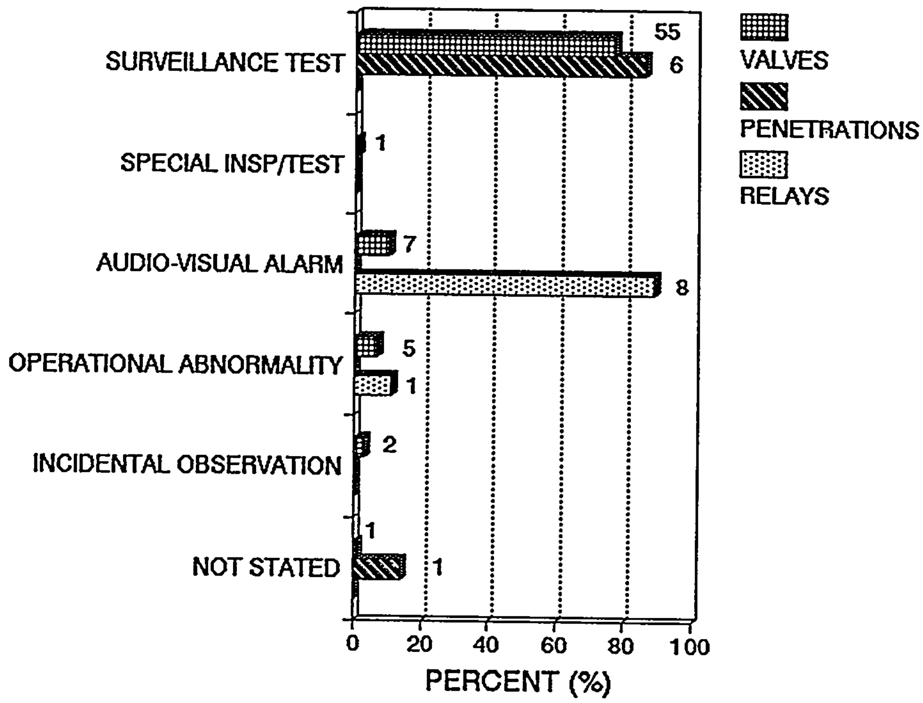


Figure 6.44 Methods of detecting BWR CI failures (1988-1993 LERs)

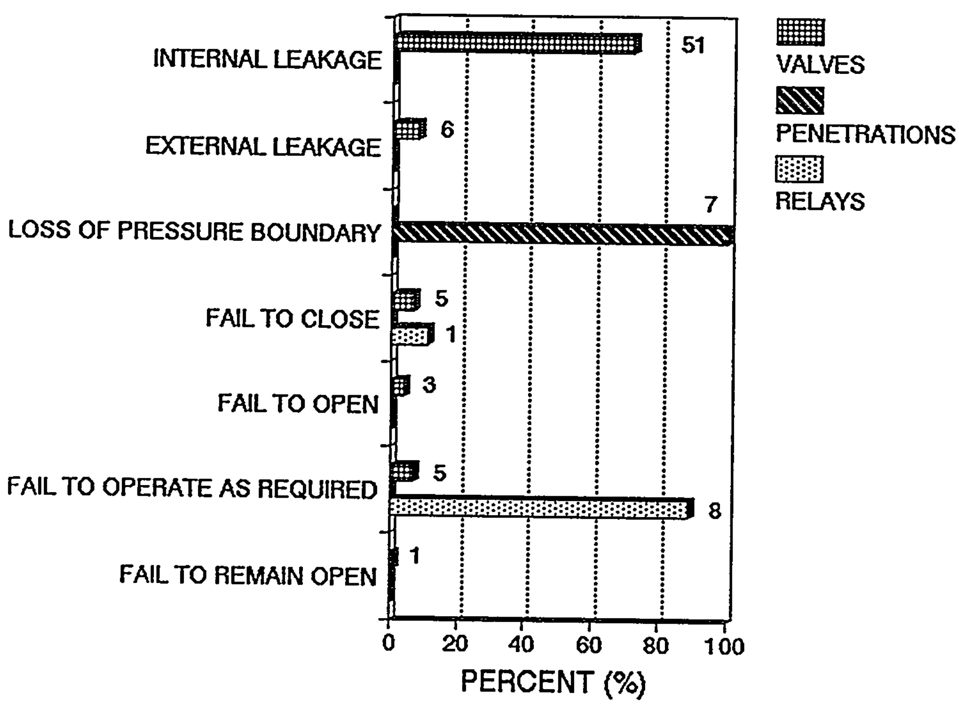


Figure 6.45 BWR CI component failure modes (1988-1993 LERs)

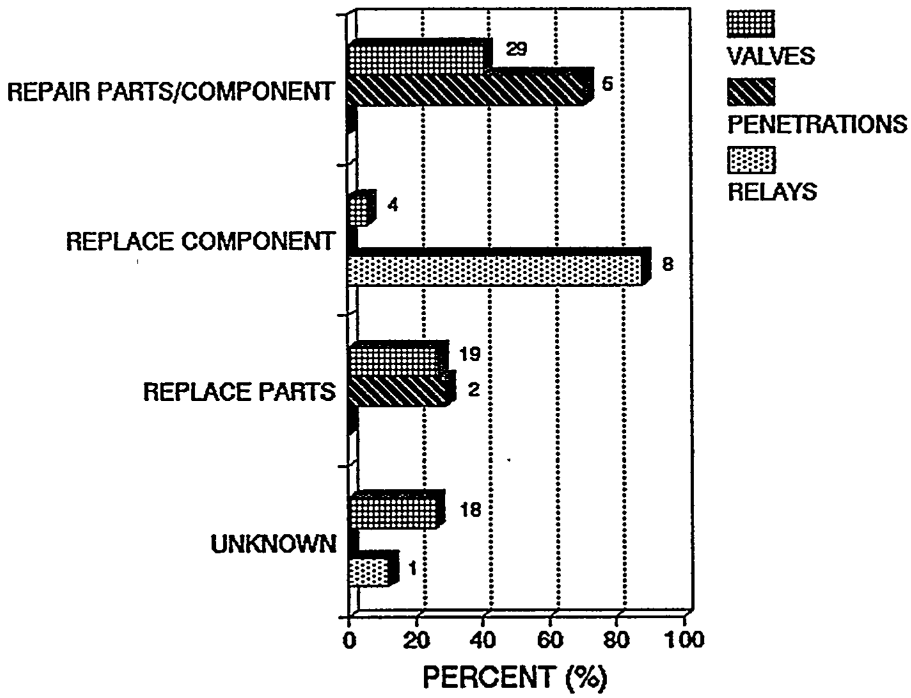


Figure 6.46 BWR CI component corrective actions (1988-1993 LERs)

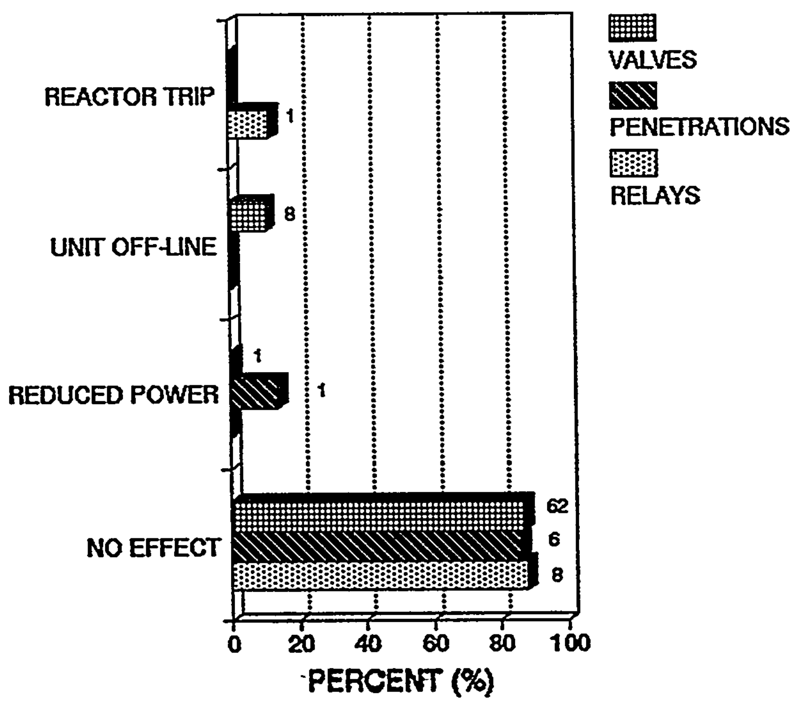


Figure 6.47 BWR plant effects from CI failures (1988-1993 LERs)

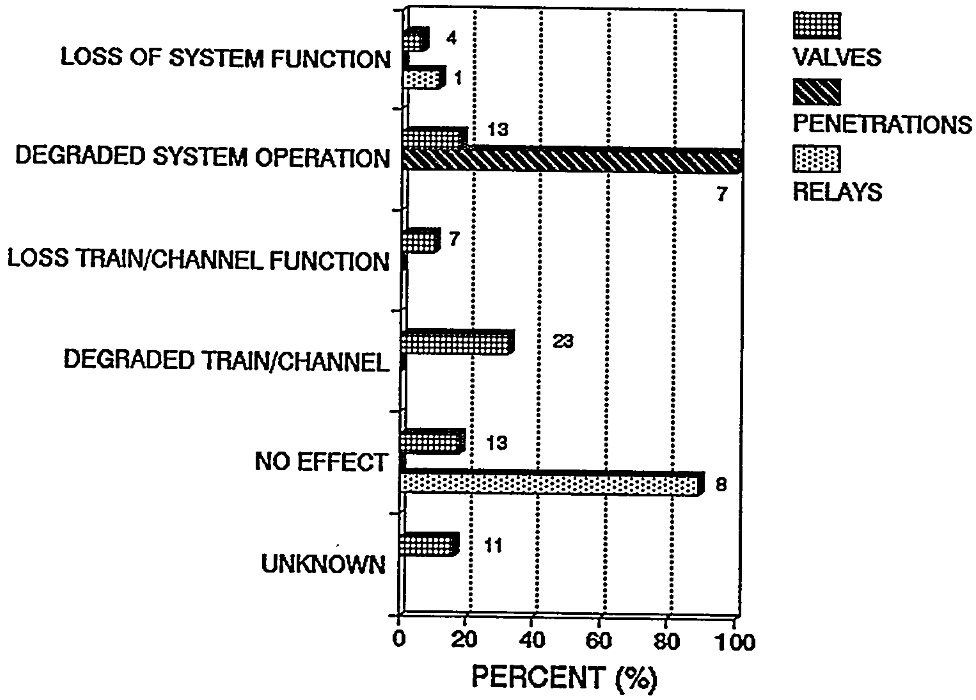


Figure 6.48 BWR system effects from CI component failures (1988-1993 LERs)

6.3 Summary of BWR Operating Experience

During the period studied, 1988 - 1993, most of the failures of the components in the BWR containment isolation function were detected either during the surveillance testing or immediately after the occurrences. However, some failures affected the plant's operations, and as a result, affected its cost. This study showed that the failure frequency is strongly affected by aging, which indicates that aging-related failures will affect the plant operations more as the systems and components get older. Thus, in the long run, the initial investment for proper aging management will lower the cost of operation, and at the same time, enhance safety.

The major containment isolation components with failures reported most often (not the failure rate) are valves (35%), bistables/switches (30%), valve operators (14%), and radiation detectors/transmitters (12%). About 86% of the valve failures were aging-related failures; 85% and 67% were aging-related failures for bistables/switches and valve operators, respectively.

Unlike PWRs, gate valves for BWRs are the major contributor to the valve failures in the CI function (41%), followed by globe valves (27%) and check valves (23%). For PWRs, 41% of the failures were those of globe valves. The failure frequency of valves peaks at 6 years, then drops low until 15 years, after which it begins to increase. This curve reflects the typical aging effects on the failure frequency of BWR water valves.

Gate valves show high failures at 4 years, after which the frequency decreases continuously until 11 years, when it begins to increase. The major failure causes are aged seats/discs, corrosion product/dirt buildup, and worn packing, which result in internal leaks, failure to close, and external leaks, respectively.

The failure frequency of globe valves increases gradually during the first 6 years, after which it decreases to a very low value at 11 years. There are two more peaks at 15 and 21 years, both of which are caused by corrosion product/dirt buildup. When the globe valves are less than 10 years old, aged seats/discs, worn packing, and corrosion product buildup are the major failure causes, which result in internal leaks, external leaks, and failure to close, respectively. After 11 years, corrosion product/dirt buildup predominates, causing about 63% of the failures.

The failure frequency of check valves increases continuously with a peak at 6 years of age, when it begins to fall to a very low value at 11 years. Thereafter, the frequency increases gradually without any distinct peaks, and is caused by gradual wear and corrosion of discs and seats. For check valves, internal leak is the main failure mode accounting more than 50% of the failures. During the first 10 years, corrosion product/dirt buildup caused the most failures of check valves (41%), followed by aged seats/discs (33%). The failures of older check valves are mainly due to the latter (54%).

The types of valve operators which the failures are most often reported to NPRDS are pneumatic (49%) and AC electric (34%). About 66% of the valve operator failures are aging-related. There is a rapid increase of failures of AC electric valve operator during the first 6 years when they peak. During the first 10 years, the failures of torque switches and limit switches caused 36% and 18% of the AC electric valve operator failures, respectively. After the peak, the failure frequency decreases to a very low value, but around 15 years of age, it begins to increase, mainly due to the increase of valve motor failures.

For pneumatic valve operators, the failure frequency stays almost constant during the first 10 years, and there is a dramatic increase at the age of 16 years. For both 1 - 10 years and 11 - 20 years, failure of the solenoid valve is the main cause for the pneumatic valve operator failures contributing 50% and 65%, respectively. The failure frequency of the solenoid valves stay almost constant for about 15 years, after which there is a large increase. This trend is identical for the pneumatic valve operators for both BWRs and PWRs.

Switch failures increase continuously during the first 6 years, after which the failure frequency resembles a typical bath-tub curve. Pressure switches and temperature switches are the ones whose failures are most often reported, with the former showing more significant aging effects on failure frequency than the latter.

The failure frequency of a radiation detectors is almost directly proportional to their age. Approximately 45% of the failed radiation detectors had Geiger-Muller elements for radiation detections. Most of the failures of penetrations for personnel access (airlocks) occurred before 12 years. Seal failures (31%) and failed ball valves (20%) were the major failure causes for the airlocks.

There were 109 LERs caused by aging-related failures of the components in the containment isolation function; about 65% of these are due to failures of valves, 8% due to relays, and 6% due to penetrations. The failures of valve operators and circuit breakers resulted in 6% of the LERs each. Unlike the information reported to NPRDS, most of the LERs did not contain failure causes, but for those that showed the causes for valve failures (45% of the aging-related LERs), aged seat/disc (41%), worn packing (25%), and corrosion product buildup (22%) were the major ones. These data are consistent with the information from the NPRDS analyses.

7. ANALYSIS OF PLANT OPERATING EXPERIENCE

The effect of aging on the containment isolation components was obtained from reviewing the operating data contained in the NPRDS and LER databases (Sections 5 and 6). These databases encompass component failures and degradation which affected plant operation, as well as those discovered during regular scheduled maintenance. Another valuable source of aging-related information can be obtained from reviewing plant-specific component maintenance records. In addition to failures reported to the databases, additional data from regularly scheduled preventive maintenance and performance testing can be gathered.

To supplement the insights obtained from the database review, a BWR-6 plant (Mark III type containment) was visited to obtain additional information on containment isolation ISM&M and other insights from plant personnel. This plant (Plant A) has been in service for eight years. The following sections present the results of the review of the plant's design, surveillance, and maintenance records for the containment isolation valves.

7.1 Review of Plant A Containment Isolation Design and Surveillance

Plant A has 79 containment penetrations, consisting of 248 containment isolation valves. Since containment isolation is not provided by one dedicated system, these valves are part of several systems (e.g., RHR, RCIC, LPCI,). Six different types of valves (Figure 7.1) are used to isolate these penetrations, depending upon their specific location and application. Globe and gate valves are the most frequently used types. The containment isolation valves are positioned in response to automatic or manual signals by five actuator types (Figure 7.2). Electric motors are the most frequently used in this Plant.

In the event of power loss, the majority of the motor-operated valves remain as-is. Air-operated valves close upon loss of air. Both types have redundant sets of position indication lights, and are located in the control room, one at the valve control switch, and the other on the isolation status panel. Position is controlled both by locking devices and administrative controls.

The majority of these valves are located outside the containment (75%), with the normal operating position equally divided between open and closed. Most containment isolation valves receive Type C tests conducted in accordance with 10 CFR50, Appendix J. Airlocks and penetrations receive the Type B test (also in accordance with 10 CFR50, Appendix J) carried out at the containment's design-basis accident pressure. Satisfactory completion of either assures that leakage through the systems and components which penetrate the primary containment do not exceed the allowable leakage rates, as specified in the plant's Technical Specifications. Additionally, the successful completion of these leak tests assures that the valves are adequately maintained.

Type B and C leak rate tests are performed by local pressurization using either a flow-meter, or by monitoring pressure decay over time. The test pressure is applied in the same direction as that which the valve would be required to fulfill its safety function. The frequency of these tests is specified in the plant's Technical Specifications. Typically, the plant operator performs these tests during shutdowns or refueling outages.

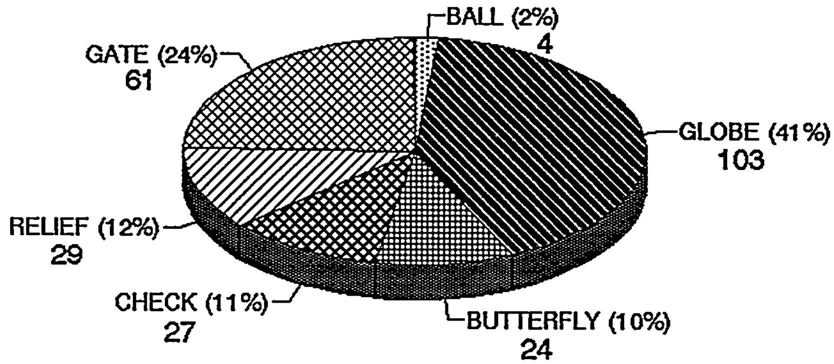


Figure 7.1 Containment isolation valve types-Plant A

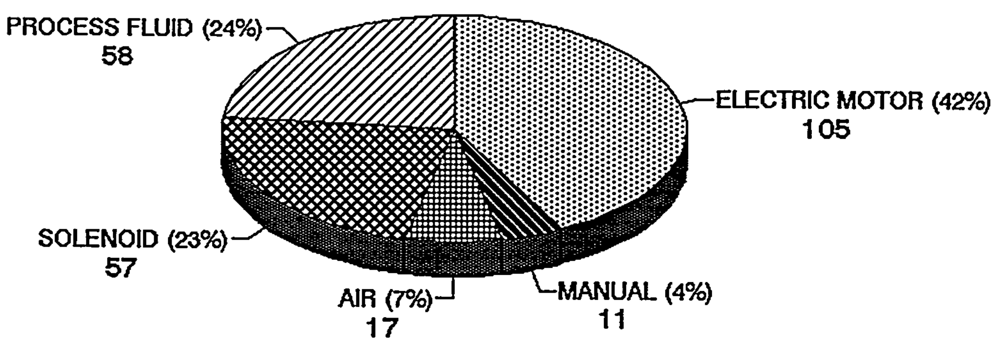


Figure 7.2 Containment isolation valve actuator types-Plant A

7.2 Review of Plant A Containment Isolation Maintenance

For the 10-year period between 1984 to 1994, Plant A generated 613 maintenance work orders for the containment isolation valves (not including airlocks). Each individual work order was reviewed and tabulated according to component, and maintenance category (corrective or preventive maintenance and performance testing). For the records documenting corrective maintenance, additional information was tabulated, including the component which failed, failure mode, and whether the failure was aging-related. At Plant A, work orders are generated for both preventive and corrective maintenance, as well as for the required testing (e.g., leak rate testing).

At this Plant, the effort is approximately evenly divided between corrective and preventive maintenance and testing (Figure 7.3). Preventive maintenance includes tasks such as lubrication, and replacing gaskets and seals. The effectiveness of a maintenance program may be assessed by comparing the time spent for corrective and preventive maintenance. If most of the time and effort is spent on failures, it may be concluded that insufficient time is spent on performance testing and preventive maintenance. For components which are normally not operated during normal operation, indications of degradation and failure during preventive maintenance and performance testing are the only regular means to ensure the component can perform when required.

The majority of the work orders were generated for valve operators (61%), as opposed to valves (33%) (Figure 7.4); only 6% were for relief valves. This trend correlates well with the distribution of valve actuator types for this Plant, and the emphasis placed on MOV operability and inspection by the NRC (i.e., Generic Letter 89-10).

The review of work orders for preventive maintenance and performance testing concluded that aging was not a factor for performing preventive maintenance. However, the opposite was found with the corrective maintenance orders, where the majority were in response to aging or potentially aging events (Figure 7.5). Based upon the limited descriptions in these work orders, a significant fraction were classified as potentially aging. Information on the root cause of failure was not included with the maintenance listing, so based upon the maintenance undertaken, engineering judgement dictated that these were similar to failures documented as aging-related in the operating event database.

In addition to trending operating parameters (i.e., stroke time) for signs of decreased performance, plant operators should be aware of the particular sub-components which account for most of the failures. Hence, increased attention can be given to these components, and less to other, more reliable components. For the containment isolation valves, the most frequently failed sub-component was packing, followed by seal and binding, and stem degradation (Figure 7.6). The relief valves did not experience any significant failures requiring extensive corrective maintenance (Figure 7.7). The relatively infrequent failures did not indicate any particular sub-component which was commonly failing; therefore, these events were considered to be isolated occurrences for this review. For valve operators, degraded switches were the most frequently failed component (Figure 7.8). Approximately 40% of the work orders required corrective maintenance on the limit, torque, or reed switches. Sufficient attention should be placed on identifying the root cause of the reported failures; for a significant number of valves, valve operators, and relief valves, this was not identified. Numerous examples of valve operator failures were reported, particularly for switches, which were replaced, but the root cause was not given.

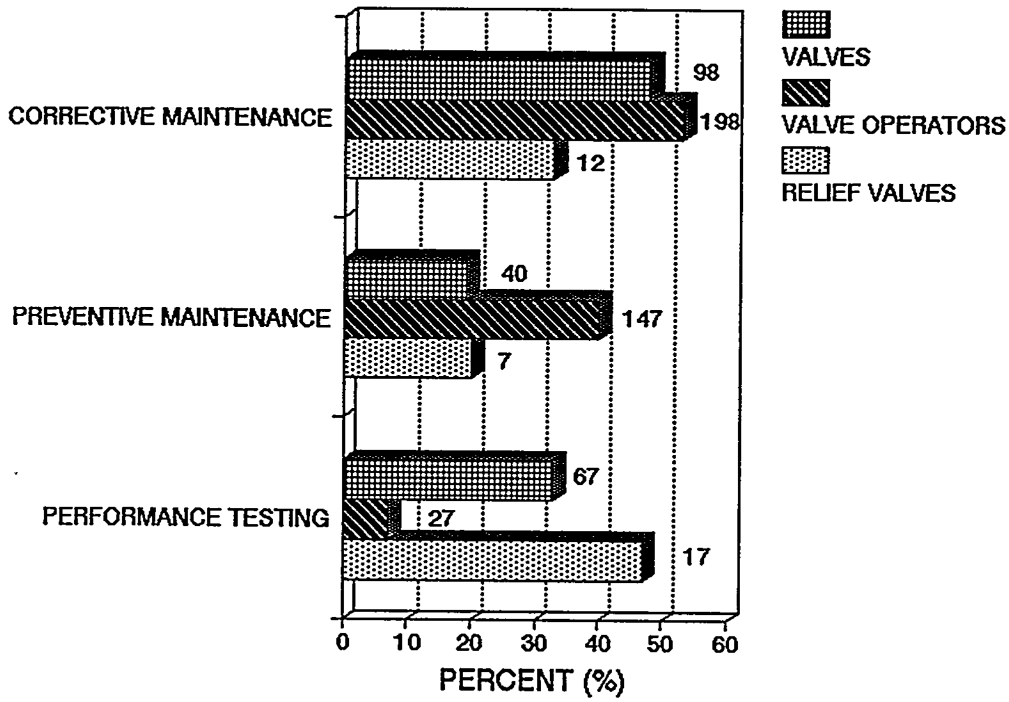


Figure 7.3 Work order category-Plant A

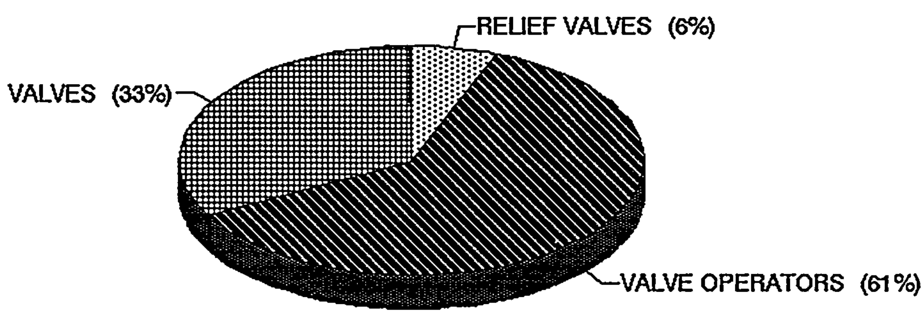


Figure 7.4 Work orders by component-Plant A

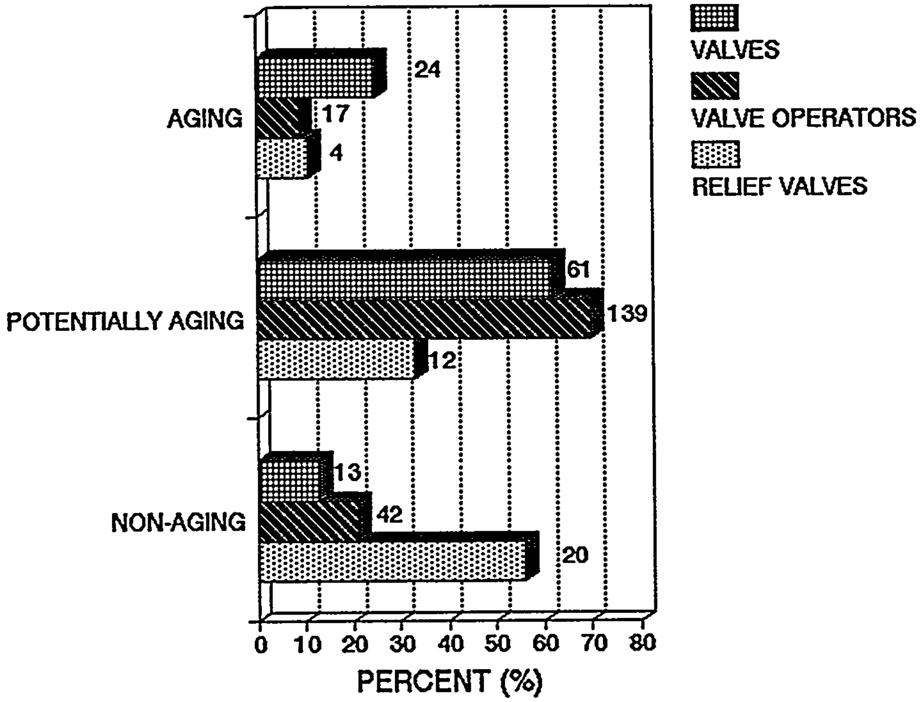


Figure 7.5 Corrective maintenance work orders caused by aging-Plant A

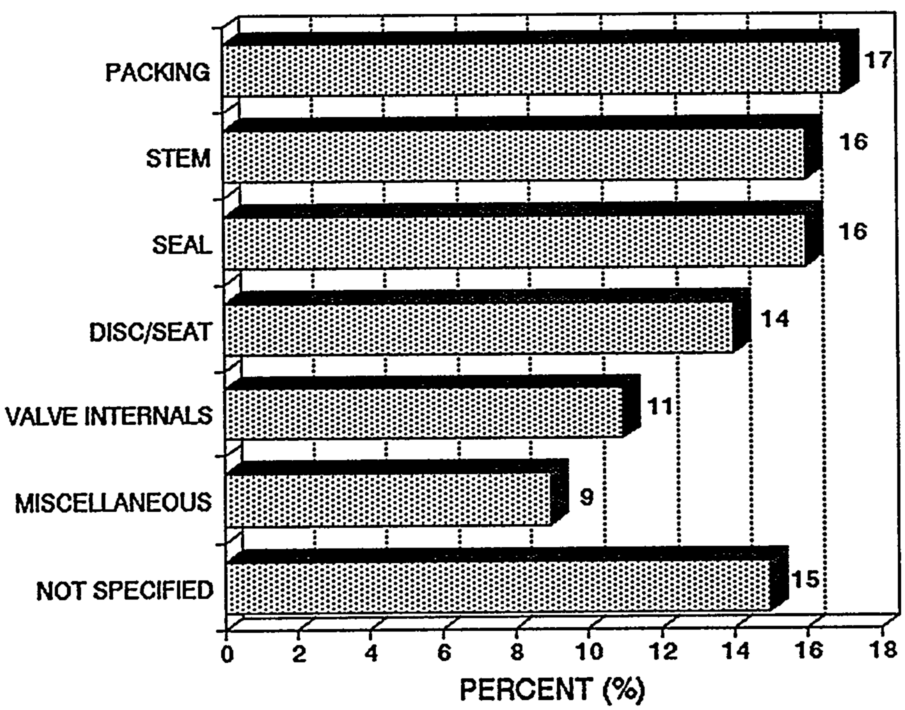


Figure 7.6 Failed valve sub-components-Plant A

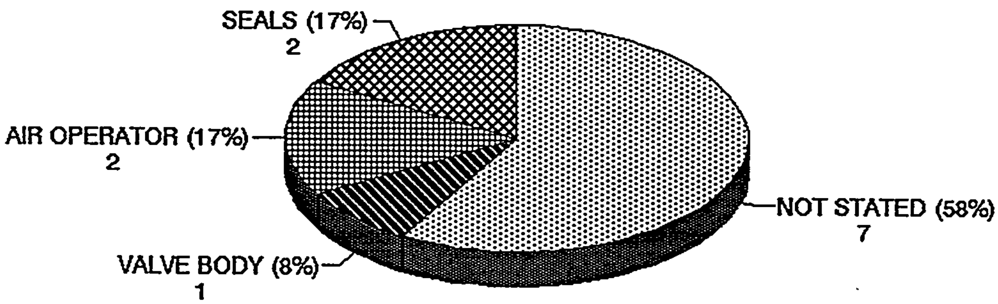


Figure 7.7 Failed relief valve sub-components-Plant A

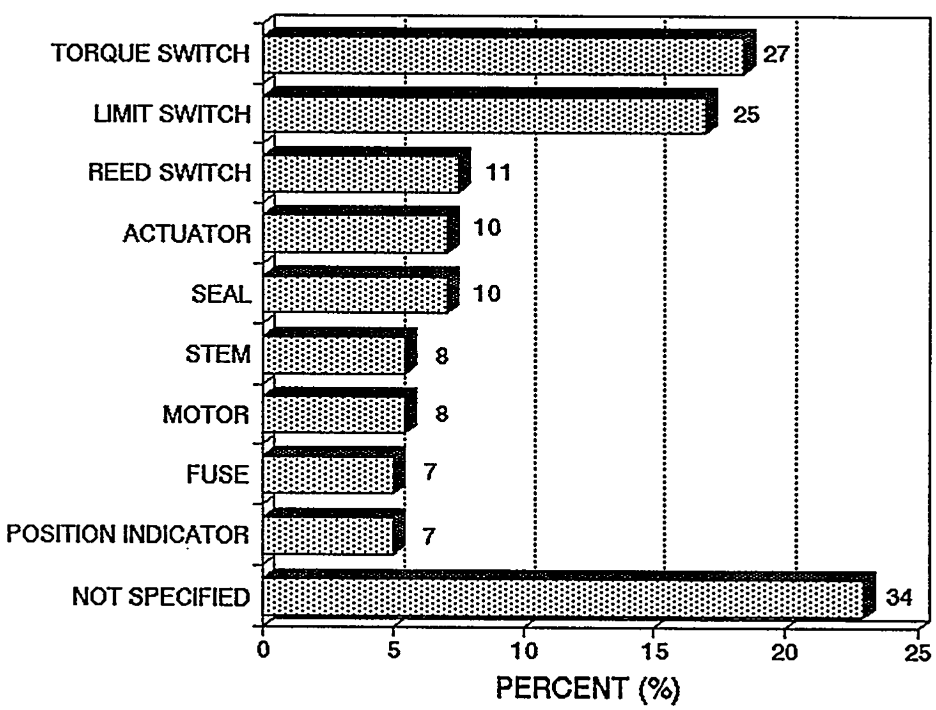


Figure 7.8 Valve operator failed sub-components-Plant A

Since the switches were the most frequently degraded sub-component in the valve operator, it would be anticipated that the effect of these occurrences would be improper operation. Similarly, since packing and seals were commonly reported failed sub-components of valves, leakage (internal and external) was anticipated since they are part of the pressure boundary. These predictions were confirmed by the reported failure modes (Figure 7.9). At Plant A, failures of valves and valve operators affected the components operation (i.e. loss of position indication, leakage), but did not significantly result in their inoperability. The failures of the relief valves also resulted in the component's failure to operate properly. However, since these faults were typically detected during bench testing, with the component removed from service, the effect upon both plant and system operation was minimized.

Both IST and surveillance testing (e.g., Technical Specification) accounted for the detection of approximately 40% of the valve failures (Figure 7.10). For valve operators, operational abnormalities (i.e., loss of, or erroneous position indication) were the most frequent method of detection. Since relief valves are not operated normally during plant operation, the majority of failures were detected during their removal and bench testing (ASME Section XI). Most of the failures did not necessitate component replacement (Figure 7.11). For valve operators, the most common corrective action was recalibrating or replacing the switches. For valves, repair and replacement of the packing, seals, and gaskets were the most common actions. For relief valves, most often the setpoint was adjusted. All of these components are readily accessible for inspection during preventive maintenance, and with their relatively high frequency of failures, operators may consider reviewing the maintenance program to ensure these sub-components receive adequate attention.

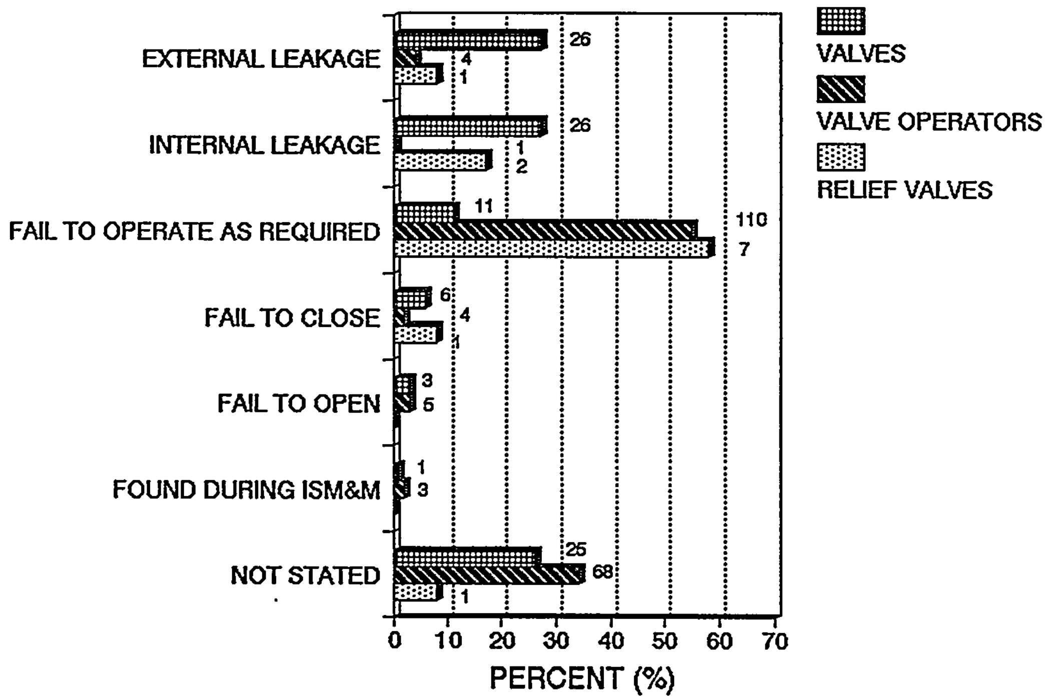


Figure 7.9 Containment isolation component failure modes-Plant A

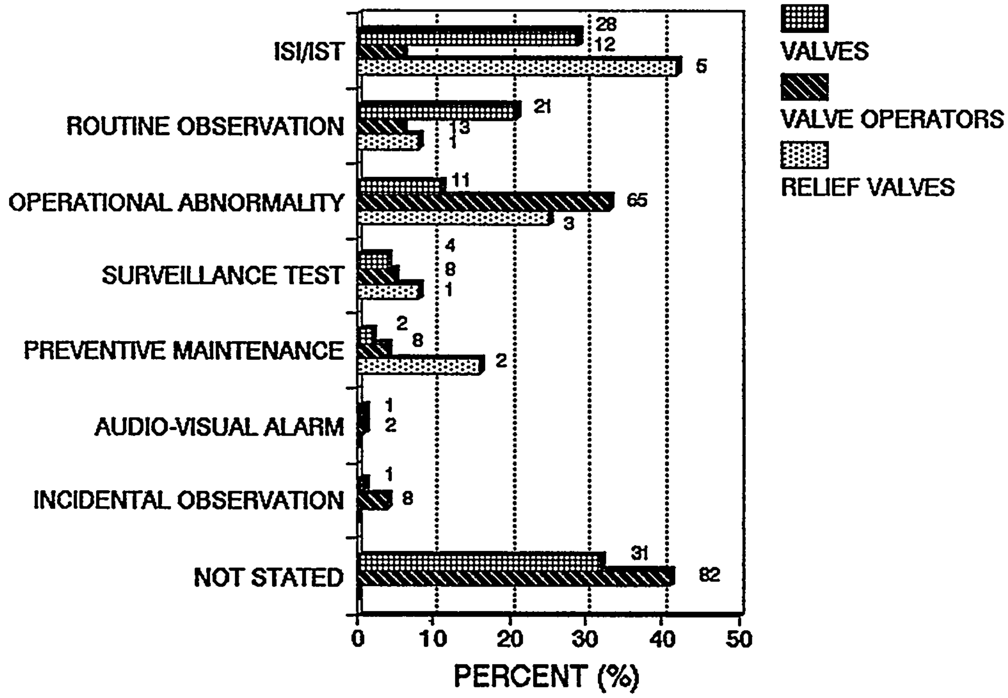


Figure 7.10 Failure detection methods-Plant A

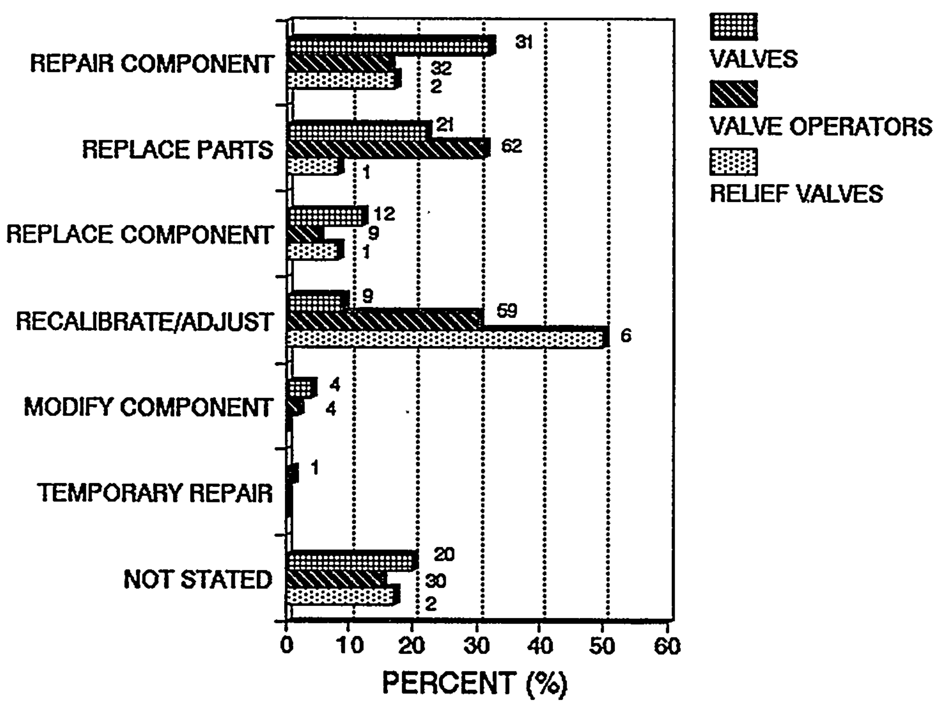


Figure 7.11 Containment isolation component corrective actions-Plant A

7.3 Summary

Plant A generated 613 maintenance work orders for valves and valve actuators used for containment isolation between 1984 and 1994. These were evenly divided between corrective and preventive maintenance, and testing. The preventive maintenance and performance testing activities were primarily in response to IST and Technical Specification requirements. Based upon a review of these, the majority of the corrective maintenance activities were in response to aging (14%) and potentially aging related failures (64%).

There were 85 aging-related valve failures at this plant during the 10 year period, and the major cause was aged disc/seat/internals, followed by worn packing, degraded/binding stem, and degraded seal/gasket. These results agree with those from the national operating data analyses.

One hundred and fifty six valve operators failed during the 10 year period, far more than those of valves. On the other hand, the national data base showed many more valve failures than those of valve operators. The exact reason for this difference is unknown, but may be due to much higher number of solenoid valves employed at this plant (23% vs. 4% for all BWRs). This difference also suggests that the aging management program should also consider the plant specific information in addition to the knowledge obtained from the national data analysis. The major causes for the valve operator failures are failed limit and torque switches, which agrees with the results from the NPRDS data analysis.

The importance of a thorough root failure cause analysis program was highlighted. Numerous examples of component failures were identified which did not have a cause specified. Numerous valve operator failures were attributed to switch failures, which resulted in replacement of the sub-component. However, no information was provided to determine if the switch failed each time, or failed as a result of some other cause (i.e., electrical surge, environmental degradation, etc.). Efforts should be made to identify the causes of failures to eliminate the potential for repeat type failures.

8. INSPECTION, SURVEILLANCE, AND MONITORING (IS&M) PRACTICES

In this section, inspection, surveillance, and monitoring methods used for the containment isolation valves and valve operators are reviewed to provide an evaluation of industry practices to detect and mitigate aging degradation. Also, using the information in the NPRDS data, we attempt to assess the effectiveness of these programs in detecting degradation and failures.

8.1 Review of Current Practices

General Design Criteria 54, of Part 50 Appendix A, requires piping penetrating the containment to be designed so that the isolation valves can be tested periodically to determine if valve leakage is within acceptable limits. These containment isolation valves are tested in accordance with the requirements of 10CFR50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-cooled Power Reactors." Appendix J describes three types of leakage tests, designated as Type A, B, and C. The "Type A" test measures the overall integrated rate of containment leakage three times every ten years, and is referred to as the ILRT (Integrated Leak Rate Test). This test ensures that the summation of leakages through all potential leakage paths including the containment's welds, valves, fittings, and components that penetrate containment are below the acceptance limits for the reduced and peak pressure tests (i.e., 0.75 of the maximum allowable leakage rate, as specified in plant technical specifications). "Type B" test measures local leakage across each pressure-containing or leakage-limiting boundary at containment penetrations, air-lock door seals, and doors with resilient seals or gaskets. The "Type C" test is performed on containment isolation valves that:

1. Directly connect the inside and outside atmospheres of the containment under normal operation, e.g., purge and ventilation valves.
2. Close automatically upon receipt of a containment isolation signal.
3. Operate intermittently under post-accident conditions.
4. Are in the main steam and feedwater piping and other systems which penetrate the containment of BWRs.

Type B and C tests are performed every refueling outage, not to exceed 2 years, and are referred to as LLRT (Local Leak Rate Test). Additionally, the air locks are tested after being opened, and at 6-month intervals. The tests determine the penetration leakage, e.g., the leakage through two CIVs installed in a line penetrating the containment. Leakage rates of individual valves generally are not measured during the Appendix J tests. The combined leakage for all penetrations and valves shall be less than 0.60 the maximum allowable leakage rate. Currently, the regulations, 10CFR50.55a(b) require assigning limiting leakage rates for each penetration, or group of valves, for monitoring the condition of the valves and taking corrective action. The NRC is considering revising the regulations to eliminate this requirement.

Individual leakage rates are specified, however, for those CIVs that also are classified as pressure isolation valves (per Generic Letter 87-06) and for certain valves or penetrations specified in plant technical specifications (e.g., MSIVs and primary containment purge valves with resilient seals in BWRs).

As discussed in SECY-94-283, [Ref. 37] and in NUREG-1493, [Ref. 38], the Type B and C tests are very effective at detecting leakages in the containment; over 97% were detected with most identified by the Type C test of CIVs. The NRC proposed revising the regulations to make Appendix J less prescriptive and more performance-based. The Type A test interval may be extended to once every 10 years based on satisfactory performance of two previous tests. Additionally, the interval for Type B and Type C tests may be extended up to ten and five years, respectively, based on the experience history of the components. Additionally, the NRC will consider establishing a risk-based performance standard for the tests, i.e., the allowable containment leakage rate.

In addition to Appendix J, the regulations require in-service testing of valves in accordance with ASME Section XI. Motor- or air-operated valves are stroke-tested quarterly, or, if this is impractical, during cold shutdowns or refueling outages. Containment isolation stroke times are established to prevent offsite radiological doses from exceeding the requirements of 10CFR100 and, in the case of the feedwater and main steam valves, to limit the maximum containment pressure after a main steam line break (MSLB). ECCS containment isolation valves' opening times are based on peak clad temperatures during design-basis accidents. The stroke time is trended, and corrective action is taken if it exceeds preset action and alert levels.

CIVs that are check valves with a safety function to open, in addition to their function to close, are full-flow tested quarterly, or during cold shutdowns or refueling outages. The excess flow check valves used predominately in BWRs to isolate instrumentation lines are tested at refueling outages, in accordance with the plant's technical specifications and Section XI.

The regulations also impose the in-service inspection requirements of ASME Section XI. The CIVs and piping between the inboard and outboard valves generally are classified as ASME Class 2 (unless the piping is part of the RCS, in which case it is classified as ASME Class 1). Passive pressure-retaining piping is pressure tested in accordance with Section XI once every three years, hydrostatically tested once every ten years, and a sample of welds (7.5% of Class 2 welds) are non-destructively tested once every ten years in accordance with Section XI.

Additionally, plant technical specifications require verification every 31 days of the closure of primary containment purge valves, manual valves, and blind flanges located outside containment. Manual valves and blind flanges located inside containment must be verified before entering Mode 3 (for BWRs, if the containment was de-inerted) or Mode 4 (for PWRs). The technical specifications for BWRs also require verification of the transversing incore probe (TIP) isolation valves' explosive charge every 31 days.

Many CIVs are located inside the containment and must be replaced periodically due to environmental qualification concerns. For example, solenoid valve's coils, o-rings and gaskets are periodically replaced. Additionally, valve motor and pneumatic operators must be maintained, and must include MOV testing in accordance with Generic Letter 89-10. [Ref. 39]

Table 8.1 summarizes the inspections, surveillances and monitorings carried out on the components of the containment isolation function.

8.2 Failure Detection Methods

The NPRDS database was reviewed to determine the effectiveness of IS&M practices in detecting the degradation and failures of components before they fail on demand. Only the major components, valves and valve operators, of both PWRs and BWRs were studied. The results show that most of the failures were detected by surveillance testing. However, there are noticeable differences in the

Table 8.1 Typical Inspection, Surveillance and Monitoring Practices for the Containment Isolation Function

Component	IS&M Practice	Frequency
Containment isolation valves and valve operators	Verify closed position[2]	31 days
	Verify stroke time[1,2]	Quarterly/cold shutdown/ refueling outages
	Verify full stroke (check valves)[1]	Quarterly/cold shutdown/ refueling outages
	Verify valve seat leakage[1,2,3] (Type C test)	2 years
	Verify automatic valves actuate to correct position during simulated automatic actuation[2]	18 months
	Verify continuity TIP isolation valve explosive charge[2]	31 days
	Verify position indication[1]	2 years
Piping penetrating primary containment	Verify no system external leakage during inservice test[1]	3 years
	Hydrostatically test piping[1]	10 years
	Non-destructively examine sample of piping welds[1]	10 years

1. ASME Section XI requirement
2. Technical Specification requirement
3. Appendix J requirement

percentages of the failures detected between different types of valves or valve operators, and also between PWRs and BWRs for the same type of components. These findings are briefly discussed for each type of component in the following sections.

8.2.1 Valves

Three different types of valves are commonly cited for failure in both PWRs and BWRs, globe, gate, and check valves. The globe valve is the major type in PWRs, accounting for about 41% of the aging-related valve failures. The gate valve is the major type for BWRs, also accounting 41% of the aging-related failures. Also, PWR CI functions employ a significant number of butterfly valves, which accounted about 15% of the aging-related valve failures. In this section, we discuss how the failures of each type of valve were detected for both PWRs and BWRs.

NPRDS has recently consolidated the failure detection methods from 9 categories to 4. Studies on effectiveness of in-service testing concluded, (though this was not reflected by the NPRDS data), that all the detection methods (e.g., ISI/IST, surveillance testing, and special inspection) must be combined to correctly assess effectiveness [Ref. 40]. A review of the failure narratives indicated that most surveillance testing is, in fact, IST, i.e., during regularly scheduled quarterly testing, though it is not classified as such. This may be attributed to the fact that the technical specification surveillance requirements reference ASME Section XI for valve testing. Since the data we used were reported before the new method was employed, they are analyzed as reported.

Globe Valves

Figure 8.1 shows the numbers of failures of globe valves detected by each method for PWRs and BWRs. Both exhibit a similar trend except that in-service inspection detected a higher percentages of failures, 18%, for the BWRs, compared to that for PWRs, 6%. However, this probably is due to the differences in reporting between different plants. Thus, in this report, all the failures detected by surveillance testing, in-service inspection and special inspection are combined, and referred to as failures detected by testing. For PWRs and BWRs, testing detected 63% and 76% of the failures of globe valves, respectively.

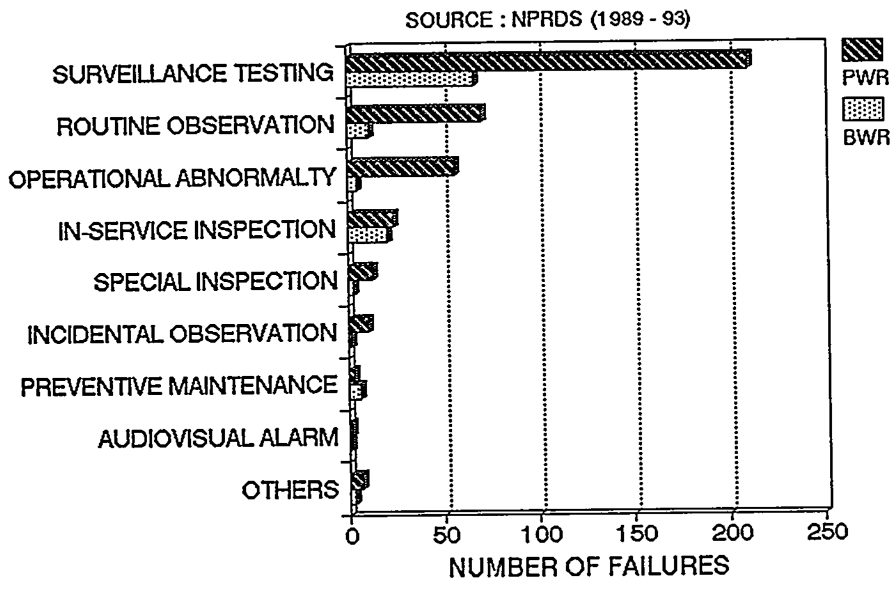


Figure 8.1 Detection methods for globe valves in PWR and BWR CI functions

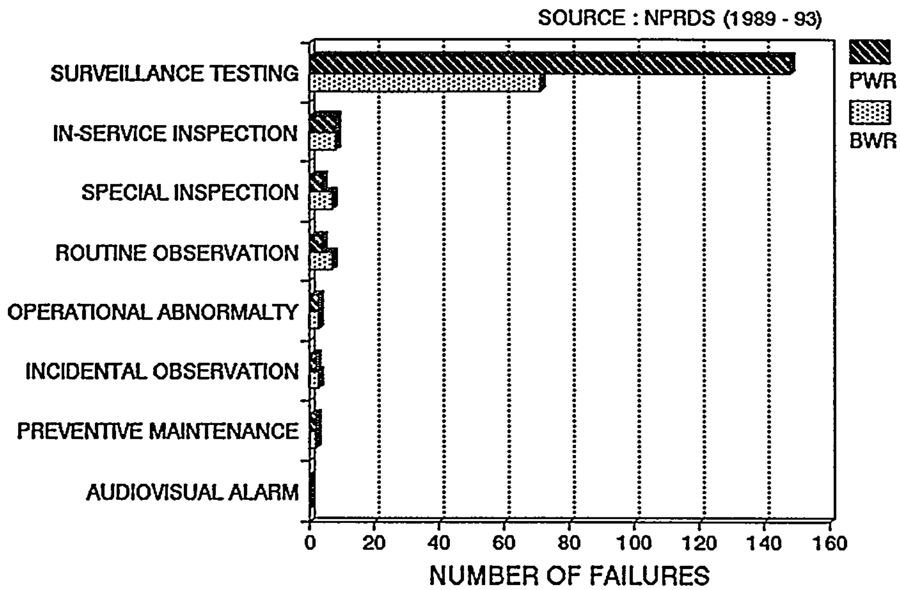


Figure 8.2 Detection methods for check valves in PWR and BWR CI functions

Check Valves

Figure 8.2 shows most of the failures of check valves were detected by surveillance testing, 94% of PWRs and 85% for BWRs, much higher than those for other types of valves. As discussed in Sections 5 and 6, the main failure modes for check valves are internal leakage and fail to close, which cannot be easily detected by routine or incidental observation. For globe valves and gate valves, external leakage is one of the major failure modes, and here, a significant portion of the failures are detected by routine or incidental observation.

Gate Valves

Figure 8.3 shows how the failures of containment isolation gate valves in PWRs and BWRs were detected, and highlights differences between the two. Testing detected about 58% of the gate valve failures in PWRs, which is far lower than the 78% for BWRs. About 27% of the gate valve failures in PWRs were detected by routine observation compared with 9% for BWRs. However, the root cause for this difference should be identified, so that testing procedures and schedules can be improved to detect more of the failures of gate valves in PWRs; this may be achieved by comparing the detailed testing procedures for PWRs and BWRs.

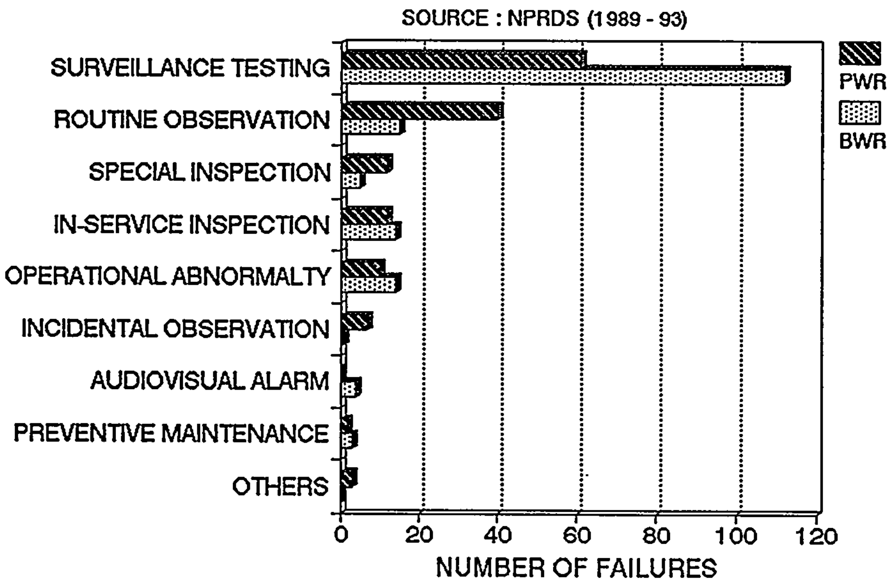


Figure 8.3 Detection method for gate valves in PWR and BWR CI functions

Butterfly Valves

As discussed earlier sections (Sections 5 and 6), there were fewer reported failures of butterfly valves in the BWRs, probably due to their much lower population. Figure 8.4 shows that the way the failures of butterfly valves were detected is very similar to that for check valves. Again, the failure mode analysis showed that external leak is not a major failure mode for butterfly valves, which is same for the check valves.

8.2.2 Valve Operators

AC Electric Operators

Figure 8.5 shows that about 63% and 65% of the failures of AC electric valve operators were detected by testing at PWRs and BWRs. These percentages are much lower than those identified for the valves. Also, 28% of the failures of BWR AC operators, were detected by operational abnormality compared to 15% for PWRs; this is discussed again in a later section.

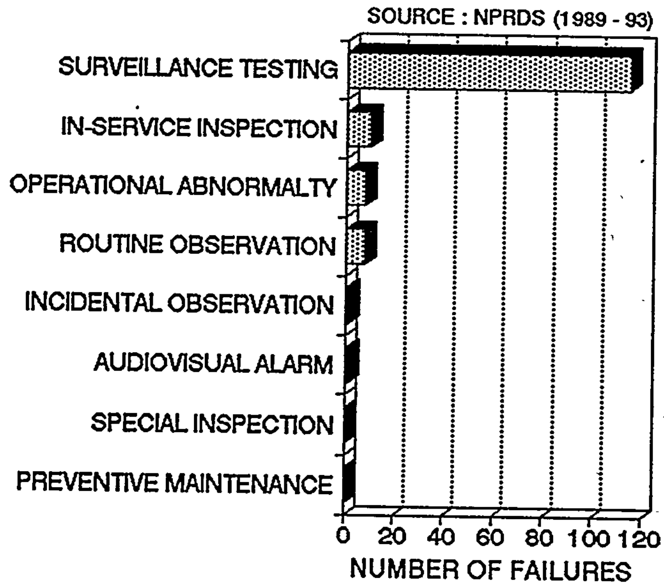


Figure 8.4 Detection methods for PWR CI butterfly valves

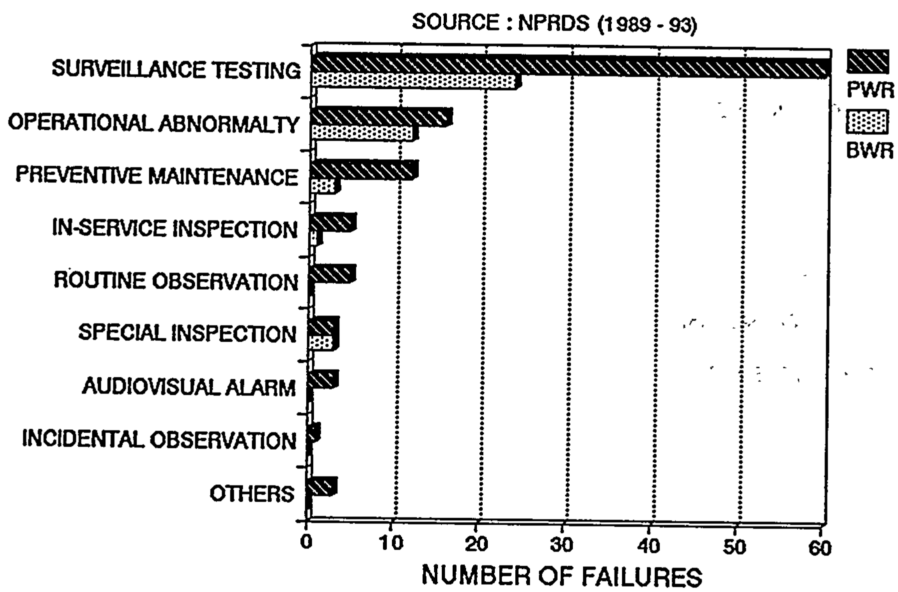


Figure 8.5 Detection methods for AC electric valve operators in PWR and BWR CI functions

Pneumatic Operators

Figure 8.6 shows that only 47% of the failures of pneumatic valve operators in the PWRs were detected by testing, while 61% were for BWRs. Further, 31% of such failures in PWRs were detected by operational abnormality, but 21% for BWRs.

8.3 Effectiveness of Testing

It is desirable to detect most failures during inspections, surveillance testing, and periodic in-service testing, and very few during operation. Thus, a criterion that can be used to assess the effectiveness of the testing practices is identifying what fraction of failures is detected by operational abnormalities; the fraction for each component is shown in Figure 8.7. For valve operators, testing failed to detect a large portion of the problems, which resulted in operational abnormalities. Thirty one percent of the failures of pneumatic valve operators for PWR CI functions and 27% for AC electric valve operators for BWR CI functions were detected as operational abnormalities. These percentages are very high, and it is desirable to reduce these drastically. The fraction for PWR pneumatic valve operators (21%) and for AC electric valve operators (15%) were quite high, also.

As discussed earlier, there is a big difference between the fraction of failures for PWR globe valve (14%) and that for BWR (4%), and a detailed study is warranted to investigate the cause for the difference, and so improve the effectiveness of the testing practices.

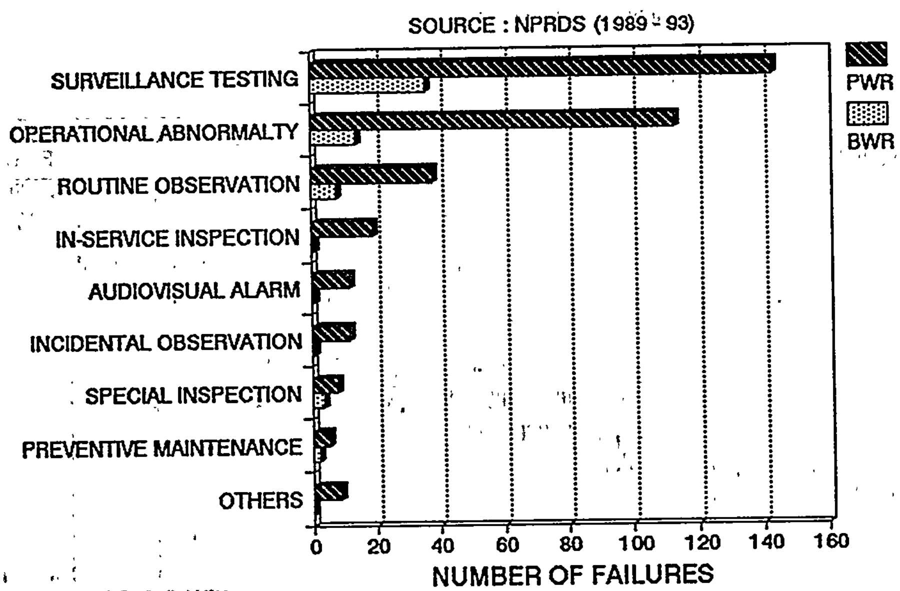


Figure 8.6 Detection methods for pneumatic valve operators in PWR and BWR CI functions

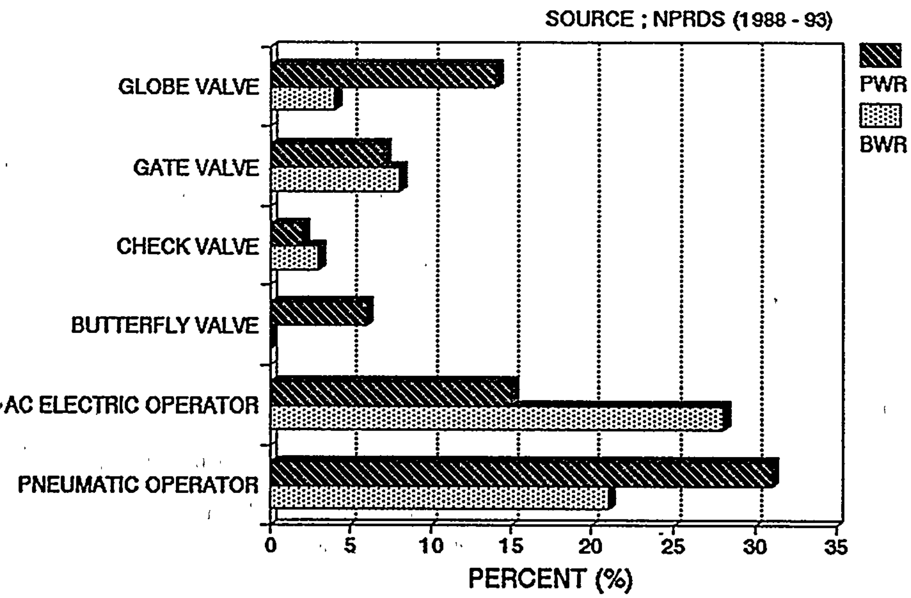


Figure 8.7 Fractions of the failures detected by operational abnormality for different components

8.4 Summary

Due to the importance of the containment isolation functions for both PWRs and BWRs, the valves and valve operators receive rigorous testing specified in the Technical Specifications. Thus, most failures are detected before they affect the operation and the safety of the plant. It is generally believed that the current IS&M practices are satisfactory in detecting failures and in preventing the containment isolation function failures. However, our analysis of operating data shows the areas that may need improvements.

At BWR plants, testing detected much higher percentages of failures of globe and gate valves than at PWR plants; 76% and 78% for BWR globe and gate valves, respectively, 63% and 58% for PWR globe and gate valves, respectively. For valve operators, about 63% and 65% of the AC electric valve operator failures were detected by testing for PWRs and BWRs, respectively. However, only 47% of the pneumatic valve operator failures in the PWRs were detected by testing, while 61% was for BWRs. In short, BWR plants detected a much larger portion of the failures of both valves and valve operators by testing than PWR plants.

9. SUMMARY AND CONCLUSIONS

The containment isolation (CI) function is an engineered safety feature (ESF) that prevents and minimizes the release of radioactivity in the unlikely event of design basis accidents (DBAs). The CI function is not a single, well defined system in either GE BWRs or Westinghouse PWRs. For the Westinghouse PWRs, the containment isolation function consists mainly of containment isolation valves and valve operators. Automatic containment isolation is initiated by the engineered safety features actuation system (ESFAS), which is not included in this study. However, for BWRs, most of the containment isolation valves and valve operators are in the nuclear steam supply shutoff system, which also initiates the containment isolation signals. For this reason, the failure data of switches, bistables, radiation detectors, and transmitters in the CI functions of the BWRs were analyzed in this study.

This study reviewed the safety analyses related to the containment isolation function and focussed on those components which were most risk-significant based upon specific accident scenarios. Previous NRC studies indicated that immediate detection of the failures is important in further reducing the safety significance of failures in the containment isolation function.

We found that a large portion of failures reported to NPRDS during the period of 1988 - 1993 are aging related, 76% for PWRs and 80% for BWRs. The stresses that cause aging degradation of the components in the containment isolation functions include environmental stresses and stresses from operations and testing. Since the CI components are required to support safe plant operation, they are frequently tested and the components experience stresses from both operations and testing. Also, a significant portion of the components are located inside the primary containment, where the environmental stresses, including radiation, temperature, and humidity, are greater than outside.

There were 2,042 component failures in the containment isolation (CI) functions of the Westinghouse PWRs reported to NPRDS during the 5 years, 1989 - 1993. About 76% of these were aging-related failures. Most failures were those of valves (965;47%) or valve operators (668;33%), which, together, accounted for more than 80% of all the component failures in the CI function. The major type of valve for CI functions in the Westinghouse PWRs is the globe valve, which accounted for 41% of all the aging-related failures. The other types are check valves (18%), gate valves (15%), and butterfly valves (14%). The major types of valve operators are pneumatic operator and AC electric motor operator, which accounted for about 74% and 22% of the aging-related valve operator failures in the CI functions.

The failure frequency curve for all the PWR valves shows a high peak at 3 years, which decreases slowly until 11 years, after which it increases rapidly. When the failure frequencies for different types of valves are plotted, globe and gate valves showed a similar trend as that for all the valves combined. On the other hand, check valves and butterfly valves showed a different trend; the failure frequency did not change much, except to peak at the age of 18 for butterfly valves and at 23 years for check valves.

The major failure modes for the globe valves are internal leaks, external leaks, and failure to close, caused primarily by aged seat/disc, worn packing, and corrosion product/dirt buildup, respectively. As the globe valves get older, the fractions of the failures caused by aged seat/disc and worn packing increase, resulting in increased fractions of internal leak and external leak, respectively. After about 10 years, there is a large increase of failures caused by worn gaskets, which also result in external leaks.

There are only two major failure modes for check valves, internal leak and failure to close; these were not affected by aging, and stayed about the same over the years. Aged (worn/corroded/damaged) seats/discs and corrosion product/dirt buildup were the main causes for check valve failures, which also were not affected significantly by aging. For gate valves, the main failure modes are internal and external leaks. As the valves get older, more failures are due to internal leaks than external leaks, caused by worn packing, aged seat/disc, and corrosion product/dirt buildup. The butterfly valves failed mainly with internal leaks; the major causes being corroded/worn/damaged seat/disc and corrosion product/dirt build-up, demonstrating that most valve failures were caused by corrosion and aging of seats and discs.

The failure frequency of the valve operators continuously increases with age, which is quite different from the effects of aging on failure frequency for valves. Because most of the valve operator data are from the pneumatic operators (74%), they dominate the combined frequency curve for valves. For pneumatic valve operators, failure to open is the leading failure mode, followed by failure to close, and failure to operate as required. For the containment isolation function, "failure to close" is the only important mode, which makes the number of failures that affect the function performance much less than those analyzed in this report. The failure frequency of pneumatic valve operators increases continuously though slowly for the first 15 years. Starting at 16 years, the failures began to increase rapidly. The major causes are failures of the diaphragm, solenoid valve, limit switch, and air regulator/line. The effects of aging on these causes all are similar; the numbers of failures caused by them slowly increase during the first 14 - 15 years, followed by rapid increases.

During the six years, 1988 - 1993, there were 139 aging-related failures of components reported as LERs for the Westinghouse containment isolation functions. The components generating the LERs are valves (47%), valve operators (32%), penetrations (10%), and pipes (4%). The percentages for failures of valves and valve operators are very close to those reported to NPRDS, which are 49% and 34%. The causes of failure for valves identified from LERs are worn seats/discs, corrosion, and seal/gasket degradation, agreeing with our analyses of NPRDS data. For valve operators, the identified causes of failure were failed solenoid valves/coils and failed torque/limit switches, also in agreement with the NPRDS data analysis. The major failure mode for the valves was internal leakage, accounting for almost 70% of the LERs caused by valve failures. For valve operators, failure to close is the major failure mode, accounting for about 60% of LERs; this value differs from the results of the NPRDS data analysis because most of the component failures are reported to NPRDS, while only safety-significant failures are reported as LERs.

There were 1,477 component failures in the CI functions of the GE BWRs reported to NPRDS during the 6 years, 1988 - 1993. About 80% of these were aging-related failures. The major containment isolation components for which failures were reported most often (not the failure rate) are valves (35%), bistables/switches (30%), valve operators (14%), and radiation detectors/transmitters (12%). About 86% of the valve failures were aging-related; 85% and 67% for bistables/switches and valve operators, respectively.

Unlike PWRs, gate valves are the major contributor to valve failures in the CI function (41%) for BWRs, followed by globe valves (27%) and check valves (23%). For PWRs, 41% of the failures were those of globe valves. The failure frequency of all the BWR CI valves peaks at 6 years, then decreases until 15 years, when it begins to increase. The major failure causes for gate valves are aged seats/discs/wedges, corrosion product/dirt buildup, and worn packing, which result in internal leaks, failure to close, and external leaks, respectively. For globe valves less than 10 years old, aged seats/discs, worn packing, and corrosion product buildup are the major failure causes, generating internal leaks, external leaks, and failure to close, respectively. Beyond 11 years, corrosion product/dirt buildup is the predominant cause, accounting for 63% of the failures. For check valves, internal leak is the main failure mode accounting for more than 50% of the failures. During the first 10 years, corrosion product/dirt buildup caused the most check valve failures, then aged seat/disc. The failures of older check valves mainly are caused by aged seat/disc.

The valve operator types whose failures were most often reported to NPRDS are pneumatic (49%) and AC electric (34%). About 66% of the failures are aging-related. There is a rapid increase of failures of AC electric valve operators during the first 6 years, when they peak. During the first 10 years, the failures of torque switches and limit switches caused 36% and 18% of these failures, respectively. After the peak at 6 years, the failure frequency decreases to a very low value. Around 15 years, the frequency begins to increase, mainly due to the increase of valve motor failures. For pneumatic valve operators, the failure frequency remains relatively constant during the first 10 years, and there is a dramatic increase at 16 years. Failure of the solenoid valve is the main cause for the BWR pneumatic valve operator failures; the failure frequency of the solenoid valves stays almost constant for about 15 years, after which there is a large increase. This trend is identical for the pneumatic valve operators of both BWRs and PWRs.

There were 109 LERs caused by aging-related failures of the components in the containment isolation function of the BWRs; about 65% were due to failures of valves, 8% due to relays, and 6% due to penetrations. The failures of valve operators and circuit breakers each generated 6% of the LERs. Unlike the information reported to NPRDS, most LERs do not contain root causes. Among the LERs that contained the causes of valve failures, aged seats/discs, worn packing, and corrosion product buildup were the major ones. These findings are consistent with the information from the NPRDS analyses.

At BWR plants, testing detected much higher percentages of failures of globe valves and gate valves than at the PWR plants; 76% and 78% for BWR globe and gate valves, respectively, while 63% and 58% for PWR globe and gate valves, respectively. For valve operators, about 63% and 65% of the AC electric valve operator failures were detected by testing for PWRs and BWRs, respectively. However, only 47% of the pneumatic valve operator failures in the PWRs were detected by testing, while 61% was for BWRs. In short, BWR plants detected a much larger portion of the failures of both valves and valve operators than PWR plants. When the effectiveness of testing and preventive maintenance is measured by the percentages of the failures detected by operational abnormality, it was found that they were less effective for PWR globe valves, BWR AC electric valve operators, and PWR pneumatic valve operators.

As a result of this Phase I study, the aging processes used for the components of the containment isolation functions in both PWRs and BWRs are better understood through our aging characterization of major components. The components most frequently affected by aging degradation should have higher priorities in aging management. The information obtained from this study provides a technical basis for future work.

The following conclusions are made based on the findings from this study:

- A large portion of the component failures in the CI functions reported to NPRDS are aging-related, 76% for PWRs and 80% for BWRs. The results of this study also have shown that aging-related failures increase as the components in the CI functions age, and the failure modes and causes are affected by aging.

- Even though a large number of component failures in the containment isolation functions were reported to NPRDS and LERs, most of them were not safety-significant. However, for both PWRs and BWRs, during the period studied, there were several failures of components in the CI function that affected the plant operation, either resulting in unit off-line, or in a reduction of power operation.

- Many reported failures were caused by regular maintenance items, such as valve packings. Utilizing the insights from the operating experiences and aging studies may be beneficial in reducing these aging-related failures.

- This study characterized the aging degradation of different types of valves and valve operators separately, and the results show that different component types show different aging effects. For operating data analyses, it is recommended that the groupings should be as small as possible, as long as there are enough data available for meaningful analyses.

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APPENDIX A

Descriptions of Different Containments

Note: The following information is abridged from NUREG/CR-4242 titled "Survey of Light Water Reactor Containment Systems, Dominant Failure Modes, and Mitigation Opportunities, Final Report" prepared by J. N. Castle et al. [Ref. 41].

A.1 BWR MARK I Containment

The Mark I containment for GE BWR is the oldest containment used for BWRs, which has been replaced with Mark II and III containment in the later designs. The Mark I primary containment consists of lightbulb-shaped drywell and the torus-shaped wetwell containing the suppression pool of water. In this containment design, the steam and noncondensible gases generated during a severe accident are routed through the suppression pool. The primary containment boundary formed by drywell, the suppression pool, and the interconnecting duct work is designed to be leak tight. The plant's reactor building completely surrounds the primary containment, which is the secondary containment.

In general, two isolation valves are provided for all fluid lines connected to the reactor pressure boundary or containment atmosphere. Valves that are pneumatically or electrically actuated are spring-loaded to fail in the closed position for nonemergency systems. The valve closure time and valve leakage rates are such that the site boundary dose criteria of 10CFR100 are satisfied during DBAs.

Containment isolation is also dependent on the proper functioning of numerous seal and gaskets around containment penetrations which cannot be welded to the containment itself or to the containment liner. A variety of elastomeric materials are used for this sealing function. The materials are, however, are susceptible to temperature, humidity, and radiation. Consequently, their performance is checked periodically when the overall containment leakage rate is measured while the containment is deliberately being pressurized. The containment temperature limitations ar a consequence of the temperature sensitivity of the elastomeric materials.

There are 24 operating GE plants with Mark I containment.

A.2 BWR MARK II Containment

Mark II containment replaced the earlier Mark I design, and 10 US plants have this type containment. The suppression pool is located below the cone-shaped drywell. Steam released to the drywell during an accident is routed to the suppression pool by multiple steel downcomer pipes. The design leak rate is 0.5 to 1.1 percent of the primary containment volume per day at the full design pressure. The leakage rate can be measured during normal plant operations at a pressure of 115 percent of the design pressure. The drywell design temperature is 340 °F, and that for suppression pool is 275 °F.

The advantages of the Mark II containment over Mark I are the following:

1. increased drywell volume to accommodate steam and ECCS piping as well as recirculation pumps.

2. simpler vent configuration from the drywell to the wetwell via straight pipes.

3. increased wetwell volume

4. reduced reactor building volume

In Mark II containment plants, the containment is inerted with nitrogen for hydrogen combustion control, which was a result of TMI-2 accident.

There are 8 operating GE plants with Mark II containment.

A.3 MARK III Containment

The nuclear steam supply system and associated safety systems in the Mark III plants are very similar to those for Mark II plants. The suppression pool is an annular configuration around the inner, lower perimeter of the primary containment, and the water volume in the suppression pool is about the same as that for Mark II. The design pressure of the primary containment is 15 psig at a maximum temperature of about 185 °F, which is much lower that of Mark I and II. For this containment, nitrogen inerting is not required because of the large containment volume. This fact facilitates maintenance activities within the containment.

There are 4 operating GE plants with Mark III containment.

A.4 PWR Large Dry Containment

In this design, large volume and mass of the containment structure is used to minimize the temperature and pressure in case of a DBA, which will result in minimum leakage of the radioactivity to the outside of the containment.

In contrast with BWR pressure vessels, the reactor vessel in most PWR plants is supported at the upper head flange or at the main nozzles on the side of the vessel. BWR vessels are supported by a circumferential skirt attached to the vessel heal. The BWR vessel skirt is susceptible to thermally induced damage because of its position in the event of a core meltdown accident. PWR vessels are likely to remain in place despite serious melting in the lower portions of the vessel. This feature helps prevent gross vessel motion from tearing out or rupturing piping penetrations in the primary containment.

There are 35 Westinghouse plants that have large dry containment.

A.5 PWR Ice Condenser Containment

Traditionally, dry containments relied on a combination of structural strength and volume to contain potential steam and radioactivity releases from the reactor primary system in the event of an accident. There are 8 plants with this type of containment in the US, and six of them have secondary containment. The ice condenser is essentially a cold storage room holding approximately 2.3 million pounds of borated ice in perforated metal baskets.

The notable advantages of this type of containment over the large dry containment are the following:

1. The containment volume is smaller, and the design pressure is reduced by 75%.

2. The lower pressure during DBAs cause less leakage through the containment. The time the containment is at an elevated pressure during an accident is reduced also further reducing leakage of radioactive materials.

There are 8 operating Westinghouse plants with ice condenser containment.

A.6 PWR Subatmospheric Containment

All of the subatmospheric containments are built with reinforced concrete without secondary containments. These primary containments are normally at an operating pressure of 9 to 11 psig. Following a LOCA, the pressure is returned to subatmospheric by the ECCSs, when the driving force for the containment leakage does not exist. This is the technical basis for the absence of the secondary containment. However, extra maintenance is required to maintain the subatmospheric atmosphere.

There are 7 operating Westinghouse plants with subatmospheric containment.

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10. SUPPLEMENTARY NOTES
J.E. Jackson and S. Aggarwal, NRC Project Managers

11. ABSTRACT *(200 words or less)*

A study was performed to assess the effects of aging on the Containment Isolation (CI) functions of Westinghouse Pressurized Water Reactors and General Electric Boiling Water Reactors. This study is part of the Nuclear Plant Aging Research (NPAR) program, sponsored by the U.S. Nuclear Regulatory Commission. The objectives of this program are to provide an understanding of the aging progress and how it affects plant safety so that it can be properly managed. This is one of a number of studies performed under the NPAR program which provide a technical basis for the identification and evaluation of degradation caused by age.

The failure data from national databases, Nuclear Plant Reliability Data System (NPRDS) and Licensee Event Reports (LERs), as well as plant specific data were reviewed and analyzed to understand the effects of aging on the CI functions. This study provided information on the effects of aging on component failure frequency, failure modes, and failure causes. Current inspection, surveillance, and monitoring practices were also reviewed.

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