Westinghouse Non-Proprietary Class 3

WCAP-14575-A

Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components

Westinghouse Energy Systems

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-14575-A

AGING MANAGEMENT EVALUATION FOR CLASS 1 PIPING AND ASSOCIATED PRESSURE BOUNDARY COMPONENTS

Frank Klanica
Charlie Gay

December 2000

Funded by:

Westinghouse Owners Group (WOG)
Life Cycle Management/License (LCM/LR) Program

Approval:______________________
R. Llovet, Westinghouse Program Manager
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Prepared by Westinghouse Electric Company for use by members of the Westinghouse Owners Group. Work performed in Shop Order MUHP-6119 under direction of the WOG LCM/LR Program Core Group.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 8, 2000

Mr. Roger A. Newton, Chairman
Westinghouse Owners Group
Wisconsin Electric Power Company
231 West Michigan
Milwaukee, Wisconsin 53201

SUBJECT: ACCEPTANCE FOR REFERENCIN_ _. __.__..C LICENSE RENEWAL PROGRAM TOPICAL REPORT ENTITLE, "LICENSE RENEWAL EVALUATION: AGING MANAGEMENT FOR CLASS I PIPING AND ASSOCIATED PRESSURE BOUNDARY COMPONENTS," WCAP-14575, REVISION 1, AUGUST 1996

Dear Mr. Newt ɔn:

The staff of the U.S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation has reviewed the topical report entitled, "License Renewal Evaluation: Aging Management for Class I Piping and Associated Pressure Boundary Components," WCAP-14575, which the Westinghouse Owners Group (WOG) submitted in August 1996, as part of the Generic License Renewal Program (GLRP). The resultant final safety evaluation report (FSER) is transmitted to you as an enclosure to this letter.

As indicated in the FSER, the staff found the topical report acceptable for GLRP member plants to reference in a license renewal application to the extent specified and under the limitations delineated in the staff FSER and the associated topical report. The limitations include committing to the accepted aging management programs defined in the topical report, and completing the renewal applicant action items described in Section 4.1 of the FSER. An applicant referencing the topical report and meeting these limitations will provide sufficient information for the staff to make a finding that there is reasonable assurance that the applicant will adequately manage the effects of aging so that the intended functions of the Class I piping and associated pressure boundary components covered by the scope of the report will be maintained consistent with the current licensing basis during the period of extended operation.

The staff does not intend to repeat its review of the matters described in the report and found acceptable in the FSER when the report appears as reference in a license renewal application, except to ensure that the material presented applies to the specified plant.

In accordance with the procedures established in NUREG-0390, "Topical Report Review Status," the staff requests that the WOG publish the accepted version of WCAP-14575 within three months after receiving this letter. In addition, the published version will incorporate this letter and the enclosed FSER between the title page and the abstract.

To identify the version of the published topical report that was accepted by the staff, the WOG will include "-A" following the topical report number (e.g., WCAP-14575-A).

Sincerely,

CI Grimes

Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project No. 686

Enclosure: Final Safety Evaluation Report

cc w/encl: See next page

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FINAL SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CONCERNING

WESTINGHOUSE OWNERS GROUP REPORT, WCAP-14575, REVISION 1

"LICENSE RENEWAL EVALUATION: AGING MANAGEMENT FOR CLASS I PIPING AND ASSOCIATED PRESSURE BOUNDARY COMPONENTS"

PROJECT NO. 686

Table of Contents

FINAL SAFETY EVALUATION
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
CONCERNING "LICENSE RENEWAL EVALUATION:
AGING MANAGEMENT FOR CLASS I PIPING AND
ASSOCIATED PRESSURE BOUNDARY COMPONENTS"
WESTINGHOUSE OWNERS GROUP REPORT NUMBER WCAP-14575, REVISION 1

1.0 INTRODUCTION

Pursuant to Section 50.51 of Title 10 of the Code of Federal Regulations (10 CFR 50.51), licenses to operate nuclear power plants are issued by the U.S. Nuclear Regulatory Commission (NRC) for a fixed period of time not to exceed 40 years; however, these licenses may be renewed by the NRC for a fixed period of time, including a period not to exceed 20 years beyond expiration of the current operating license term. The Commission's regulations in 10 CFR Part 54 (60 FR 22461), published on May 8, 1995, set forth the requirements for the renewal of operating licenses for commercial nuclear power plants (Reference 1).

Applicants for license renewal are required by the license renewal rule to perform an integrated plant assessment (IPA). The first step of the IPA, 10 CFR 54.21(a)(1), requires the applicant to identify and list structures and components that are subject to an aging management review (AMR); 10 CFR 54.21(a)(2) requires the applicant to describe and justify the methods used in meeting the requirements of 10 CFR 54.21(a)(1); and 10 CFR 54.21(a)(3) requires that for each structure and component identified in 10 CFR 54.21(a)(1), the applicant demonstrates that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. Furthermore, the applicant must provide an evaluation of time-limited aging analyses (TLAAs) as required by 10 CFR 54.21(c), including a list of TLAAs, as defined in 10 CFR 54.3.

1.1 Westinghouse Owners Group Topical Report

By letter dated August 28, 1996, the Westinghouse Owners Group (WOG) submitted topical report WCAP-14575, "License Renewal Evaluation: Aging Management for Class 1 Piping and

Enclosure

Associated Pressure Boundary Components" (Reference 2), for staff review and approval. The focus of the report is on the management of the effects of aging of Class 1 piping and associated pressure boundary components during any extended period of operation. WOG defined Class 1 piping as piping that contains primary reactor coolant. In this safety evaluation (SE), Class 1 piping is referred to as reactor coolant system (RCS) piping.

The WOG report evaluated the aging management of the RCS piping for domestic commercial nuclear power plants with a Westinghouse nuclear steam supply system (NSSS). The objectives of the topical report are to

- Identify and evaluate aging effects that degrade intended functions
- Identify and evaluate TLAAs
- Provide options, in terms of activities and program attributes, to manage the aging effects identified in the topical report

1.2 <u>Conduct of Staff Review</u>

The staff reviewed the report to determine whether the requirements set forth in 10 CFR 54.21(a)(3) can be met. The staff issued a request for additional information (RAI) after completing its initial review (Reference 3). WOG responded to the staff's RAI (Reference 4) and provided further clarification of its response to the RAI in a meeting on July 10, 1997, with the staff.

2.0 <u>SUMMARY OF TOPICAL REPORT</u>

WOG topical report WCAP14575 contains a technical evaluation of the effects of aging of the Westinghouse RCS piping and associated pressure boundary components. The report was submitted to the NRC staff to demonstrate that WOG member plant owners can adequately manage the effects of aging during the period of extended operation. This evaluation applies to the plants listed in Table 1-1 of the topical report. The license renewal applicant should verify that its plant is bounded by the topical report. **This is Renewal Applicant Action Item 1.**

2.1 Components and Intended Functions

2.1.1 Intended Functions

Section 2.2 of the topical report identified the following intended function for the Class 1 piping and associated components, based on the requirements of 10 CFR 54.4(a):

- maintain the integrity of the reactor coolant pressure boundary.

The staff has concluded that there is an additional intended function of an associated component of the Class 1 piping, namely, the flow restrictors (see Section 3.1 of this SE).

2.1.2 Components

The report addresses the plant-specific piping and associated components of the RCS that are within the scope of the license renewal rule. The scope of the topical report includes the following categories of components:

- Class 1 piping
- Class 1 valve bodies
- reactor coolant pump (RCP) casings
- associated pressure boundary components

Section 2.0 of the topical report provides a discussion of the Class 1 piping and associated components within the scope of the rule and subject to an AMR. As discussed in Section 2.0 of the report, the associated pressure boundary components include closure bolting for the RCPs and Class 1 valves and flange bolts for the Class 1 piping.

Detailed descriptions of Class 1 piping and the associated pressure boundary components, its intended functions, and its interactions and interdependence are presented in Section 2.3 of the report. As described in the report, Class 1 piping includes large- and small-bore seamless steel pipe and fittings. For piping larger than 2 inches, butt-welded construction was used. For piping smaller than 2 inches, socket-welded or butt-welded construction was used. Exceptions include thermowells, which may use threaded connections, and safety valves and resistance temperature detector (RTD) bypass lines, which use flanged connections.

RCS piping is comprised of large seamless stainless steel pipe and fittings. The piping design specifications, in conjunction with the governing code of record, define the design and loading conditions as well as the allowable stresses.

The RCS consists of two, three, or four heat transfer loops connected in parallel to the reactor pressure vessel (RPV). Each reactor coolant loop (RCL) contains an RCP and a steam generator (SG). In addition, the RCS includes a pressurizer (PZR), a pressure relief tank, interconnecting piping, and instrumentation necessary for operational control. During operation, RCPs circulate pressurized fluid through the RPV and RCL. The fluid, which serves as a coolant, moderator, and solvent for boric acid, is heated as it passes through the nuclear core. The fluid in each loop flows from the RPV through the hot leg and into the SG, where heat is transferred to the steam supply system for electrical power generation. The fluid flows from the SG to the RCP in the crossover leg and then is pumped back into the RPV in the cold leg. The hot legs, crossover legs, and cold legs of the loop comprise the RCL piping. The RPV, SG, and PZR safe-end nozzle weld to the RCS piping is a similar metal weld and is included in the scope of this evaluation because the stainless steel (piping) to carbon steel (equipment) bimetallic weld is part of the equipment design and analysis.

On the basis of the intended functions previously set forth, the Class 1 portions of the auxiliary piping systems that were identified in the report as being within the scope of license renewal and requiring AMR are described below. It was also noted in the report that each plant may have additional specific commitments to NRC to increase or decrease the scope of license renewal.

- PZR surge line from one RCL hot leg to the PZR vessel inlet/outlet nozzle

- PZR spray lines from the reactor coolant cold legs, including the PZR spray scoop, to the spray nozzle on the PZR vessel

- RTD bypass lines, including RTD scoops, direct immersion RTDs, and the RTD manifolds

- Loop bypass lines

- PZR safety and relief lines from nozzles on top of the PZR vessel up to and through the power-operated PZR relief valves and PZR safety valves

- Class 1 portions of seal injection water and labyrinth differential pressure lines to or from the RCP inside the reactor building

- Reactor vessel head vent lines

- Charging line and alternate charging line from the Class 1 system isolation valves up to the branch connections on the RCL

- Letdown line and excess letdown line from the branch connections on the RCL to the

- Class 1 system isolation valve

- Residual heat removal (RHR) lines to or from the RCLs up to the designated Class 1 check valve or isolation valve

- High-head and low-head safety injection lines from the Class 1 check valve to the RCLs

- Accumulator lines from the designated Class 1 check valve to the RCLs

- Loop fill, loop drain, sample (including the sample scoop), and instrumentation lines to or from the designated Class1 isolation valve to or from the RCLs

- Auxiliary spray line from the Class 1 isolation valve to the PZR spray line header

- Sample lines from the PZR to the Class 1 isolation valve

- Boron injection lines from the designated Class 1 check valve to the RCL

The following associated pressure boundary components of Class 1 piping that are within the scope of license renewal and are subject to AMR were also identified:

NOZZLES AND SPECIAL NOZZLE ITEMS

In all of the lines previously described, the nozzle from the Class 1 component is considered part of the Class 1 component. For example, the reactor vessel head vent nozzle is part of the RPV, and the PZR surge nozzle on the hot leg is part of the hot leg. The nozzles that are included are as follows:

- Wide-range thermowells (Class 1 with no fluid system safety class interface)
- RTD fast-response thermowells with and without scoop (Class 1 with no fluid system safety class interface)
- Sample scoop and PZR spray scoop
- Three-inch and larger nozzle with thermal sleeve
- Two-inch and smaller nozzle with thermal sleeve
- Three-inch and larger nozzle without thermal sleeve
- Two-inch and smaller nozzle without thermal sleeve
- Forty-five-degree accumulator nozzles

Where installed, the thermal sleeve, thermowells, and scoop are considered in the design analysis of the nozzle.

BRANCH LINE RESTRICTORS

The scope of this report only addresses the Class 1 portion of the instrument connections and branch lines. Several instrument connections and some branch lines of the RCS are equipped with 3/8-inch-diameter flow restrictors. These restrictors limit the maximum flow through a broken line to a value below the makeup capability of the chemical and volume control system (CVCS). By providing the flow restrictions, the safety classification of the lines is downgraded from Safety Class 1 to Safety Class 2.

The staff has concluded that in addition to the pressure boundary function, the flow restrictors have an additional function, and that the effects of aging must be adequately managed so that every intended function of the component will be maintained (see Section 3.1 of this SE).

VALVES

The aging effect of the pressure boundary valve body is considered in this evaluation. The valve types include check valves, manual valves, pneumatic valves (air-operated valve), motor-operated block valves, solenoid-operated valves, and safety valves. The evaluation considers the effects of aging on the pressure boundary functions associated with the valve bodies consistent with the requirements of 10 CFR 54.21(a)(1)(i).

Valve bodies and bonnets that form part of the pressure boundary are classified as long-lived passive components and their pressure-retaining function will be addressed in this evaluation. Valve operators, discs, and seats are classified as active components and thus are not considered in this evaluation. The functions of valve operators, discs, and seats are periodically tested to ensure their functions are maintained.

<u>PZR Safety Valves</u>: The PZR safety valves are of the totally enclosed pop type and are spring-loaded and self-actuating with backpressure compensation features. These valves provide overpressure protection for the RCS and are sized to limit system pressure to below 110 percent of the system design pressure. In addition, these valves are set to the system design pressure, which is typically 110 percent of the operating pressure. The boundary between the piping and the safety valve is a flanged connection.

Power-Operated Relief Valve (Air-Operated Valve): The power-operated (pneumatic) relief valve (PORV) limits system pressure during large system transients. The valves are operated automatically from a pressure-sensing system or manually from the control room. The valves are designed to limit PZR pressure to a value below the high-pressure trip setpoint for all design transients up to and including the design percentage step load decrease, with steam dump but without reactor trip. The valves are also used with the cold overpressure mitigation system to control pressure during cooldown. PORVs have two valves in parallel to ensure that either can perform the relief function.

Head Vent Valves: The solenoid-operated reactor head vent valves are used to remove non condensable gases or steam from the reactor vessel head to mitigate potential inadequate core cooling events or impaired natural circulation resulting from the accumulation of non condensable gases.

Motor-Operated Block Valves: Motor-operated block valves are installed on lines where it is possible to have flow out of the RCS, such as RHR suction, letdown, and PORVs. The typical valve arrangement consists of two valves in series that stop flow by closing either valve. These valves provide a pressure boundary to prevent the flow of fluid out of the RCS.

Check Valves - Interconnecting Systems: Interconnecting system check valves are used to allow flow of fluid from systems required to operate in support of plant operations or an emergency situation and to prevent the backflow of reactor coolant into the support system. The check valves serve as a boundary by preventing flow out of the system.

Loop Stop Valves: Some RCL designs include loop isolation stop valves to isolate the RCLs, SG, and RCP from the RPV. During normal operation, these valves are in the open position. Although some plants have these valves, none are currently licensed to operate with the SG and RCP out of service.

THERMAL BARRIER AND RCP SEALS

The aging effect of the pressure boundary RCP casing is considered in this evaluation. In addition to the RCP casing's being a part of the Class 1 pressure boundary, the tubes of the thermal barrier heat exchanger within the RCP are considered to be part of the pressure boundary. The aging processes affecting stainless steel tubes are essentially the same as the balance of the Class 1 piping and are discussed in that context in this report.

RCP seals are also part of the pressure boundary. During normal operations, Class 1 seal water injection lines inject approximately 8 gallons per minute (gpm) into the No.1 seal area. This flow splits, with 5 gpm going into the RCS and 3 gpm bypassing and cooling the No. 1 seals. In the event charging flow is lost and the thermal barrier heat exchanger is functioning, the seal will leak cool water at 3 gpm. However, this leak will be reactor coolant water rather than charging water. The 3 gpm is within the normal reactor coolant makeup capacity. If both the charging flow and the component cooling flow are lost, the 3-gpm leakage will be hot water that will have a deleterious effect on the RCP seals. These combinations of RCP seal flow configurations are considered operating modes and not aging effects, and thus were not discussed further in the report. However, because RCP seals perform a pressure boundary function they were considered in the WOG's AMR and evaluated by the staff in Section 3.1 of this report.

2.2 Effects of Aging

Section 2.6 of the topical report lists the following aging effects that WOG considers potentially significant for the RCS piping and associated components:

- Fatigue-related cracking for fatigue-sensitive items
- Cracking and material degradation due to corrosion/stress-corrosion cracking
- Cracking due to irradiation embrittlement
- Thermal aging-related cracking of austenitic steel static castings
- Material wastage due to erosion and erosion/corrosion

- Material loss caused by wear of the RCP and Class 1 valve closure elements

- Loss of bolt preload due to creep or stress relaxation of bolted RCP and Class 1 valve closures

The staff notes that cracking is not caused by either irradiation embrittlement or thermal aging. Rather, these mechanisms cause a reduction in the fracture toughness of the material.

Section 3.0 of the topical report describes the AMR. The WOG review included operating experience of the RCS piping relating to the effects of aging. A summary of the identified potential aging effects is provided in Table 3.17 of the report. The table lists the following as potential effec's of aging for the specific RCS piping components:

Component	Potential Effects of Aging
Piping	Fatigue cracking Thermal aging of cast stainless steel Loss of material from corrosion and wear
Valve bodies	Fatigue cracking Thermal aging of cast stainless steel Loss of material from corrosion and wear
RCP casings	Thermal aging of cast stainless steel Loss of material from corrosion and wear
Closures, flanges, and bolting	Fatigue cracking (flange, flange bolts, and RCP closure) Loss of material from corrosion and wear Loss of bolting preload

2.3 Aging Management Programs

Section 3.4 of the topical report identifies the following aging effects that need a specific aging management program (AMP) to manage these aging effects during an extended period of operation:

- Fatigue-related cracking for fatigue-sensitive items
- Thermal aging-related cracking of austenitic stainless steel castings
- Material loss caused by wear of RCP and Class 1 valve closure elements
- Loss of bolt preload due to stress relaxation of bolted RCP and Class 1 valve closures

Section 4.0 of the topical report describes the options for managing these aging effects during an extended period of operation. The report lists seven proposed AMPs. Two of these rely on existing programs:

- AMP for wear of enclosures (AMP-3.1) relies on the American Society of Mechanical Engineers Boiler and Pressure Code (ASME Code) Section XI in-service inspection (ISI)

- AMP for stress relaxation of bolts (AMP-3.2) relies on the ASME Code Section XI Class 1 ISI, supplemented by plant commitments in response to NRC Generic Letter (GL) 88-05 (Reference 5) on boric acid corrosion

Three of the proposed AMPs (AMP 3.3 through AMP 3.5) address fatigue-sensitive components. The remaining proposed programs (AMP-3.6 and AMP-3.7) address thermal aging of stainless steel castings.

2.4 Time-Limited Aging Analyses

Section 2.5 of the topical report identifies the following TLAAs that are applicable to the piping and associated components:

- Fatigue

- Leak-before-break evaluations

Section 3.0 of the report presents WOG's proposed AMPs to address each TLAA. The license renewal applicant should provide a summary description of the programs and evaluations of TLAAs in the FSAR supplement. **This is Renewal Applicant Action Item 2.**

3.0 STAFF EVALUATION

The staff reviewed the topical report and additional information submitted by WOG to determine if it demonstrated that the effects of aging on the RCS piping covered by the report will be adequately managed so that the components' intended functions will be maintained consistent with the CLB for the period of extended operation in accordance with 10 CFR 54.21(a)(3). This is the last step in the IPA described in 10 CFR 54.21(a).

Besides the IPA, Part 54 requires an evaluation of TLAAs in accordance with 10 CFR 54.21(c). The staff reviewed the topical report and additional information submitted by WOG to determine if the TLAAs covered by the report were evaluated for license renewal in accordance with 10 CFR 54.21(c)(1).

3.1 Components and Intended Functions

The staff reviewed Sections 1.0 and 2.0 of the subject topical report to determine whether there is reasonable assurance that the Class 1 piping and associated pressure boundary components and supporting structures within the scope of license renewal, and subject to AMR, have been identified in accordance with the requirements of 10 CFR 54.4 and 10 CFR 54.21(a)(1). This evaluation was accomplished as discussed below.

As part of the evaluation, the staff determined whether the applicant had properly identified the systems, structures, and components within the scope of license renewal and subject to an AMR, pursuant to 10 CFR 54.4(a) and 10 CFR 54.21(a)(1). The staff reviewed portions of a representative updated final safety analysis report (the UFSAR for Calvert Cliffs) for the Class 1 piping and associated pressure boundary components and compared the information in the UFSAR with the information in the report to identify those portions that the report did not identify as being within the scope of license renewal and subject to an AMR. The staff then reviewed structures and components that were identified as not being within the scope of license renewal. The staff requested that the WOG provide additional information and/or clarifications for a selected number of these structures and components to verify the following:

(1) that these structures and components do not have any of the intended functions delineated under 10 CFR 54.4(a), and

(2) for those structures and components that have an applicable intended function(s), verify that they either perform this function(s) with moving parts or a change in configuration or properties, or that they are subject to replacement based on a qualified life or specified time period, as described in 10 CFR 54.21(a)(1).

The staff also reviewed the UFSAR for any functions delineated under 10 CFR 54.4 (a) that were not identified as intended functions by the WOG to verify that all the systems, structures, and components having an intended function(s) were considered within the scope of the rule.

After completing its initial review, the staff issued RAIs regarding the Class 1 piping and associated pressure boundary components. In a conference call on June 25, 1999, (see WOG letter dated July 19, 1999, documenting the call), the WOG provided the staff with its responses to those RAIs.

In RAI 1, the staff requested the WOG to clarify items from Section 2.3.2.2, "Branch Line Restrictors" of the report. These items are discussed below.

The staff questioned whether the Class 2 pipes and the flow restrictors in Class 2 pipes should be within the AMR. The response from WOG was that WCAP-14575 covers only Class 1 piping and those flow restrictors installed in Class 1 piping. Class 2 piping and flow restrictors installed in Class 2 piping are not included in this report and are evaluated for AMR in a separate report.

The report has listed only one intended function for flow restrictors, which is the pressure boundary function, in accordance with 10 CFR 54.4(a)(1)(i). However, the report also indicates that the 3/8-inch flow restrictors are relied upon to limit mass flow rate during postulated breaks.

The staff requested WOG to explain why one of the intended functions of flow restrictors, which is to prevent or mitigate the consequences of design-basis accidents, was not identified as an intended function in accordance with 10 CFR 54.4(a)(1)(iii). The rule requires the applicant to demonstrate that the effects of aging will be adequately managed so that all the intended functions of a component will be maintained consistent with the CLB for the period of extended operation. Therefore, any structure and component that meet any of the scoping criteria under 10 CFR 54.4, that performs an applicable intended function(s) without moving parts or without a change in configuration or properties, and that are not subject to replacement based on qualified life or specified time period should be identified and listed in the report. WOG responded that the report states that "restrictors limit the maximum flow through a broken line to a value below the makeup capability of the CVCS." Therefore, any line break downstream of a flow restrictor would not be a design-basis accident, because of this design feature. WOG therefore concluded that the absence of a design-basis accident eliminated 10 CFR 54.4(a)(1)(iii) as a reason for including this flow restrictor function as an intended function. However, after discussions with the staff, the WOG modified Section 2.3.2.2 and the "summary" sections of the report. The WOG identified "limit flow due to a downstream break to a value less than the normal RCS makeup capability" as an applicable intended function for the flow restrictors. The WOG further stated that because the flow restrictor forms an integral part of the piping where it is installed, subsequent discussion of aging effects and aging management for the piping is applicable also to the flow restrictors.

In its report, Section 2.3.2.4, "Thermal Barrier and RCP Seals," the WOG states that " the RCP seals are replaceable components and, as such, are exempt from license renewal." The staff disagrees with this conclusion. As allowed by the rule under 10 CFR 54.21(a)(1)(ii), structures and components can be excluded from AMR if they are replaced based on qualified life or specified time period. Therefore, for the staff to concur with the generic exclusion of RCP seals from an AMR, the WOG needs to provide a description, if appropriate, of a replacement program that is based on the qualified life or specified time period for these components.

In response to the staff's request for additional information (RAI 2), the WOG stated that RCP seals are a highly visible and closely monitored element of the RCS. Unlike other parts of the system, they do not maintain a pressure boundary but rather allow controlled leakage, which is acknowledged in plant technical specifications. This leakoff is closely monitored in the control room, and a high leakoff flow is alarmed as an abnormal condition requiring corrective action. Certain parts of the RCP seal "package" (e.g., backup seals) are subject to wear, and these parts are frequently replaced, as are installed O-rings. The main RCP seal is routinely inspected during plant outages on the basis of the manufacturer's recommendations and is replaced on the basis of either the results of that inspection or on leakoff performance during operation. The RCP seal was never intended to be a long-lived (life of the plant) component, although the specific time period for replacement of the seals will vary between plants, depending on individual operating practices and experience. The usual period ranges between 3 and 6 fuel cycles of operation. Although the WOGs description of the RCP seal replacement activities did not include a qualified life or specified time period, it did include a description of a replacement program based on performance and condition monitoring activities that provide reasonable assurance that the intended function of the RCP seals will be maintained in the period of extended operation. In the SOC, 60 FR 22478, the Commission allows an applicant to provide a site-specific justification for the use of performance and condition monitoring to provide the necessary reasonable assurance. Although the staff cannot generically exclude RCP seals from an AMR for all applicable Westinghouse plants, an applicant can submit a description of its performance and condition monitoring activities for RCP seals to exclude these components from an AMR. In general, if an applicant's program consists of the performance and condition monitoring activities described above, and the plant operating experience demonstrates the effectiveness of these activities, the staff will consider excluding these components from an AMR.

On the basis of the staff's review of the information provided in Sections 1.0 and 2.0 of the subject topical report, the supporting information in the UFSAR, and WOG's response to the staff's RAIs, the staff did not find, with the exception of the items previously discussed, any omissions in the report and, therefore, concludes that there is reasonable assurance that the report adequately identified those portions of the Class 1 piping and associated pressure boundary components that fall within the scope of license renewal and are subject to an AMR in accordance with 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

3.2 Effects of Aging

As indicated in Section 2.2 of this SE, the effects of aging evaluated in WCAP-14575 are as follows:

- Fatigue-related cracking
- Corrosion/stress-corrosion cracking
- Reduction of fracture toughness (irradiation embrittlement and thermal aging of cast stainless steel)
- Loss of material (erosion, erosion/corrosion, and wear)
- Loss of bolting preload (creep and stress relaxation of mechanical closures)

Westinghouse reviewed these effects of aging for their specific applicability to the RCS piping, valve bodies, RCPs, and bolting. After reviewing the report and published aging research results, the staff agrees that WOG's report properly identified the potential aging effects for the RCS piping components. A discussion of the specific aging effects on the various RCS components follows.

Westinghouse reviewed information from operating experience of the RCS piping relating to the effects of aging. Although the effects of aging were correctly identified by Westinghouse, the staff found that generic communications were not discussed in the report, for example, Bulletin 82-02 on bolting and GL 85-20 on thermal sleeves. In its response to RAI 5, Westinghouse indicated that Section 3.1 of the report would be revised to describe the process used by WOG to review generic communications. Also, it stated that an updated review would be performed to capture any additional items that occurred, or were missed, since the original review was performed. At this time, this updated version is not available and thus was not reviewed by the staff. The renewal applicant should complete the updated review of generic communications and capture any additional items not identified by the original review. **This is Renewal Application Action Item 3.**

3.2.1 Fatigue

The report indicates and the staff agrees that degradation sustained from the effects of fatigue was determined to be potentially significant for the fatigue-sensitive Class 1 piping and piping components, the Class 1 valve bodies greater than 4-inch nominal pipe schedule, and the RCP pressure boundary closure components. This determination has its basis in analysis, test, and experience. WOG proposed programs to manage fatigue-sensitive components during the period of extended operation. The staff's assessment of these programs is contained in Section 3.3.2 of this SE.

3.2.2 Corrosion/Stress Corrosion

The topical report indicates that operating experience has shown that general corrosion and stress corrosion are not a concern for primary loop materials used in Westinghouse NSSSs. NSSS components are fabricated from austenitic stainless steel. The staff agrees with the WOG assessment that austenitic stainless steel is not susceptible to corrosion and stress corrosion in pressurized water reactor (PWR) primary coolant. However, austenitic stainless steel is susceptible to stress-corrosion cracking if the outside surface of the material comes in contact with halogens. Therefore, applicants for license renewal must provide a description of all insulation used on austenitic stainless steel NSSS piping to ensure the piping is not susceptible to stress-corrosion cracking from halogens. **This is Renewal Applicant Action Item 4.**

The topical report identifies wastage of external surfaces caused by the leakage of borated water as a concern for RCS components. Degradation sustained from the effects of corrosion was determined to be potentially significant near the bolted or flanged connections that may be subject to boric acid corrosion from leaking primary coolant. WOG indicated that this could be managed by the existing ISI program. The staff's assessment of this program is contained in Section 3.3.1 of this SE.

3.2.3 Loss of Material

The report indicates that the effect of erosion is not considered significant for the Class 1 piping and associated components on the basis of the following considerations:

- The fluid flow velocity is relatively low in the Class 1 piping and components.
- The water is filtered before injection into the primary system.
- The operating pressure of a PWR precludes cavitation erosion.
- The inside diameter of the primary loop is 100-percent machined or ground.

The staff agrees with the WOG assessment that erosion is not significant for Class 1 piping and associated components.

Mechanical wear affects RCP and Class 1 valve bolted closure elements, such as closure flanges and bolting, because of relative motion caused by loss of bolt preload or by infrequent disassembly and reassembly. WOG indicated that this concern could be managed by the existing ISI program. The staff's assessment of this program is contained in Section 3.3.1 of this SE.

3.2.4 Reduction of Fracture Toughness

The topical report indicates that thermal aging-related cracking of austenitic steel castings are aging effects that WOG considers potentially significant for the RCS piping and associated components. However, thermal aging does not cause cracking, it causes a reduction in fracture toughness of the material. As discussed below, the reduction in fracture toughness results in a reduction in the critical flaw size that could lead to component failure.

The report indicates that irradiation embrittlement is not a concern for the RCS piping components because the expected neutron fluence is much less than the threshold level at which changes in properties of the materials would occur. The staff agrees with this conclusion.

The staff concurs with Westinghouse that thermal aging is a potential aging effect on cast austenitic stainless steel (CASS) components. The thermal aging effect is a reduction in fracture toughness of CASS components. This reduced fracture toughness causes a reduction in the critical flaw size for the component, which is defined as the size flaw that could lead to failure. The staff agrees with Westinghouse that welds in the primary loop also thermally age but usually respond more slowly because of low ferrite. WOG proposed programs to manage the effects of thermal aging of CASS components during the period of extended operation. The staff's assessment of these programs is contained in Section 3.3.3 of this SE.

3.2.5 Loss of Closure Integrity

The report indicates that creep is not a concern for austenitic alloys below 1000 °F. The staff agrees with this conclusion. However, the report does indicate that loss of preload can occur from stress relaxation on the RCP and Class 1 valve bolted closures. WOG indicated that this could be managed by the existing ISI program. The staff assessment of this program is contained in Section 3.3.1 of this SE.

3.3 Aging Management Programs

Table 4-1 of the report lists the six attributes that form the basis for the existing and additional AMPs. These attributes include the scope of the program, the surveillance techniques used to detect aging effects, the frequency of the surveillance, the acceptance criteria to determine when corrective actions are necessary, the corrective actions, and confirmation techniques. WOG indicated, in Section 4.0 of the topical report, that the plant-specific details of the AMPs will be developed during the preparation of license renewal applications and that all six attributes may not be necessary for an AMP. Therefore, license renewal applicants should describe how each plant-specific AMP addresses the following 10 elements: (1) scope of the program, (2) preventive actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience. **This is Renewal Applicant Action Item 5.**

WOG evaluated existing programs and found them adequate, with a few exceptions, in managing the effects of aging so that the intended function of the RCS piping components will be maintained consistent with the CLB for any period of extended operation. As described in Section 2.3, the existing programs include ASME Code Section XI ISI programs and licensee commitments in response to NRC generic communications. These existing programs are used to address wear of closures and stress relaxation of bolts. WOG proposed additional programs to address fatigue and thermal aging.

The staff reviewed the existing and additional programs and concluded that the license renewal applicant should provide a new evaluation of CASS components to the criteria in Electric Power Research Institute (EPRI) TR-106092 with additional criteria discussed in Section 3.3.3 of this SE (see Renewal Applicant Action Item 7). The staff believes that the license renewal applicant should propose to perform additional inspection of small-bore RCS piping, that is, less than the 4-inch-size, for license renewal. These additional examinations would provide assurance that the potential for cracking of small-bore RCS piping is adequately managed during the period of extended operation. **This Is Renewal Applicant Action Item 6.**

3.3.1 Wear of Closures and Stress Relaxation of Bolts

WOG relies on existing ASME Code Section XI ISI to manage wear and stress relaxation for the RCP and Class 1 valve bolted closure elements. The elements of these programs are shown in Tables 4-2 and 4-3 of the topical report. The topical report describes the ASME Code Section XI Class 1 ISI program "Examination Categories B-G-1, B-G-2, and B-P," the response to GL 88-05, including pump and valve inservice testing, as necessary to manage the effects of aging of the RCS bolted closure elements during the period of extended operation to maintain the reactor coolant pressure boundary.

ASME Code Section XI "Examination Category B-P" covers system leakage and hydrostatic tests. "Examination Categories B-G-1 and B-G-2" are as follows:

"Examination Category"	Component description	Size (Inches)	Examination
B-G-1	Pressure-retaining bolting	≥2	Volumetric Visual "VT-1" of associated surfaces
B-G-2	Pressure-retaining bolting	<2	Visual "VT-1" of pump and valve studs and bolts

These examinations and tests are carried out at each inspection interval of the plant's ISI program or at each refueling outage in the case of system leakage tests. Valve bolting examination is limited to bolting on valves that are selected for examination under Examination Category B-M-2. "Visual VT-1" examination is conducted to determine the condition of the component or surface examined, including such conditions as cracks, wear, corrosion, erosion, or physical damage on the surfaces of the components. Flaws detected in "Examination Categories B-G-1 and B-G-2" may be acceptable for continued service if they meet the acceptance standards in IWB-3517.

AMP-3.1 is applicable to wear of bolted closures. The staff finds the ASME Code Section XI examination proposed by WOG adequate in managing potential wear of bolted closures because the closure surfaces and bolts will be examined when the closures are disassembled for inspections. Mechanical closure integrity can also be monitored through "Examination Category B-P" system leakage and hydrostatic tests.

AMP-3.2 is applicable to loss of preload by stress relaxation. The program relies on ASME Code Section XI in-service examinations and tests supplemented by the boric acid wastage surveillance programs implemented by licensees in response to NRC GL 88-05 as necessary in managing the potential loss of material of low-alloy steel bolting during the period of extended operation. The staff finds the ASME Code Section XI examination and tests supplemented by programs committed to by licensees in response to GL 88-05 to be acceptable for managing the aging effect of loss of material for low-alloy steel bolting within the scope of this report during the period of extended operation. Licensees programs and actions in response to GL 88-05 are documented in NUREG/CR-5576 "Survey of Boric Acid Corrosion of Carbon Steel Components in Nuclear Plants".

3.3.2 Fatigue

WOG presented three AMPs for fatigue. AMP-3.3 covers ASME Code Class 1 piping, valve bodies 6 inches and larger, and RCP closure fatigue-sensitive locations. AMP-3.4 covers fatigue- sensitive RCS piping designed to United States of America Standard (USAS) B31.1. AMP-3.5 covers valve bodies 6 inches and larger and the RCP closure. WOG presented several options to manage fatigue for each program. The staff's evaluation of these options is discussed below.

WOG evaluated the RCS components and summarized the fatigue-sensitive locations in Table 4-4 of the report. For the fatigue-sensitive locations identified in Table 4-4, WOG proposed an AMP. According to WOG, the objectives of the fatigue management program are to

(1) Maintain the CLB for fatigue for the current license renewal terms by justifying that existing fatigue analyses are valid or by extending the period of evaluation of the analyses so they remain valid or

(2) Justify that the effects of fatigue will be adequately managed for the license renewal term if the applicant cannot or chooses not to justify or extend the existing fatigue analyses.

WOG proposed AMP-3.3 for the ASME Code Class 1 components and AMPs 3.4 and 3.5 for USAS B31.1 designs. For each AMP, WOG proposed several options to meet the above objectives. In addition to program scope, each AMP specifies surveillance techniques (parameters monitored), monitoring frequency, acceptance criteria, corrective actions, and confirmation techniques. The AMPs present four alternatives for demonstrating the adequacy of the components for the extended period of operation. These alternatives are discussed in Section 4.2.1 of the topical report.

The first alternative for Class 1 components (Step 1A of the proposed program) involves demonstrating that the CLB analysis will remain valid through the period of extended operation by ensuring that the number of transients assumed in the design is not exceeded during the period of extended operation or recalculating the fatigue usage using operating experience. The first alternative for USAS B31.1 designs (Step 1B of the proposed program) involves assessing the thermal stresses during the period of extended operation. The process is described in Section 4.2.1.2 of the topical report. The process for evaluating USAS B31.1 designs involves several steps. The steps provide the following alternatives to qualify the component : (1) demonstrate that the design basis cycles of transient operation will not be exceeded during the period of extended operation, or (2) demonstrate the expansion stresses meet a reduced stress limit to account for a projected 50% increase in number of transient cycles, or (3) perform detailed analysis of the component for the period of extended operation considering design or actual operating cycles to demonstrate that either the USAS B31.1 expansion stress limits will not be exceeded or the ASME Class 1 fatigue limits will not be exceeded. The staff finds that the options specified in the first alternative provide acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).

The second alternative (Step 2 of the proposed program) allows the component to be included in an existing or enhanced ASME Code Section XI ISI program with ISI procedures adequate to detect flaw sizes that can be shown to not propagate to failure between inspection intervals. In

RAI 2a, the staff requested that WOG discuss how this alternative provides assurance that the licensing basis criteria has been met at a facility. In response to the RAI, WOG proposed to modify the topical report to provide an additional discussion of this alternative. This alternative would allow the CLB fatigue cumulative usage factor (CUF) to be exceeded during the period of extended operation. The staff has not endorsed this position on a generic basis at this time. An applicant wishing to pursue this alternative would have to obtain staff review and approval on a case-by-case basis.

The third alternative (Step 3) provides for an augmented inspection program to determine the acceptability of the components for the period of extended operation. The alternative allows for the use of a flaw tolerance evaluation or a leak-before-break analysis to demonstrate the adequacy of the components. In RAI 2b, the staff requested that WOG discuss how this alternative provides assurance that the licensing basis criteria has been met at a facility. In response to the RAI, WOG proposed to modify the topical report to provide an additional discussion of this alternative. This alternative would allow the licensing basis CUF to be exceeded during the period of extended operation. The staff has not endorsed these positions at this time. The staff notes that the WOG reference to a leak-before-break analysis only involves the use of the analysis methodology. The staff would not approve the use of leak-before-break methodology to eliminate postulated pipe breaks under General Design Criterion (GDC) 4 for locations where the CLB CUF may be exceeded during plant operation. An applicant wishing to pursue the third alternative would have to obtain staff review and approval on a case-by-case basis.

The fourth alternative (Step 4) is to replace the component if the licensing basis fatigue criteria cannot be met during the period of extended operation. The staff finds that this alternative satisfies the requirements of 10 CFR 54.21(c)(1)(iii).

3.3.3 Thermal Aging of CASS Components

A recent EPRI report provides a framework for effective management of thermal aging of CASS components (Ref. 6), with appropriate modifications. This framework consists of a susceptibility screening process and an examination (ISI) flaw evaluation process. The susceptibility screening process is used to determine which CASS components are potentially susceptible to thermal aging and hence require additional evaluation or examination.

Susceptibility Screening Method

Determination of the susceptibility of CASS components to thermal aging can use a screening method based upon the Molybdenum (Mo) content, casting method, and δ-ferrite content. (Alternatively, components can be assumed as "potentially susceptible" without considering such screening.) Specific acceptable screening criteria are outlined in Table 1 and are applicable to all primary pressure boundary components constructed from SA-351 Grade CF3, CF3A, CF8, CF8A, CF3M, CF3MA, or CF8M, with service conditions above 250 °C (482 °F). The δ-ferrite content of the component can be determined from measurements or calculations. Note that calculations of δ-ferrite should use Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values).

The significance of finding a particular component not susceptible or potentially susceptible is described below for each component type. The examination requirements for each component type are given in Table 2. In addition, acceptable flaw evaluation procedures are described.

Table 1: CASS Thermal Aging Susceptibility Screening Criteria

Mo Content (Wt. %)	Casting Method	δ-Ferrite Level	Susceptibility Determination
High (2.0 to 3.0)	Static	≤ 14 %	Not susceptible
		> 14 %	Potentially susceptible
	Centrifugal	≤ 20 %	Not susceptible
		> 20%	Potentially susceptible
Low (0.5 max.)	Static	≤ 20 %	Not susceptible
		> 20 %	Potentially susceptible
	Centrifugal	ALL	Not susceptible

Table 2: Examination Requirements for CASS Components

Component	Grouping	Not Susceptible	Potentially Susceptible

Piping (Base Metal)	NPS ≥ 4 in.	None	Inspection or evaluation
	NPS < 4 in.	None	Inspection or evaluation
Valve Bodies (Base Metal)	NPS ≥ 4 in.	ASME Code Section XI requirements	ASME Code Section XI requirements
	NPS < 4 in.	ASME Code Section XI requirements	ASME Code Section XI requirements
Pump Casings (Base Metal)	NPS ≥ 4 in.	ASME Code Section XI requirements	ASME Code Section XI requirements
	NPS < 4 in.	ASME Code Section XI requirements	ASME Code Section XI requirements

Current Inspection Requirements

Current inspection requirements in Table IWB-2500-1 of Section XI of the ASME Code for CASS components are as follows:

- Piping (Category B-J): Volumetric and surface examination of pressure-retaining welds for NPS ≥ 4 in.; surface examination of pressure-retaining welds for NPS < 4 in.

- Valve bodies (Categories B-M-1 and B-M-2): Visual VT-3 examination of internal surfaces and volumetric examination of pressure-retaining welds for NPS ≥ 4 in.; surface examination of pressure-retaining welds for NPS < 4 in.

- Pump casings (Categories B-L-1 and B-L-2): Visual VT-3 examination of internal surfaces and volumetric examination of welds

As described in Table 2, these requirements are sufficient in some cases for management of thermal aging even for components "potentially susceptible" to thermal aging, notably RCP casing and valve bodies. However, in the case of piping base metal the current ASME Code Section XI requirements may not be adequate and additional evaluation or examination is warranted as follows:

Piping (Base Metal)

Since the base metal of piping does not receive periodic inspection in accordance with Section XI of the ASME Code, the susceptibility of piping constructed from CASS should be assessed for each heat of material. Alternatively, an assumption of "potentially susceptible" can be assumed for each heat or specific heats.

If a particular heat is found to be "not susceptible," no additional inspections or evaluations are required because the material has adequate toughness.

If a particular heat is found or assumed to be "potentially susceptible" and subject to plausible degradation (e.g., thermal fatigue), aging management can be accomplished through volumetric examination or plant/component-specific flaw tolerance evaluation. The volumetric examination, using a method capable of detecting flaws in CASS components, should be performed on the base material of each heat, with the scope of the inspection covering the portions determined to be limiting from the standpoint of applied stress level, operating time, and environmental considerations. Alternatively, a plant-specific or component-specific flaw tolerance evaluation, using specific geometry and stress information, can be used to demonstrate that the thermally embrittled material has adequate toughness.

Valve Bodies and Pump Casings

Valve bodies and pump casings are adequately covered by existing inspection requirements in Section XI of the ASME Code, including the alternative requirements of ASME Code Case N-481 for pump casings. Screening for susceptibility to thermal aging is not required during the period of extended operation because the potential reduction in fracture toughness of these components should not have a significant impact on critical flaw sizes. Accordingly, the current ASME Code inspection requirements are sufficient.

AMP-3.7 provides aging management for RCP casings through the demonstration of compliance with Code Case N-481. The one-time fracture mechanics evaluation, specified in this AMP, must incorporate bounding material properties for the end of the period of extended operation.

Volumetric Examination

Current volumetric examination methods are not adequate for reliable detection of cracks in CASS components. If an acceptable method for volumetric examination of CASS components is developed, the performance of the equipment and techniques should be demonstrated through a program consistent with ASME Code, Section XI, Appendix VIII.

Flaw Evaluation

Flaws detected in CASS components should be evaluated in accordance with the applicable procedures of IWB-3500 in Section XI of the ASME Code. If the δ-ferrite content does not exceed 25 percent, then flaw evaluation would be in accordance with the principles associated with IWB-3640 procedures for submerged arc welds (SAWs), disregarding the ASME Code restriction of 20 percent in IWB-3641(b)(1). If the CASS material is "potentially susceptible" and the δ-ferrite content exceeds 25 percent, then flaw evaluation would be on a case-by-case basis using fracture toughness data supplied by the licensee, such as that published by Jayet-Gendrot, et al (Reference 7).

The license renewal applicant should address thermal-aging issues in accordance with the staff comments above, and revise AMP-3.7 as appropriate. **This is Renewal Applicant Action Item 7.**

3.4 Time-Limited Aging Analyses

TLAAs are defined in 10 CFR 54.3 as those licensee calculations and analyses that

1. involve systems, structures, and components within the scope of license renewal, as stated in 10 CFR 54.4(a);

2. consider the effects of aging;

3. involve time-limited assumptions defined by the current operating term, for example, 40 years

4. were determined to be relevant in making a safety determination;

5. involve conclusions or provide the bases for conclusions related to the capability of the system, structure or component to perform its intended functions, as stated in 10 CFR 54.4(b); and

6. are contained or incorporated by reference in the CLB.

Section 54.21(c)(1) requires the applicant to demonstrate that

1. the analyses remain valid for the period of extended operation;

2. the analyses have been projected to the end of the period of extended operation; or

3. the effects of aging on the intended functions(s) will be adequately managed for the period of extended operation.

The TLAAs evaluated in WCAP-14575 for the Class 1 piping are

1. Fatigue of metallic components
2. Leak-before-break evaluations.

3.4.1 Fatigue (Including Environmentally Assisted Fatigue)

Section 3.3 of WCAP-14575 describes the fatigue evaluation methodology for the RCS piping and associated components. The methodology depends on the component type and its design code. The design requirements are discussed in Sections 2.4.6, 2.4.7,and 3.2 of the topical report. The specific design criteria are discussed below.

Piping

Section III of the ASME Code was used for plants designed since 1971. USAS B31.7 was used for plants designed between 1969 and 1971. The design criterion for these codes involves calculating a specific quantity called the CUF. The fatigue damage caused by each thermal or pressure transient depends on the magnitude of the change in the stresses in the component caused by the transient. The CUF sums the fatigue resulting from each transient. The design criterion requires that the CUF not exceed 1.0. USAS B31.1 was used for plants designed before 1969. USAS B31.1 does not require an explicit fatigue analysis of local thermal stresses resulting from operational transients. Instead, the criterion requires a reduction in the allowable bending stress range if the number of full-range cycles of bending stress exceeds the value specified in the Code.

Valves

Section III of the ASME Code was used for plants designed since 1971. The Draft ASME Pump and Valve Code was used for plants designed between 1969 and 1971. The design criterion for these did not require a fatigue evaluation of valves 4 inches or less. A fatigue evaluation was required for larger size valves. Before 1969, valves were covered by USAS B31.1, which did not require a fatigue analysis of valves.

RCP Casings

According to WOG, detailed fatigue analyses of RCP casings were not required because the ASME Code conditions specified in NB-3222.4(d)(1) through (6) were met. The ASME Code does not require an explicit fatigue analysis if these limits are satisfied.

RCP parts other than the casings are discussed in Section 3.3.5 of the report. According to WOG, some of the seal injection and component cooling water nozzles have high fatigue usage factors and are, consequently, considered fatigue-sensitive areas.

WOG indicated that Westinghouse maintains a generic fatigue database for the Class 1 piping systems that have been evaluated for fatigue. WOG increased the calculated CUF for each component in these systems by a factor of 1.5 to account for 60 years of design cycles. If the subsequent CUF was less than 1.0, WOG considered the component not to be fatigue-sensitive. The results of these evaluations are summarized in Tables 3-2 through 3-16. For the remaining components, either a further analysis is necessary or the component needs an AMP. WOG did not evaluate all valve bodies and RCPs. As a consequence, WOG identified these components as requiring a plant-specific evaluation. Therefore, applicants for license renewal should perform additional fatigue evaluations or propose an AMP to address these remaining components. **This is Renewal Applicant Action Item 8.**

Section 3.1.2 of the topical report contains a discussion of environmentally assisted fatigue of metal components. Current test data indicate that the design fatigue curves of the ASME Code may not be conservative for nuclear power plant primary system environments. The ASME fatigue curves were developed from laboratory specimens tested in air at room temperature. The current test data indicate there could be a significant reduction in the fatigue life of metal components in a reactor primary system environment. The staff addressed the issue of environmentally assisted fatigue in Generic Safety Issue (GSI) 166, "Adequacy of Fatigue Life of Metal Components." The staff's recommendations are contained in SECY-95-245. In SECY-95-245, the staff did not recommend the backfit of new environmental fatigue curves to operating plants. This recommendation was based, in part, on conservatism identified in the existing fatigue analyses and on a risk assessment considering a 40-year plant design life.

A further assessment was performed under GSI-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life." In SECY-95-245, the staff indicated that it would consider whether license renewal applicants need to evaluate a sample of components with high-fatigue usage factors, using the latest available environmental fatigue data. The staff further indicated that if the GSI has not been resolved before the issuance of a renewed license, the applicant should submit its technical rationale for concluding the effects of fatigue are adequately managed for the extended period or until the resolution of the GSI becomes available (60 FR 22484, May 8, 1995). The staff recommendation for the closure of GSI-190 is contained in a December 26, 1999, memorandum from Ashok Thadani to William Travers. The staff recommended that licensees address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. The evaluation of a sample of components with high-fatigue usage factors using the latest available environmental fatigue data is an acceptable method to address the effects of the coolant environment on component fatigue life.

Section 4.2.1 of the topical report contains further discussion of environmental effects on the fatigue life of components. In RAI 2c, the staff requested WOG to clarify the method by which the staff's recommendation is addressed by the AMP. In response, WOG indicated that the report will be modified to incorporate the revised proposed industry position on fatigue. This revised position considers environmental effects for an extended period of operation. The staff has not yet endorsed the industry position on fatigue (References 8, 9, 10). Therefore, a renewal applicant will be required to address the GSI-190 closure recommendation on a case-by-case basis until the staff endorses the industry position. **This is Renewal Applicant Action Item 9.**

3.4.2 Leak-Before-Break Evaluations

In Section 3.3.7 of the topical report, WOG indicated that leak-before-break (LBB) evaluations have been incorporated into the current licensing basis (CLB) for most Westinghouse plants. These evaluations followed, in general, the recommendations and criteria proposed in NUREG-1061, Volume 3 (Reference 11), and have been applied to both the main coolant loop piping as well as the Class 1 auxiliary lines at some plants. WOG proposed AMP-3.6 to reevaluate the LBB status of those CASS piping components that had been previously approved for LBB during the current licensing period. According to WOG, the LBB limiting locations must be based on appropriate material properties for base and weld metals, including any long-term material degradation effects such as thermal aging embrittlement. Therefore, WOG proposed AMP-3.6 to address the impact of thermal aging embrittlement on the LBB evaluations for the period of extended operation. Previously, in Section 3.3.3, the staff identified Renewal Application Ac.ion Item 9 regarding the thermal aging embrittlement evaluation.

The staff has reviewed AMP-3.6 and has concluded that it is, in general, an acceptable proposal for confirming the LBB status of CASS components through the period of extended operation. However, two items from Table 4-9 on AMP-3.6 require revision. First, in order to maintain conformance with the NRC staff's guidance in NUREG-1061, Volume 3, an additional assessment of the margin on the loads is required. This is addressed as item (i) in Section 5.2 of NUREG-1061, Volume 3.

Second, the corrective actions proposed in Table 4-9 in the event that the acceptance criteria are exceeded are not sufficient to reestablish its LBB status. If the CASS component is repaired or replaced per ASME Code, Section XI IWB-4000 or IWB-7000, a new LBB analysis based on the material properties of the repaired or replaced component (and accounting for its thermal aging through the period of extended operation, as appropriate), is required to confirm the applicability of LBB. The inservice examination/flaw evaluation option is, per the basis on which the NRC staff has approved LBB in the past, insufficient to reestablish LBB approval. The license renewal applicant should revise AMP-3.6, accordingly. **This is Renewal Applicant Action Item 10.**

4.0 CONCLUSIONS

The staff has reviewed the WOG topical report (Reference 6) and additional information submitted by WOG. On the basis of its review, the staff concludes that the WOG topical report provides an acceptable demonstration that the aging effects of RCS components within the scope of this topical report will be adequately managed for the WOG license renewal member plants, with the exception of the noted renewal applicant action items, so that there is reasonable assurance that the RCS components will perform their intended functions in accordance with the CLB. The staff also concludes that upon completion of the renewal applicant action items set forth in Section 4.1 herein, the WOG topical report provides an acceptable evaluation of TLAAs for the RCS components in the WOG license renewal member plants for the period of extended operation.

Any WOG member plant may reference this topical report in a license renewal application (LRA) to satisfy the requirements of (1) 10 CFR 54.21(a)(3) for demonstrating that the effects of aging on the RCS components within the scope of this topical report will be adequately managed and (2) 10 CFR 54.21(c)(1) for demonstrating that appropriate findings be made regarding evaluation of TLAAs for the RCS components for the period of extended operation. The staff also concludes that upon completion of the renewal applicant action items set forth in Section 4.1 herein, referencing this topical report in an LRA and summarizing in a final safety analysis report (FSAR) supplement the AMPs and the TLAA evaluations contained in this topical report will provide the staff with sufficient information to make the necessary findings required by Sections 54.29(a)(1) and (a)(2) for components within the scope of this topical report.

4.1 Renewal Applicant Action Items

The following are license renewal applicant action items to be addressed in the plant-specific LRA when incorporating the WOG topical report in a renewal application:

1 The license renewal applicant is to verify that its plant is bounded by the topical report. Further, the renewal applicant is to commit to programs described as necessary in the topical report to manage the effects of aging during the period of extended operation on the functionality of the reactor coolant system piping. Applicants for license renewal will be responsible for describing any such commitments and identifying how such commitments will be controlled. Any deviations from the AMPs within this topical report described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor coolant system piping and associated pressure boundary components or other information presented in the report,

such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

2 Summary description of the programs and evaluation of TLAAs are to be provided in the license renewal FSAR supplement in accordance with 10 CFR 54.21(d).

3 The renewal applicant should complete the updated review of generic communications and capture any additional items not identified by the original review.

4 The license renewal applicant must provide a description of all insulation used on austenitic stainless steel NSSS piping to ensure the piping is not susceptible to stress-corrosion cracking from halogens.

5 The license renewal applicant should describe how each plant-specific AMP addresses the following 10 elements: (1) scope of the program, (2) preventive actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience.

6 The license renewal applicant should perform additional inspection of small-bore RCS piping, that is, less than 4-inch-size piping, for license renewal to provide assurance that potential cracking of small-bore RCS piping is adequately managed during the period of extended operation.

7 Components that have delta ferrite levels below the susceptibility screening criteria have adequate fracture toughness and do not require supplemental inspection. As a result of thermal embrittlement, components that have delta ferrite levels exceeding the screening criterion may not have adequate fracture toughness and do require additional evaluation or examination. The license renewal applicant should address thermal-aging issues in accordance with the staff's comments in Section 3.3.3 of this evaluation.

8 The license renewal applicant should perform additional fatigue evaluation or propose an AMP to address the components labeled I-M and I-RA in Tables 3-2 through 3-16 of WCAP-14575.

9 The staff recommendation for the closure of GSI-190 "Fatigue Evaluation of Metal Components for 60-Year Plant Life" is contained in a December 26, 1999, memorandum from Ashok Thadani to William Travers. The license renewal applicant should address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. The evaluation of a sample of components with high-fatigue usage factors using the latest available environmental fatigue data is an acceptable method to address the effects of the coolant environment on component fatigue life.

10 The license renewal applicant should revise AMP-3.6 to include an assessment of the margin on loads in conformance with the staff guidance provided in Reference 11. In addition, AMP-3.6 should be revised to indicate if the CASS component is repaired or replaced per ASME Code, Section XI IWB-4000 or IWB-7000, a new LBB analysis based on the material properties of the repaired or replaced component (and accounting for its thermal aging through the period of extended operation, as appropriate), is required to confirm the applicability of LBB. The inservice examination/flaw evaluation option is, per the basis on which the NRC staff has approved LBB in the past, insufficient to reestablish LBB approval.

5.0 REFERENCES

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DISCLAIMER OF RESPONSIBILITY

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EXECUTIVE SUMMARY

This report evaluates aging of the Class 1 piping and associated pressure boundary components that support the reactor coolant system (RCS) to ensure that the intended function will be maintained during an extended period of operation. Class 1 piping and associated components perform the intended function of ensuring the integrity of the reactor coolant pressure boundary.

Class 1 piping and associated components are subject to an aging management review because they perform an intended function, are passive, and are long-lived. This aging management review has identified aging effects and time-limited aging analyses and provided options that manage these effects, where necessary. When implemented, these options will demonstrate that the intended function will be maintained for the extended period of operation.

The scope of this report includes domestic commercial nuclear power plants with Westinghouse nuclear steam supply systems (NSSSs). Specifically for Class 1 piping and associated components, the scope is limited to:

- Class 1 piping
- Class 1 valve bodies
- Reactor coolant pump (RCP) casings
- Associated pressure boundary components

This evaluation was performed in support of the Westinghouse Owners Group (WOG) Life Cycle Management/License Renewal (LCM/LR) program.

Effects from all design limits, time-limited aging analyses (TLAAs), aging, and industry issues have been evaluated. Options to manage aging effects that degrade the intended function are provided. For Class 1 piping and associated components, the following aging effects require management:

- Fatigue-related cracking for fatigue-sensitive items
- Thermal aging-related cracking of statically cast austenitic stainless steel
- Material loss caused by wear of reactor coolant pump (RCP) and Class 1 valve closure elements
- Loss of bolt preload due to stress relaxation of bolted RCP and Class 1 valve closures

Options to manage aging that are part of current industry practice are provided in Section 4.1, and the effectiveness of all programs during an extended period of operation is justified.

Options to manage effects that are not part of current industry practice are provided in Section 4.2. Aging effects requiring additional activities are from fatigue and thermal aging.

In conclusion, this evaluation has shown that the intended function of Class 1 piping and associated components will be maintained by these options (when implemented) during an extended period of operation. In addition, the RCS intended function, supported by Class 1 piping and associated components will be maintained.

This approved version (WCAP-14575-A) incorporates the NRC Final Safety Evaluation and the WOG responses to NRC Requests for Additional Information.

TABLE OF CONTENTS

Section	Title	Page No.

TABLE OF CONTENTS (Continued)

LIST OF TABLES

LIST OF TABLES (Continued)

LIST OF FIGURES

ACRONYMS

AE	Architect engineer
AMAPA	Aging management activities and program attributes
AMP	Aging Management Program
ANL	Argonne National Laboratories
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
BIT	Boron injection tank
BNCS	Board of Nuclear Codes and Standards
BRS	Boron recycle system
BTRS	Boron thermal regeneration system
BWR	Boiling water reactor
CASS	Cast austenitic stainless steel
CCP	Centrifugal charging pump
CCW	Component cooling water
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
CTMT	Containment
CVCS	Chemical and volume control system
DNB	Departure from nucleate boiling
ECC	Emergency core cooling
ECCS	Emergency core cooling system
EPRI	Electric Power Research Institute
ESBU	Westinghouse Electric Company Energy Systems Business Unit
FAP	Fatigue Action Plan
GFDB	Generic Fatigue Data Base
gpm	Gallons per minute
Gr	Grade
GSI	Generic Safety Issue
GTAW	Gas tungsten arc welding
GTR	Generic technical report
HHSI	High head safety injection
HSLA	High strength low alloy
HX	Heat exchanger
IB	Information Bulletin
ID	Inside diameter
IGA	Intergranular attack
IG	Intergranular
IGSCC	Intergranular stress corrosion cracking
IN	Information Notices
IR	Industry Report
ISLOCA	Intersystem LOCA
LBB	Leak-before-break
LCM/LR	Life cycle management/license renewal
LER	Licensee event reports

ACRONYMS (Continued)

LHSI	Low head safety injection
LOCA	Loss-of-coolant accident
LP	Liquid penetrant
LWR	Light water reactor
MOV	Motor-operated valve
M/U	Makeup
NEI	Nuclear Energy Institute (formerly NUMARC)
NNS	Non-nuclear safety
NPS	Nominal pipe schedule
NRC	Nuclear Regulatory Commission
NSSS	Nuclear steam supply system
NUMARC	Nuclear Management and Resources Council
NUREG	Nuclear Regulation
OM	Operation and maintenance
PLEX	Plant life extension
PORV	Power-operated relief valve
PRA	Probabilistic Risk Assessment
PRT	Pressurizer relief tank
psig	Gage pressure in pounds per square inch
PVRC	Pressure Vessel Research Council
PWR	Pressurized water reactor
PZR	Pressurizer
QA	Quality assurance
RC	Reactor coolant
RCL	Reactor coolant loop
RCP	Reactor coolant pump
RCPB	Reactor coolant pressure boundary
RCS	Reactor coolant system
RES	NRC office of Nuclear Regulatory Research
RHR	Residual heat removal system
RPV	Reactor pressure vessel
RT	Radiographic testing
RTD	Resistance temperature detector
RVHV	Reactor vessel head vent
RVHVS	Reactor vessel head vent system
RWST	Refueling water storage tank
SAR	Safety Analysis Report
SAW	Submerged arc welding
SC	Structure or component
SCC	Stress corrosion cracking
SG	Steam generator
SIS	Safety injection system
SMAW	Shielded manual arc welding
Smls	Seamless
SS	Stainless steel

ACRONYMS (Continued)

SSC	Systems, structures, and components
TASCS	Thermal stratification, cycling, and striping
TG	Transgranular
TGSCC	Transgranular stress corrosion cracking
TLAA	Time-limited aging analysis
T_m	Melting point of a metal
USAS	United States of America Standard
UT	Ultrasonic testing
VCT	Volume control tank
V/D	Vents and drains
VT	Visual testing
WCAP	Westinghouse Corporate Atomic Power - technical report
WOG	Westinghouse Owners Group

1.0 INTRODUCTION

The objectives of this evaluation are to:

- Identify and evaluate aging effects that degrade intended functions
- Identify and evaluate time-limited aging analyses
- Provide options, in terms of activities and program attributes, to manage the aging effects identified in this report

These aging management options, when implemented, will ensure that the intended function of Class 1 piping and associated pressure boundary components is maintained during an extended period of operation. The system-level intended function supported by Class 1 piping and associated components will also be maintained.

Class 1 as used in this report means Safety Class 1 per ANSI/ANS 51.1 [Ref. 1] (see Subsection 2.1.1). ASME Class 1 refers to the design analysis rules for Safety Class 1 piping and components. For plants designed before 1973, the Safety Class 1 definitions are per ANS N46.2 and the safety analysis report requirements for the plant and are essentially the same as current definitions. The B31.1 code has no equivalent to ASME, Section III, B31.7, or Pump and Valve Code Class 1, 2, or 3 classifications of analysis.

When discussing items in this evaluation, the current usual definition applies.

The general definition of Class 1 piping is "piping which contains primary reactor coolant," i.e., the water that flows through the nuclear core during normal power operations.

This evaluation starts by identifying why the system, structure, or component (SSC) is within the scope of the license renewal rule. An SSC is within the scope of the rule if it performs or supports an intended function. SSCs within the scope of the license renewal rule are:

1. The safety-related systems, structures, and components which are relied upon to remain functional during and following design-basis events (10 CFR 50.49 (b)(1)) to ensure the following functions:

   a. The integrity of the reactor coolant pressure boundary
   b. The capability to shut down the reactor and maintain it in a safe shutdown condition, or
   c. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR Part 100 guidelines

2. All nonsafety-related systems, structures, and components whose failure could prevent any of the functions identified in paragraph 1a, b, or c above.

3. All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulation for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.6), and station blackout (10 CFR 50.63).

An intended function is the basis for including an SSC within the scope of license renewal, as defined above.

The evaluation continues by determining if the structure or component (SC) is subject to an aging management review. An SC is subject to an aging management review if the SC:

- Performs an intended function
- Performs an intended function in a passive manner
- Is long-lived

Class 1 piping and associated components within the scope of the rule and subject to an aging management review are identified in Section 2.0. Section 2.0 also identifies mechanisms that cause aging effects and applicable TLAAs. The aging management review (Section 3.0) describes age-related degradation mechanisms to identify resulting aging effects. Aging effects are then evaluated to determine degradation of the intended function. Management options for aging effects that degrade the intended function and effects caused by TLAA degradation mechanisms are provided in Section 4.0.

The aging management options provided in this evaluation must be developed into programs by utilities applying for a renewed license. Implementation of these programs during an extended period of operation will ensure that aging effects are managed and that the intended function is maintained.

Reactor coolant system (RCS) level intended functions will be maintained by maintaining Class 1 piping and associated components functions that support RCS intended functions. Hereafter, those Class 1 piping and associated components functions that support RCS intended functions will be referenced as Class 1 piping and associated components intended functions. Aging management options identified in this report, when implemented, will ensure that the Class 1 piping and associated components intended function is maintained during an extended period of operation.

1.1 APPLICABILITY

This evaluation is generically applicable to domestic commercial nuclear power plants with the Westinghouse nuclear steam supply system (NSSS). Preparation of the report included establishment of boundaries by Westinghouse Electric Company as well as utility reviewer confirmation of these boundaries to a practical extent. Use of this report, as referenced by a license renewal application, should include verification of all the bounding information against plant-specific data. This verification will identify what plant-specific data are not covered by this report and shall be evaluated as part of the license renewal application.

1.2 AGING MANAGEMENT EVALUATION SCOPE

The scope of this evaluation is limited to domestic commercial nuclear power plants with Westinghouse NSSS. Table 1-1 provides a listing of plants included in this evaluation. This report addresses the aging effects for Class 1 piping. Reactor coolant loop (RCL) piping consists of the RCS piping to and from the reactor pressure vessel (RPV), steam generator (SG), reactor coolant pumps (RCPs), and pressurizer (PZR). Figure 1-1 presents a schematic arrangement for a typical RCL with the PZR. The RPV, reactor internals, SG, PZR, RPV supports, and non-Class 1 piping are covered in separate generic technical reports [Refs. 2 through 7].

Table 1-1 shows that the initial commercial operation dates for these in-scope plants range from 1968 to 1996. This broad time frame encompasses the early days of the nuclear power plant design to the present.

There are three basic designs: two-loop, three-loop, and four-loop plants. Typical reactor coolant loop configurations are shown in Figure 1-2. The reactor coolant piping for each loop is essentially the same for all plants. Though the balance of the auxiliary lines connect at slightly different locations, their basic function, size, and materials of construction are essentially the same. The environmental conditions of pressure, temperature, and water chemistry also are essentially identical. Therefore, the generic description of aging mechanisms and the effects they cause can be applied to all plants.

TABLE 1-1
GENERAL PLANT DATA

Plant Name	Plant Alpha	Utility	No. of Loops	Size (MW)	Commercial Operation
Beaver Valley 1	DLW	Duquesne Light	3	810	10/01/76
Beaver Valley 2	DMW	Duquesne Light	3	833	11/17/87
Braidwood 1	CCE	Commonwealth Edison	4	1120	07/29/88
Braidwood 2	CDE	Commonwealth Edison	4	1120	10/17/88
Byron 1	CAE	Commonwealth Edison	4	1105	09/16/85
Byron 2	CBE	Commonwealth Edison	4	1105	08/21/87
Callaway	SCP	Union Electric	4	1125	12/19/84
Catawba 1	DCP	Duke Power	4	1129	06/29/85
Catawba 2	DDP	Duke Power	4	1129	08/19/86
Comanche Peak 1	TBX	Texas Utilities	4	1150	08/13/90
Comanche Peak 2	TCX	Texas Utilities	4	1150	08/03/93
Diablo Canyon 1	PGE	Pacific Gas & Electric	4	1073	05/07/85
Diablo Canyon 2	PEG	Pacific Gas & Electric	4	1087	03/13/86
Donald C. Cook 1	AEP	American Electric Power	4	1020	08/27/75
Donald C. Cook 2	AMP	American Electric Power	4	1060	07/01/78
Farley 1	ALA	Alabama Power	3	824	12/01/77
Farley 2	APR	Alabama Power	3	828	07/30/81
Ginna	RGE	Rochester Gas & Electric	2	470	07/01/70
Indian Point 2	IPP	Consolidated Edison of NY	4	970	08/01/74
Indian Point 3	INT	NY Power Authority	4	965	08/30/76
Kewaunee	WPS	Wisconsin Public Service	2	503	06/16/74
McGuire 1	DAP	Duke Power	4	1129	12/01/81
McGuire 2	DBP	Duke Power	4	1129	03/01/84
Millstone 3	NEU	Northeast Utilities	4	1142	04/23/86
North Anna 1	VRA	Virginia Electric Power Co.	3	915	06/06/78
North Anna 2	VGB	Virginia Electric Power Co.	3	915	12/14/80

TABLE 1-1 (Continued)
GENERAL PLANT DATA

Plant Name	Plant Alpha	Utility	No. of Loops	Size (MW)	Commercial Operation
Point Beach 1	WEP	Wisconsin Electric Power	2	485	12/21/70
Point Beach 2	WIS	Wisconsin Electric Power	2	465	10/01/72
Prairie Island 1	NSP	Northern States Power	2	503	12/16/73
Prairie Island 2	NRP	Northern States Power	2	500	12/21/74
Robinson 2	CPL	Carolina Power & Light	3	665	03/07/71
Salem 1	PSE	Public Service Electric & Gas	4	1106	06/30/77
Salem 2	PNJ	Public Service Electric & Gas	4	1106	10/13/81
Seabrook	NAH	Public Service of N. H.	4	1150	08/19/90
Sequoyah 1	TVA	Tennessee Valley Authority	4	1148	07/01/81
Sequoyah 2	TEN	Tennessee Valley Authority	4	1148	06/01/82
Shearon Harris	CQL	Carolina Power & Light	3	860	05/02/87
South Texas Project 1	TGX	Houston Light & Power	4	1250	08/25/88
South Texas Project 2	THX	Houston Light & Power	4	1250	06/19/89
Summer	CGE	South Carolina Electric & Gas	3	885	01/01/84
Surry 1	VPA	Virginia Electric Power Co.	3	781	12/22/72
Surry 2	VIR	Virginia Electric Power Co.	3	781	05/01/73
Trojan (SHUTDOWN)	POR	Portland Gas & Electric	4	1095	05/20/76
Turkey Point 3	FPL	Florida Power & Light	3	666	12/14/72
Turkey Point 4	FLA	Florida Power & Light	3	666	09/07/73
Vogtle 1	GAE	Georgia Power	4	1079	06/01/87
Vogtle 2	GBE	Georgia Power	4	1100	05/20/89
Watts Bar 1	WAT	Tennessee Valley Authority	4	1177	1996
Watts Bar 2	WBT	Tennessee Valley Authority	4	1177	indef.

TABLE 1-1 (Continued)
GENERAL PLANT DATA

Plant Name	Plant Alpha	Utility	No. of Loops	Size (MW)	Commercial Operation
Wolf Creek	SAP	Kansas Gas & Electric	4	1135	09/03/85
Zion 1	CWE	Commonwealth Edison	4	1040	12/31/73
Zion 2	COM	Commonwealth Edison	4	1040	09/17/74

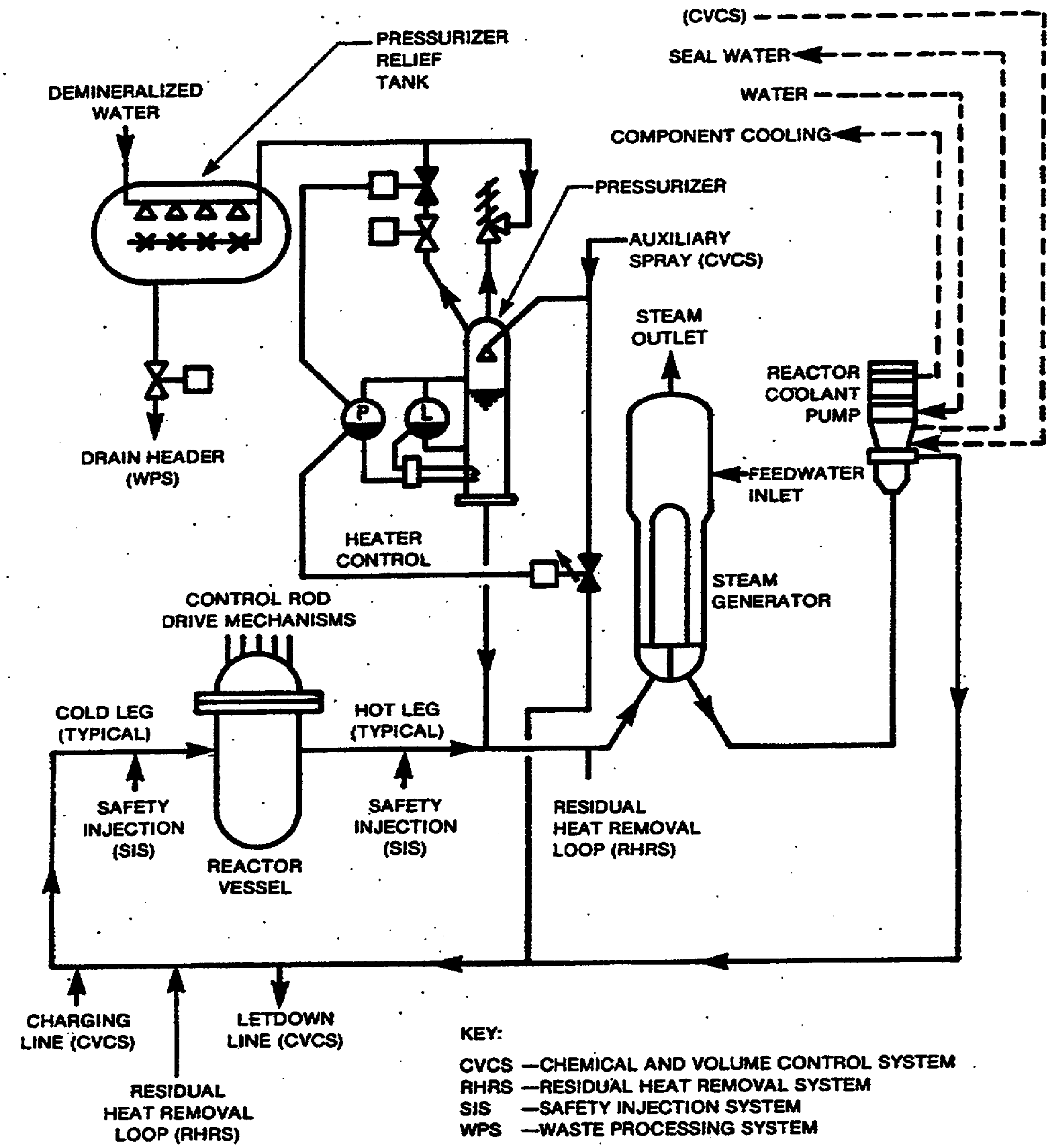


Figure 1-1 Reactor Coolant System Flow Diagram

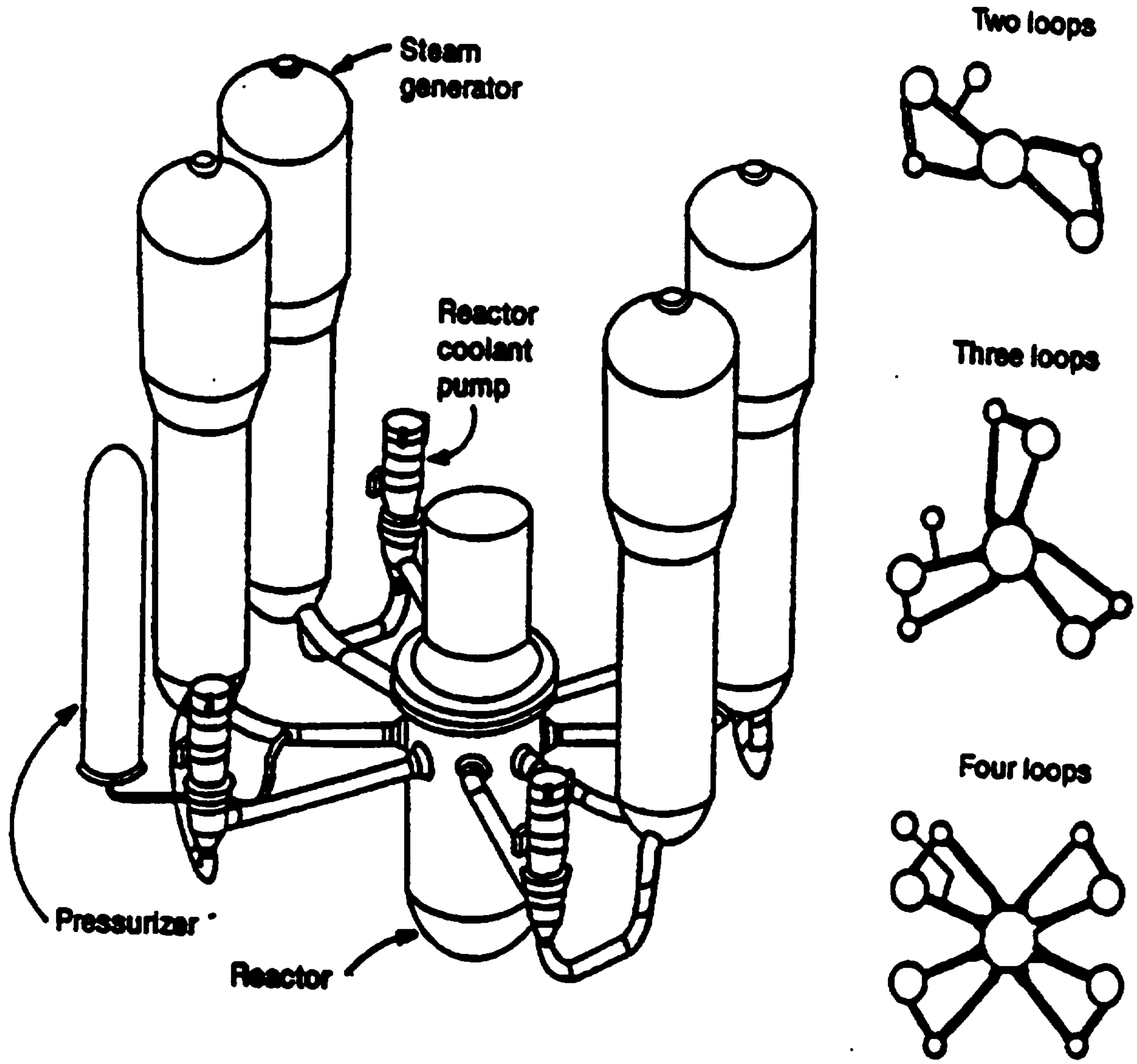


Figure 1-2 Typical Reactor Coolant Loop Configuration

2.0 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES AND AGING EFFECTS

This section identifies time-limited aging analyses (TLAAs) and aging effects related to Class 1 piping and associated pressure boundary components. First, Class 1 piping and associated pressure boundary components are described in general terms. This description includes the boundary of Class 1 piping and associated components evaluated in this report. Next, the reason why Class 1 piping and associated components are within the scope of the license renewal rule is provided. This reason identifies the intended function of Class 1 piping and associated components. The parts or subcomponents of Class 1 piping and associated components are then identified and described in detail. These detailed descriptions identify related TLAAs and age-related degradation mechanisms. Finally, aging effects resulting from age-related degradation mechanisms are identified.

2.1 GENERAL DESCRIPTION AND BOUNDARY

This report considers the scope of the nuclear steam supply systems (NSSSs) to the second isolation valve from the reactor coolant system (RCS). The piping and equipment are described in the following subsections. Class 1 piping includes the circumferential welds at equipment and ends at the second normally closed valve from the RCS or equipment and the 3/8-inch flow restrictor in the auxiliary piping line or nozzle. Note that for early plants, which were not covered by safety classifications, the 3/8-inch diameter flow restrictors may not be applicable. In addition to Class 1 piping, this evaluation also considers the associated pressure boundary components such as Class 1 valve bodies and pump casings.

2.1.1 Class 1 Piping

The three principle components of RCS piping are the 29-inch inside diameter (ID) hot leg connecting the reactor pressure vessel (RPV) outlet and the steam generator (SG) inlet, the 31-inch ID crossover leg pipe connecting the SG outlet to reactor coolant pump (RCP) suction, and the 27.5-inch ID cold leg pipe connecting the RCP outlet and the RPV inlet. In addition to these pipes, a number of smaller pipes are connected to the RCS piping and/or other Class 1 components by nozzles.

The interconnected piping is classified as Class 1 piping up to the system boundaries as defined by ANSI/ANS 51.1 "...the pressure-retaining portion of supports and equipment that form part of the reactor coolant pressure boundary (RCPB) whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal make up capability ...". An excerpt of the table of boundaries from ANSI/ANS 51.1 is included as Figure 2-7.

2.1.2 Associated Pressure Boundary Components

A general definition of the passive associated pressure boundary components is:

> The maintenance rule implementation guidance contains a provision by which licensees may classify certain systems, structures, and components (e.g., raceways, tanks, and structures) as inherently reliable. Inherently reliable systems, structures, and

components by definition generally do not require any continuing maintenance actions and should be considered as "passive."

The Commission considers structures and components for which aging degradation is not readily monitored to be those that perform an intended function without moving parts or without a change in configuration or properties. For example, if a pump or valve has moving parts, an electrical relay can change its configuration, and a battery changes its electrolyte properties when discharging. Therefore, the performance or condition of these components is readily monitored and would not be captured by this description. Further, the Commission proposes that "a change in configuration or properties" should be interpreted to include "a change in state," which is a term sometimes found in the literature relating to "passive." For example, a battery can "change its state" and therefore would not be screened under this description.

Structures or components may have multiple functions, thus some structures or components may meet the "passive" description. For example, although a pump or a valve has some moving parts, a pump casing or valve body performs a pressure-retaining function without moving parts. A pump casing or a valve body meets this description and therefore would be considered for an aging management review. However, the moving parts of the pump, such as the pump impeller, would not be subject to aging management review.

As examples of the implementation of this screening requirement, the Commission would consider structures and components meeting the passive description as including, but not limited to, the reactor vessel, the reactor coolant pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, piping supports, the spent fuel rack, pressure-retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical penetrations, mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets.

Additionally, the Commission would consider structures and components not meeting the "passive" description as including, but not limited to, the portions of pumps that do not form pressure-retaining boundaries, motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicator, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies. [Ref. 8]

The associated pressure boundary components within the scope of this report are discussed in Subsection 2.3.2.

Associated components for Class 1 valves are: the valve body, bonnet, and closure bolting.

Associated components for Class 1 RCPs are: the pump casing; thermal barrier flange; and, depending on the pump model, the main closure flange, bolting ring, diffuser flange, and the associated bolts, nuts, and studs.

Associated components for Class 1 piping are: branch connections or nozzles connected to the pipe, thermowells, scoops, sleeves, branch line restrictors, and the welds.

2.2 CLASS 1 PIPING AND ASSOCIATED COMPONENTS SUBJECT TO AN AGING MANAGEMENT REVIEW

The intended function of Class 1 piping and associated components is to maintain the integrity of the reactor coolant pressure boundary.

Class 1 piping and associated components are subject to an aging management review because they perform an intended function in a passive manner and are long-lived. Class 1 piping and associated components are considered passive components in that they perform their intended function without moving parts and without a change in configuration or properties. RCS Class 1 piping and associated components are also long-lived because they are intended to be nonreplaceable.

2.3 DESCRIPTIONS

This section is divided according to Class 1 piping (Subsection 2.3.1) and the associated pressure boundary components (Subsection 2.3.2). Each subsection describes the relevant components, their functions, and their interactions and interdependence.

2.3.1 Class 1 Piping

Class 1 piping includes large and small bore seamless stainless steel pipe and fittings. For piping larger than 2 inches, butt-welded construction was used. For piping smaller than 2 inches, socket-welded or butt-welded construction was used. Exceptions include thermowells, which may use threaded connections and safety valves, and resistance temperature detector (RTD) bypass lines, which use flanged connections.

RCS piping is comprised of large seamless stainless steel pipe and fittings. The piping is designed and fabricated to withstand system pressures and temperatures under all postulated modes of plant operation and environmental conditions. The piping design specification, in conjunction with the governing code of record, define the design and loading conditions as well as the allowable stresses. Section 2.4 contains additional information regarding typical and specific RCS piping characteristics.

The RCS consists of two, three, or four heat transfer loops connected in parallel to the RPV. Each reactor coolant loop (RCL) contains an RCP and SG. In addition, the RCS includes a pressurizer (PZR), pressure relief tank (PRT), interconnecting piping, and instrumentation necessary for operational control. During operation, the RCPs circulate pressurized fluid through the RPV and RCL. The fluid, which serves as a coolant, moderator, and solvent for boric acid, is heated as it passes through the nuclear core. The fluid in each loop flows from the RPV through the hot leg and into the SG where heat is transferred to the steam supply system for electrical power generation. The fluid flows from the SG to the RCP in the crossover leg and then is pumped back into the RPV in the cold leg. The hot legs, crossover legs, and cold legs of the loop comprise the RCL piping. The RPV, SG, and PZR safe-end nozzle weld to the RCS piping is a similar metal weld and is included in the scope of this evaluation, i.e., the stainless steel (piping) to carbon steel (equipment) bi-metallic weld is part of the equipment design and analysis.

The first principal function of RCS piping is to maintain a continuous, leak-tight pressure boundary for circulation of reactor coolant throughout the primary coolant system. The circulation of primary coolant, in turn, accomplishes subsidiary functions. The first of these is transport of thermal energy from the core to the secondary coolant (as occurs during normal operation), or to a heat sink (as occurs during normal and emergency shutdown). Additional subsidiary functions accomplished by the circulation of primary coolant are: moderating fission neutrons to produce a thermal neutron spectrum in the reactor core, and serving as a solvent for boric acid. Regulation of boric acid concentration provides chemical shim control of reactivity to compensate for the effects of xenon transients, cold shutdowns and startups, and fuel burnup.

The second principal function of RCS piping is to serve as the second barrier to contain fission products produced by the fission process. Two other fission product barriers also exist. The first barrier against fission product release is the fuel element cladding; the third, outermost barrier is the reactor containment boundary (reactor building, penetrations, etc.)

The RCS instrumentation provides the required signals for reliable and efficient operation and control of the system. The Class 1 pressure boundary for the instrumentation interface is included in the scope of Class 1 piping. However, this instrumentation is not included in an aging management evaluation because its function is preserved under current maintenance programs.

RCS piping and associated Class 1 components such as thermal sleeves and nozzles perform their functions in a passive manner. No mechanical motion is required for RCS piping to serve its functions as a pressure boundary, flow path, and fission product barrier.

Certain RCS piping components, however, do require active mechanical motion to perform their required functions. These include the PZR safety valves and power-operated relief valves (PORVs), all motor-operated valves (MOVs), check valves, and reactor vessel head vent valves.

The RCS pressure is controlled with the aid of a PZR in which liquid and vapor are maintained in equilibrium by electrical heaters and fluid spray. Class 1 piping connecting the PZR to the RCLs consists of a PZR surge line, which joins the hot leg with the PZR, and two PZR spray lines, which run from the cold legs to the PZR. To reduce the pressure in the RCS, fluid passes through the spray lines and condenses the steam in the PZR. To increase the pressure in the RCS, heating elements increase PZR fluid temperature and therefore PZR pressure (see Figure 2-1).

For protection against overpressurization in the RCS, safety valves and PORVs connected to the top of the PZR discharge to the PRT where steam is condensed and cooled by mixing with water. The piping connecting the PZR to the safety and relief valves constitutes the PZR safety and relief valve Class 1 lines (see Figure 2-1).

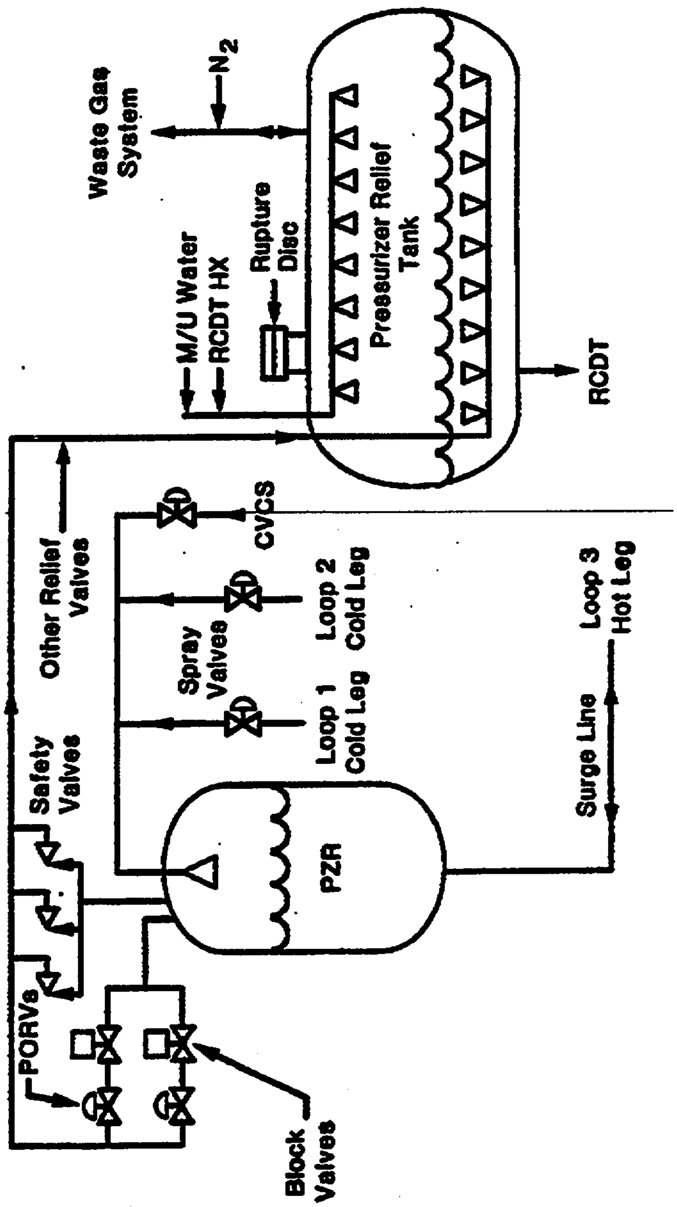


Figure 2-1 Pressurizer Safety and Relief System

The RCS is served by a number of auxiliary systems that are connected to the RCL.

The safety injection system (SIS) provides emergency core cooling (ECC) in the event of a break in either the RCS or NSSS. Borated fluid is injected into the RCS or RPV to counteract any increase in core reactivity resulting from an accident. For two-loop plant designs, the SIS injects into the RPV and RCS piping, and for three- and four-loop plant designs, the SIS injects into the RCS piping. Additional borated fluid is then employed for subsequent injection into the RPV and/or RCS piping to cool the reactor core and prevent the possibility of an uncontrolled return to criticality. In the event of an RCS pipe break resulting from a loss-of-coolant accident (LOCA), the SIS provides enough emergency coolant to the core to replace that lost via the pipe break so that the core does not become excessively overheated (see Figure 2-2).

The primary function of the residual heat removal system (RHR) is to remove heat energy from the reactor core and RCS during plant cooldown and refueling operations (see Figure 2-3).

The chemical and volume control system (CVCS) maintains a programmed fluid level in the PZR, maintains seal water injection flow to the RCP, processes effluent reactor coolant fluid to permit recovery and reuse of the soluble chemical neutron absorber and makeup fluid, and aids in filling, draining, and pressure testing the RCS. In addition, the CVCS controls the fluid chemistry conditions, activity level, and soluble chemical neutron absorber concentration and makeup (see Figure 2-4).

The reactor vessel head vent system (RVHVS) remotely removes noncondensable gas from the reactor vessel and head that may impair ECC and natural circulation cooling and that can lead to false instrumentation (level) if the head is not periodically vented during shutdown. Operation of this system is conducted via remote manual operation from the control room. The RVHVS lines connect directly to the RPV (see Figure 2-5).

The resistance temperature detector (RTD) manifold bypass loop is used to monitor the temperature of the fluid in each of the hot and cold legs of the RCL. For each hot leg manifold, the fluid enters via three scoops in the hot leg, flows through the manifold where the temperature is determined, and returns to the loop at the crossover leg. The fluid for each cold leg manifold comes from one connection in each cold leg. The fluid flows from the manifold to the same return used for the hot leg manifold flow. The RTD bypass system is part of the reactor control system (see Figure 2-6). In some plants, the RTD bypass system is eliminated. For these plants, the bypass piping is removed and fast-response RTD thermowells are installed in the hot leg 1-inch RTD scoops and cold leg 2-inch RTD nozzles. In some cases, additional thermowell penetrations may be installed in the hot and/or cold legs. Table 2-1 identifies the plants that have eliminated the RTD bypass system by using RTD fast-response thermowells. The 3-inch RTD crossover leg return nozzle is capped. Alternatively, in some early plant designs, direct immersion RTDs were installed directly in the RCL piping instead of using the RTD bypass system.

Class 1 piping is important from a plant safety perspective. The integrity of the RCS piping pressure boundary is necessary for maintaining the core cooling function and is a significant contributor in preventing the release of fission products.

TABLE 2-1
REACTOR COOLANT PIPING SPECIAL ITEMS

Plant Name	Loop Stop Valve	Plenum Elbow	Splitter Elbow(1)	Thermal Sleeve(1)	RTD Thermowells(2)
Beaver Valley 1	✔		1	2	✔
Beaver Valley 2	✔	✔		3	✔
Braidwood 1	✔	✔		no	
Braidwood 2	✔	✔		no	
Byron 1	✔	✔		no	
Byron 2	✔	✔		no	
Callaway		✔		no	
Catawba 1		✔		no	✔
Catawba 2		✔		no	✔
Comanche Peak 1		✔		no	
Comanche Peak 2		✔		no	
Diablo Canyon 1			1	1	✔
Diablo Canyon 2			1	2	✔
Donald C. Cook 1			1	2	
Donald C. Cook 2			1	2	
Farley 1			2	3	✔
Farley 2		✔		3	✔
Ginna			1	0	
Indian Point 2			1	0	
Indian Point 3			1	1	✔

TABLE 2-1 (Continued)
REACTOR COOLANT PIPING SPECIAL ITEMS

Plant Name	Loop Stop Valve	Plenum Elbow	Splitter Elbow[(1)]	Thermal Sleeve[(1)]	RTD Thermowells[(2)]
Kewaunee			1	2	
McGuire 1		✔		3	✔
McGuire 2		✔		3	✔
Millstone 3	✔	✔		3	✔
North Anna 1	✔		2	3	
North Anna 2	✔		2	3	
Point Beach 1			1	0	
Point Beach 2			1	0	
Prairie Island 1			1	1	
Prairie Island 2			1	2	
Robinson 2			1	0	✔
Salem 1			1	1	
Salem 2			1	2	
Seabrook		✔		4	✔
Sequoyah 1			1	2	✔
Sequoyah 2			1	2	✔
Shearon Harris		✔		3	
South Texas Project 1		✔		no	✔
South Texas Project 2		✔		no	✔
Summer		✔		3	✔
Surry 1	✔		1	1	
Surry 2	✔		1	1	
Trojan (SHUTDOWN)		✔		3	

TABLE 2-1 (Continued)
REACTOR COOLANT PIPING SPECIAL ITEMS

Plant Name	Loop Stop Valve	Plenum Elbow	Splitter Elbow[1]	Thermal Sleeve[1]	RTD Thermowells[2]
Turkey Point 3			1	0	✔
Turkey Point 4			1	1	✔
Vogtle 1		✔		4	✔
Vogtle 2		✔		4	✔
Watts Bar 1		✔		3	✔
Watts Bar 2		✔		3	
Wolf Creek		✔		no	
Zion 1	✔		1	2	
Zion 2	✔		1	2	

Notes:

1. The design generation number (i.e., 0, 1, 2, 3) is listed for the splitter elbow and thermal sleeves where 0 is the original design.
   "No" means that no thermal sleeves were used.
2. RTD (fast-response) thermowells are used in plants that have eliminated the RTD bypass system.

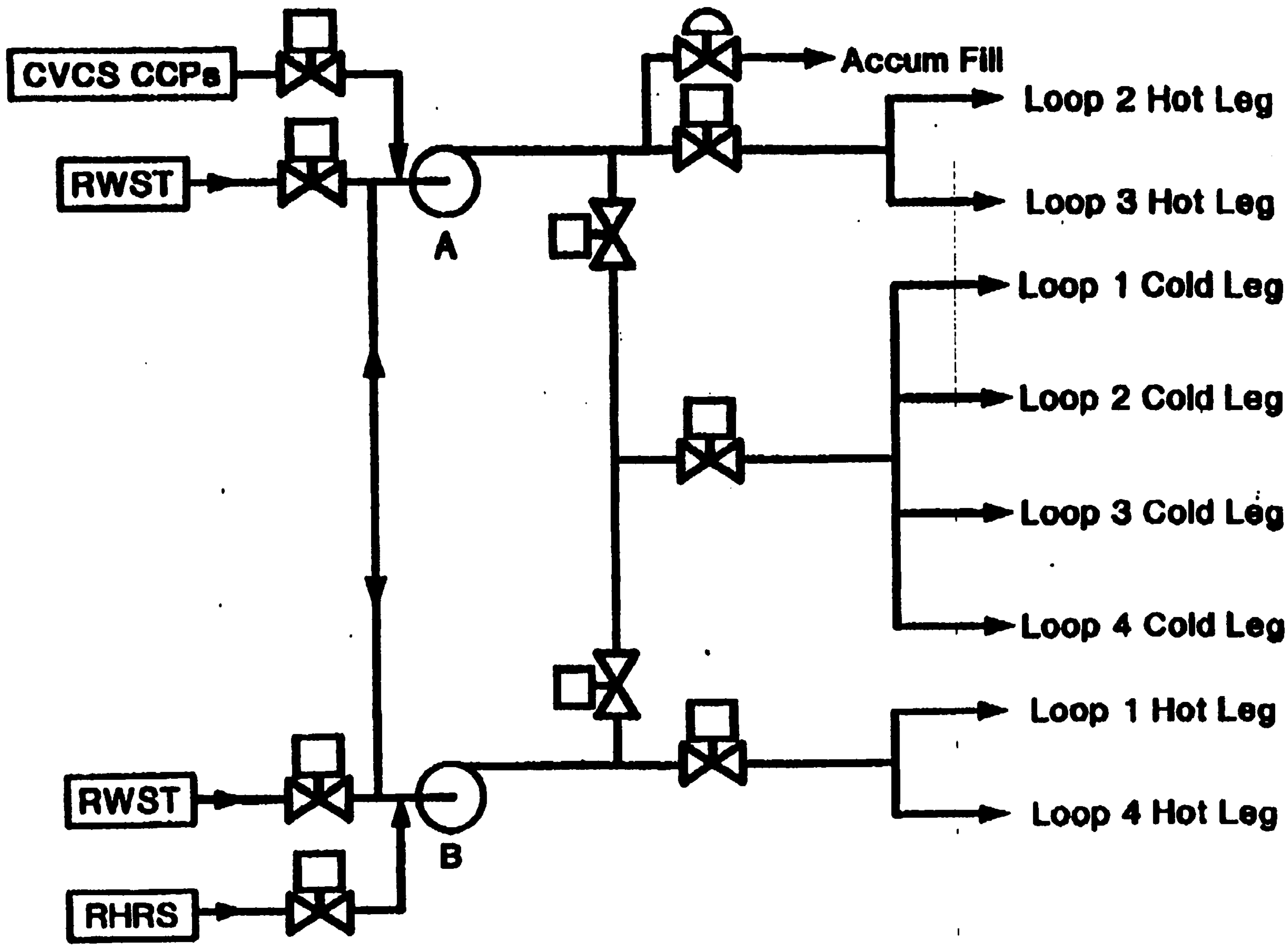


Figure 2-2 Safety Injection System

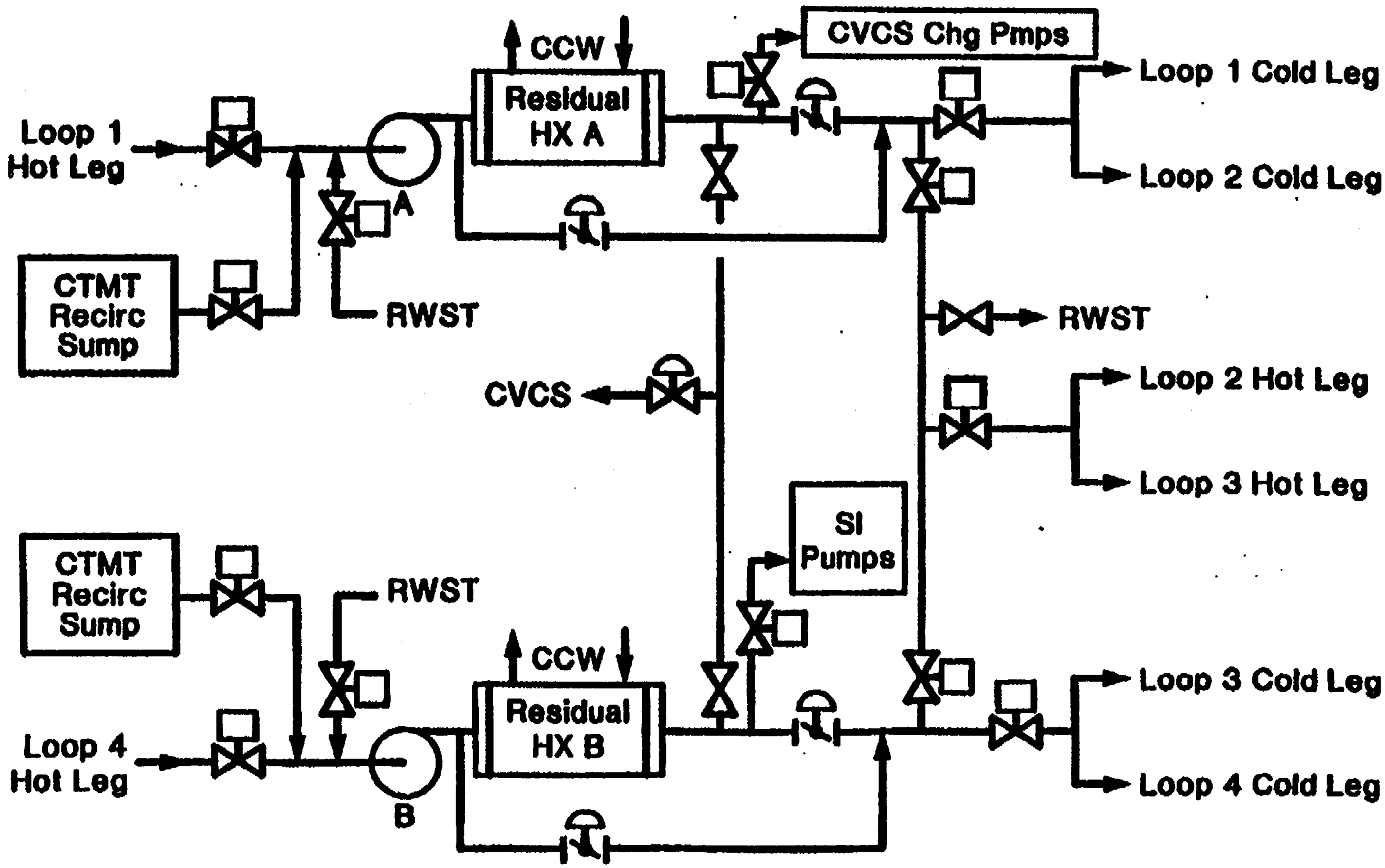


Figure 2-3 Residual Heat Removal System

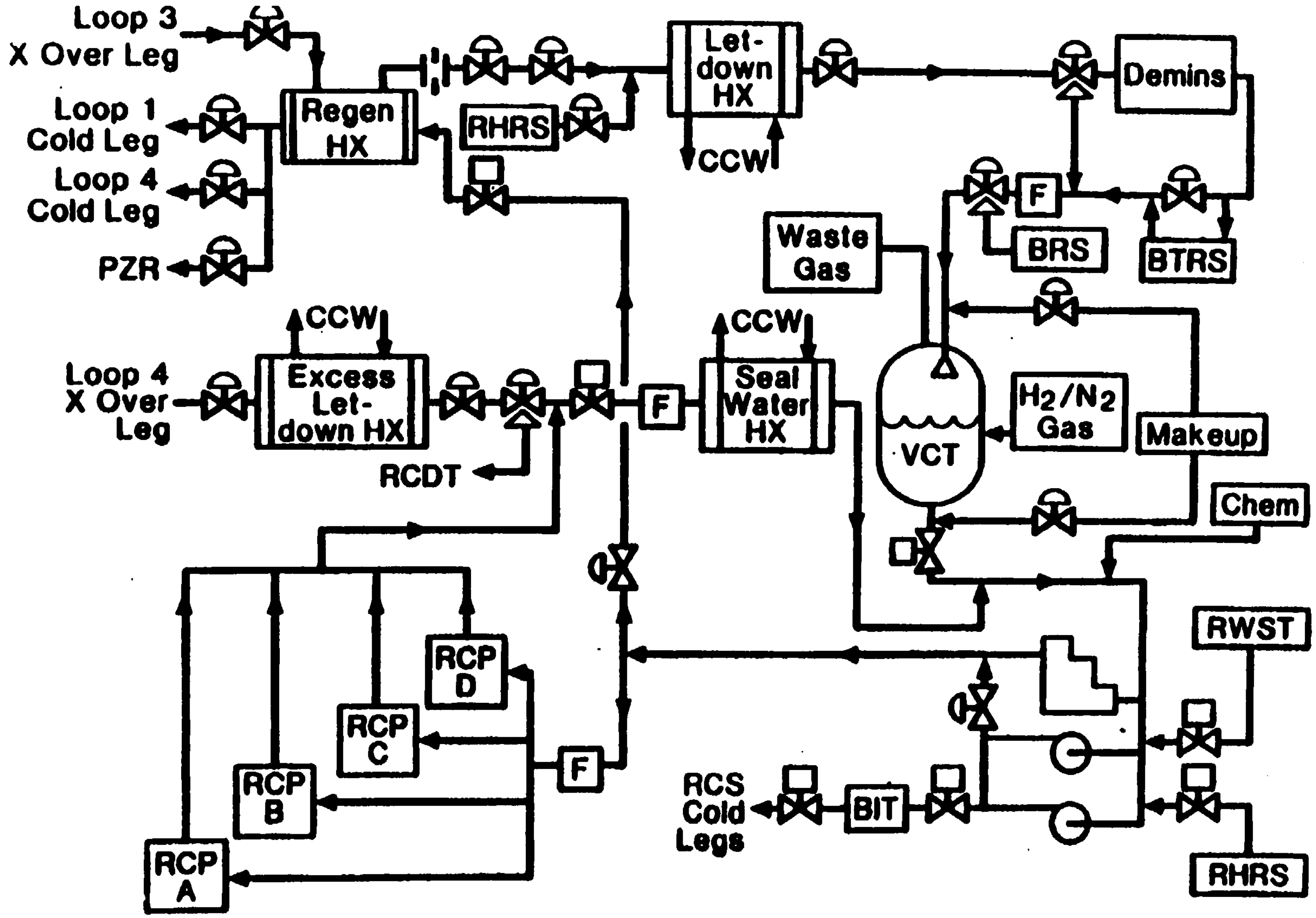


Figure 2-4 Chemical and Volume Control System

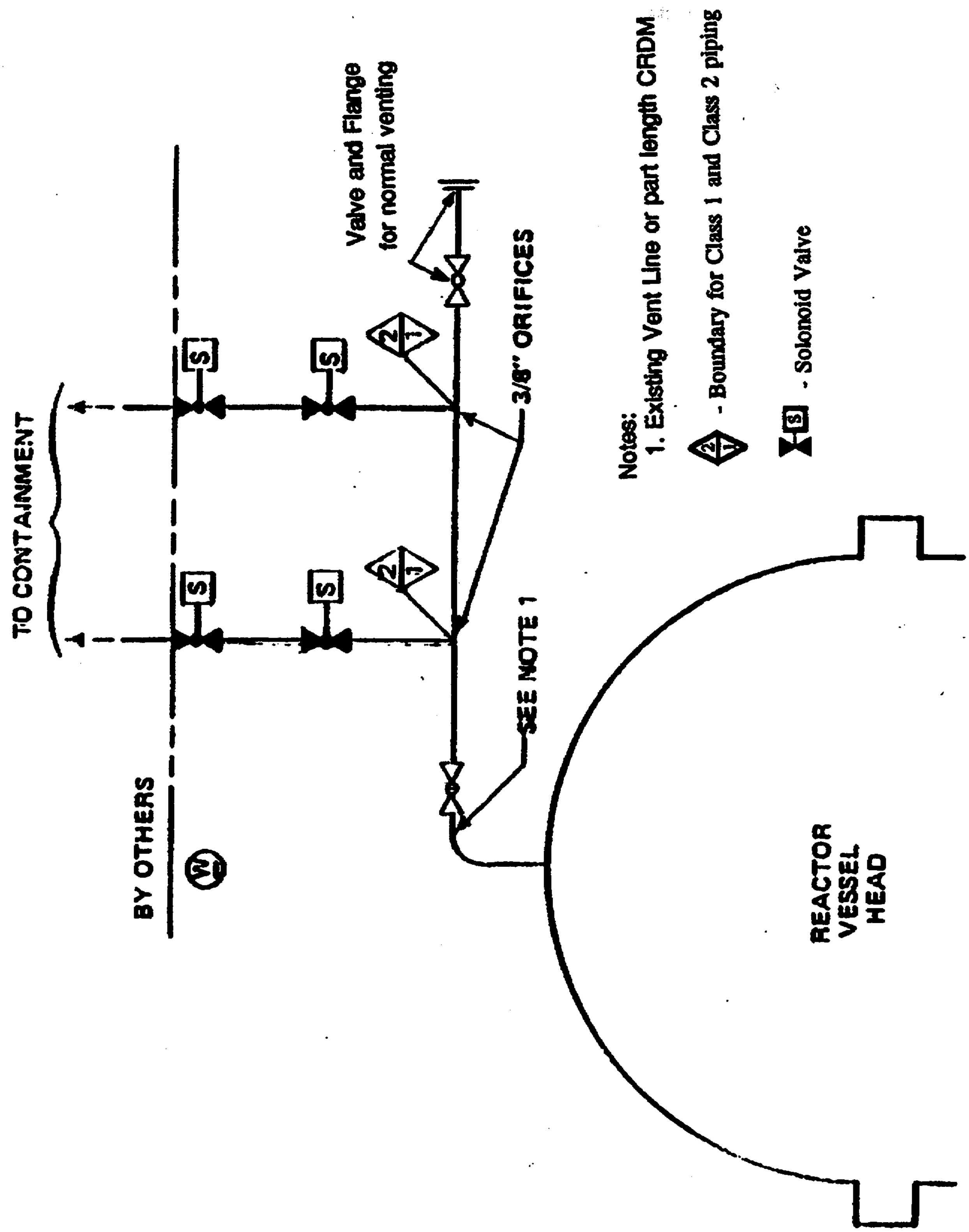


Figure 2-5 Reactor Vessel Head Vent Line

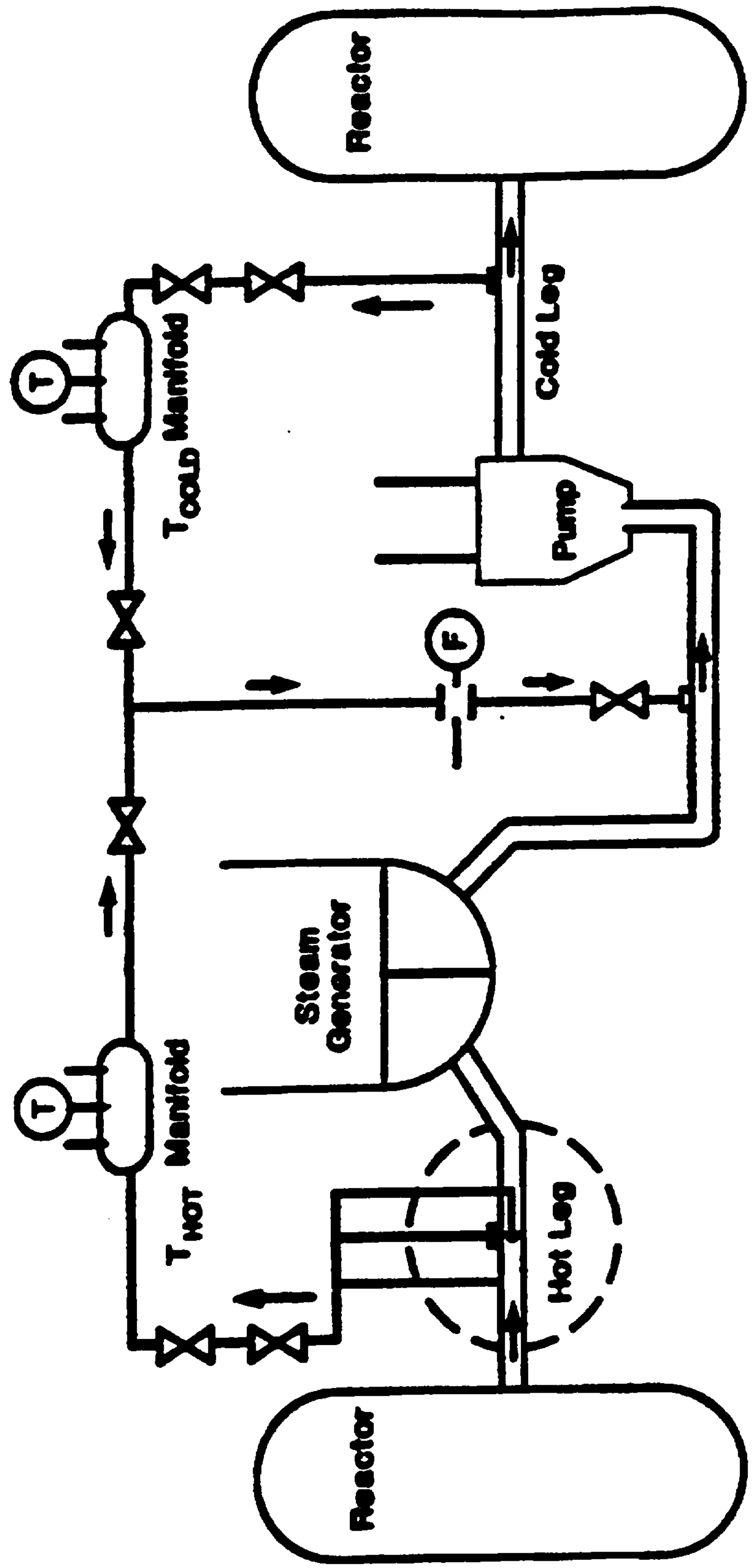


Figure 2-6 Resistance Temperature Detector Bypass System

A breach or break in Class 1 piping is a major consideration for the calculation of core damage and core melt. A number of accident analyses are based on the possibilities of a breach in Class 1 piping. Class 1 piping failures are not a major precursor to events. The consequences of a Class 1 piping failure are large, but the probability of a Class 1 piping failure is small. Thus, due to the low probability of a break occurring in Class 1 piping, the probabilistic risk assessment (PRA) measures of core damage and core melt attributable to a postulated Class 1 piping break are low.

The Class 1 portion of the auxiliary piping systems is defined below. The typical boundary of the interconnecting piping is described by the case numbers from Figure 2-7. Note that each plant may have specific commitments to the regulatory body to increase or decrease the scope.

- PZR surge line from one RCL hot leg to the PZR vessel inlet/outlet nozzle
- PZR spray lines from the reactor coolant cold legs, including the PZR spray scoop, to the spray nozzle on the PZR vessel
- RTD bypass lines including RTD scoops, direct immersion RTDs, and the RTD manifolds
- Loop bypass lines
- PZR safety and relief lines from nozzles on top of the PZR vessel up to and through the power-operated PZR relief valves and PZR safety valves (case 2c from Figure 2-7)
- Class 1 portions of seal injection water and labyrinth differential pressure lines to or from the RCP inside reactor building (case 2c from Figure 2-7)
- Reactor vessel head vent lines (case 2a from Figure 2-7)
- Charging line and alternate charging line from the Class 1 system isolation valves up to the branch connections on the RCL (case 3c from Figure 2-7)
- Letdown line and excess letdown line from the branch connections on the RCL to the Class 1 system isolation valve (case 2f from Figure 2-7)
- RHR lines to or from the RCLs up to the designated Class 1 check valve or isolation valve (suction is case 2f and return is case 3c from Figure 2-7)
- High head and low head safety injection lines from the Class 1 check valve to the RCLs (case 2g from Figure 2-7)
- Accumulator lines from the designated Class 1 check valve to the RCLs (case 2g from Figure 2-7)
- Loop fill, loop drain, sample (including the sample scoop), and instrumentation lines to or from the designated Class 1 isolation valve to or from the RCLs (loop drain is case 2d and the sample, and instrumentation lines are case 2b from Figure 2-7)
- Auxiliary spray line from the Class 1 isolation valve to the PZR spray line header (case 2f from Figure 2-7)
- Sample lines from pressurizer to the Class 1 isolation valve (case 2b from Figure 2-7)
- Boron injection lines from designated Class 1 check valve to the RCL (case 2g from Figure 2-7)

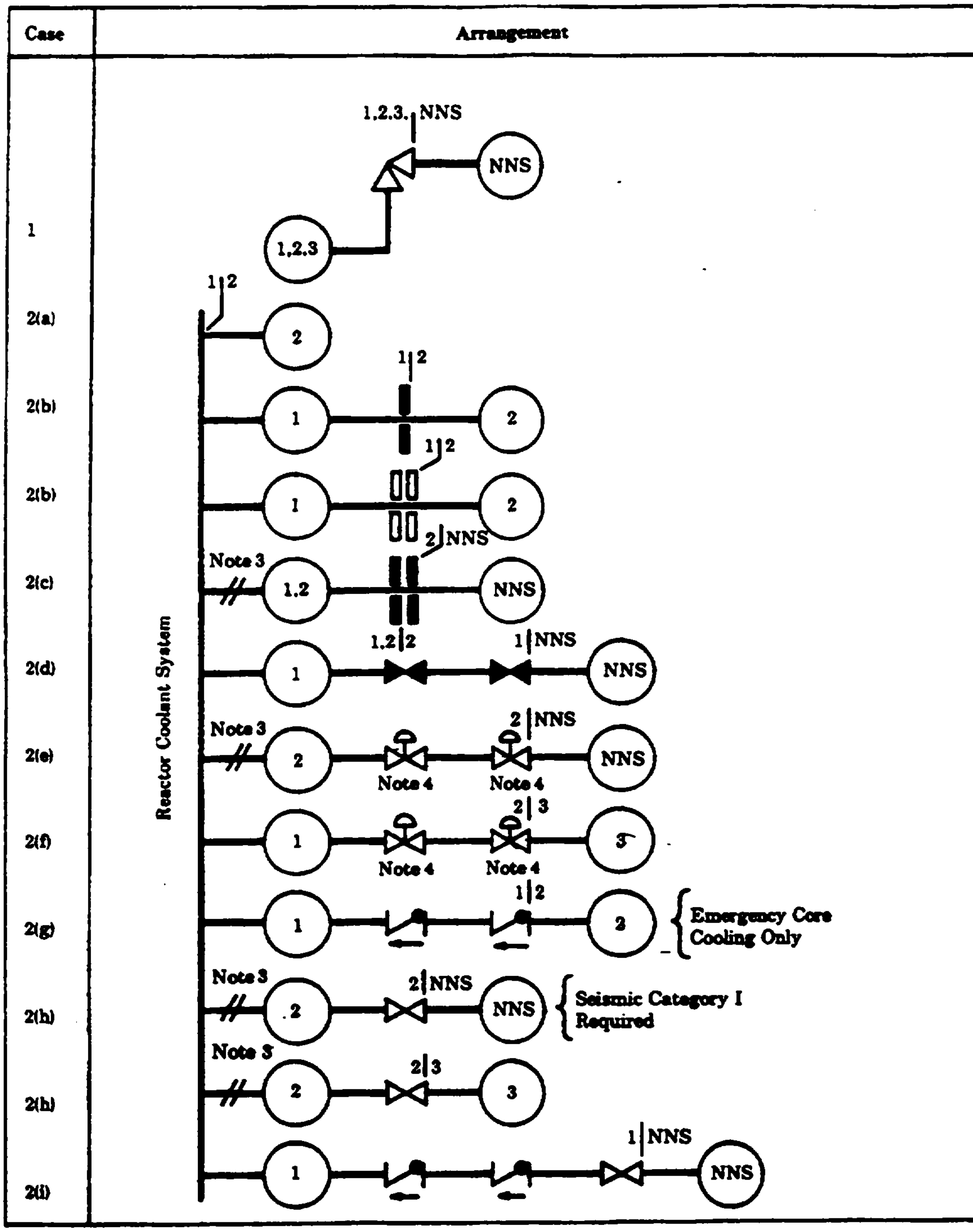


Figure 2-7 Fluid-System Safety Class Interfaces
(See Legend and Notes 1 and 2)

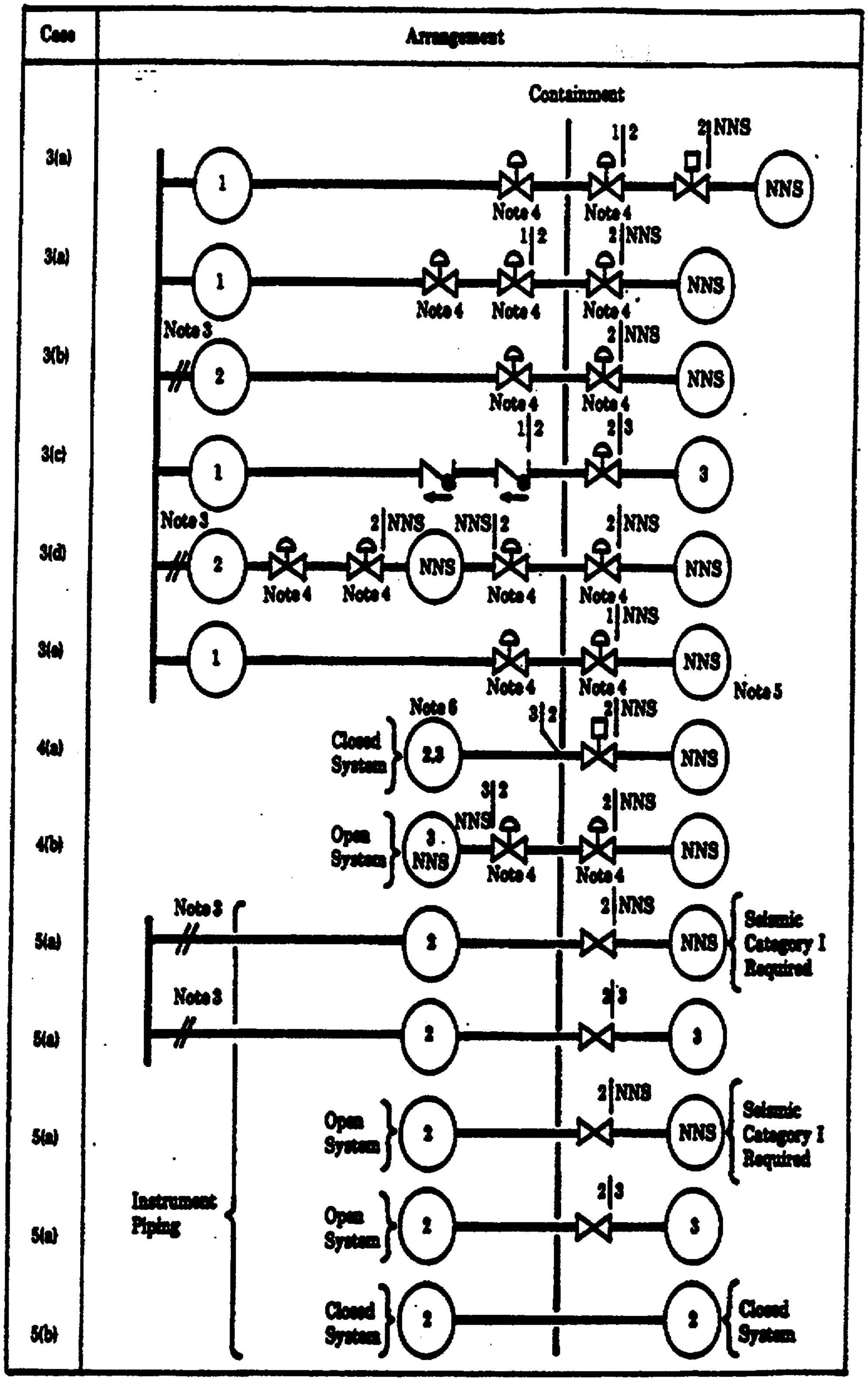
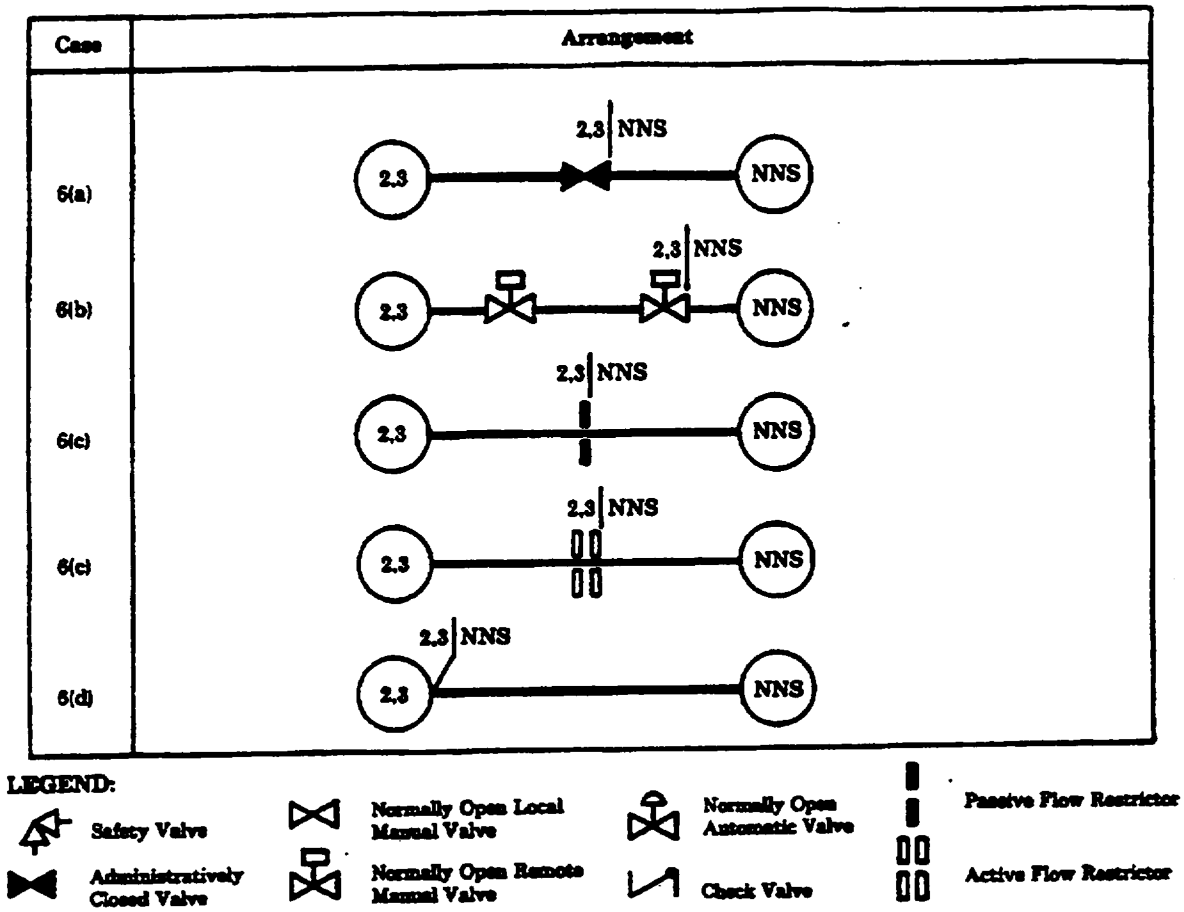


Figure 2-7 Fluid-System Safety Class Interfaces (See Legend and Notes 1 and 2) (Continued)



NOTES:

1. Starting with the lowest ranked valve, the valve hierarchy, i.e., where a given valve may be replaced by any valve higher in the ranking, is as follows: normally open local manual valve, normally open remote manual valve, normally open automatic valve, and administratively closed valve.

2. In each arrangement, the equipment classification at the extreme right of the diagram can be replaced by a more stringent classification up to and including the classification used in the preceding interface; the classification at the extreme right shall not be replaced by a less stringent classification; and the classification at the extreme left shall not be replaced by a more stringent classification and can only be replaced by a less stringent classification where shown.

3. The transition from SC-1 to SC-2 may be accomplished by various options such as illustrated by Cases 2(a), 2(b), 2(d), or 2(f); however, a maximum of three isolation valves shall be sufficient for a transition from SC-1 to NNS.

4. A check valve may be used as one of these automatic valves only in influent lines. If two isolation valves are required to accomplish a Safety Class interface, only one may be a simple check valve.

5. In this arrangement, the NNS piping shall be less than or equal to 6 inches in diameter.

6. In this arrangement, if SC-3 is chosen for the closed system, the closed system shall be low energy.

Figure 2-7 Fluid-System Safety Class Interfaces (Continued)
(See Legend and Notes 1 and 2)

2.3.2 Associated Pressure Boundary Components

2.3.2.1 Nozzles and Special Nozzle Items

In all of the lines listed in the previous section, the nozzle from the Class 1 component is considered part of the Class 1 component. For example, the RVHV nozzle is part of the RPV, and the PZR surge nozzle on the hot leg is part of the hot leg. Typical nozzles and special nozzle item details are shown in Figures 2-8 through 2-14.

These nozzles include:

- Wide-range thermowell (Class 1 with no fluid system safety class interface)
- RTD fast-response thermowell with and without scoop (Class 1 with no fluid system safety class interface)
- Sample scoop and PZR spray scoop
- 3-inch and larger nozzle with thermal sleeve
- 2-inch and smaller nozzle with thermal sleeve
- 3-inch and larger nozzle without thermal sleeve
- 2-inch and smaller nozzle without thermal sleeve
- 45-degree accumulator nozzles

Where installed, the thermal sleeve, thermowell, and scoop are considered in the design analysis of the nozzle.

2.3.2.2 Branch Line Restrictors

The scope of this report only addresses the Class 1 portion of the instrument connections and branch lines. Several instrument connections and some branch lines of the RCS are equipped with 3/8-inch diameter flow restrictors. These restrictors limit the maximum flow through a broken line to a value below the makeup capability of the CVCS. By providing the flow restrictions, the safety classification of the lines is downgraded from Safety Class 1 to Safety Class 2. Note that for early plants that were not covered by safety classifications, the 3/8-inch diameter flow restrictors may not be applicable.

The basis for establishing a maximum break size was derived from the August 1970 draft issue of the ANS document, ANSI/ANS 51.1 [Ref. 1]. ANSI/ANS 51.1 defines safety Class 1 as follows:

> Safety Class 1 applies to reactor coolant system components where failure during normal reactor operations would prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup systems.

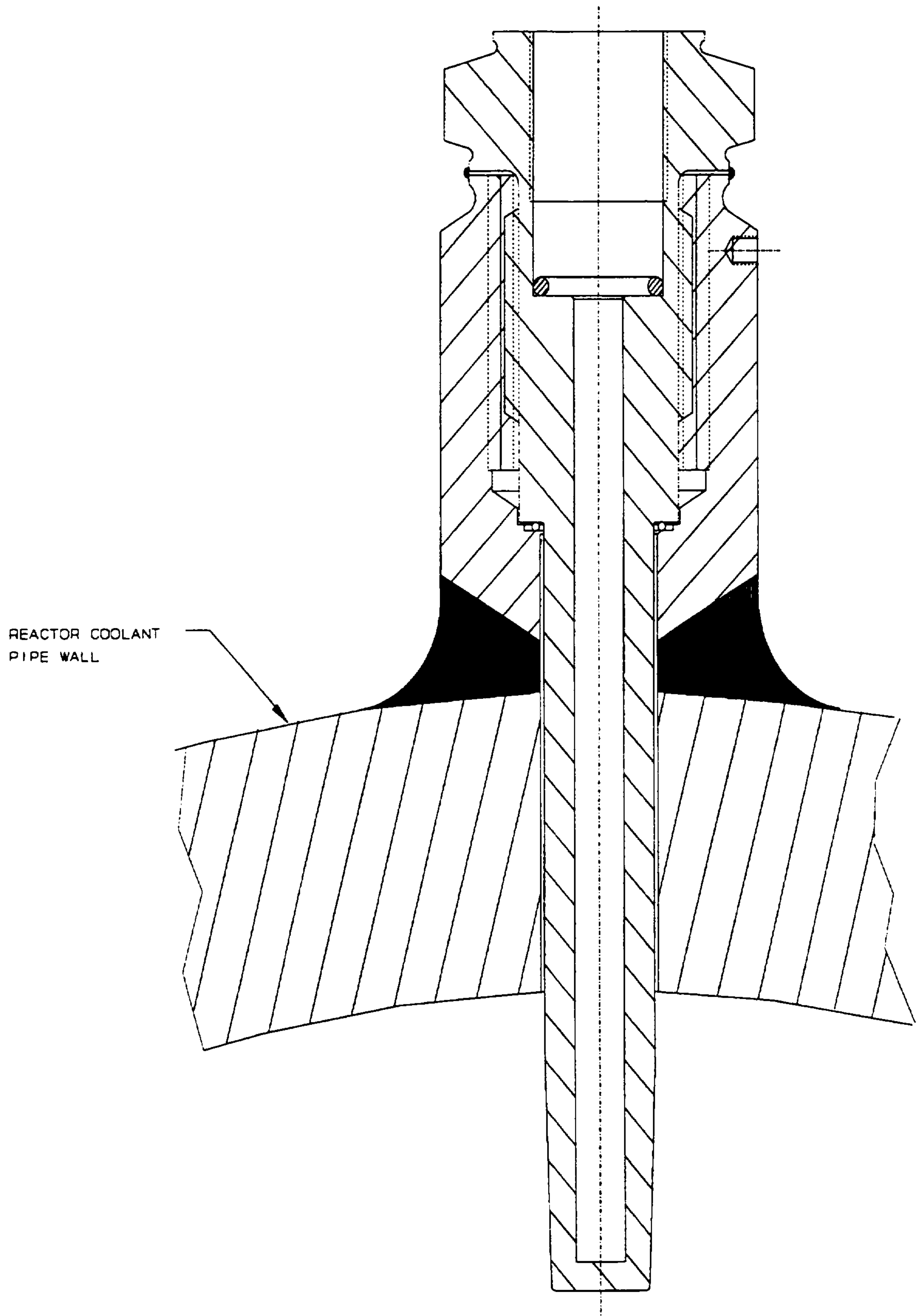


Figure 2-8 Representative Wide-Range Thermowell Installation

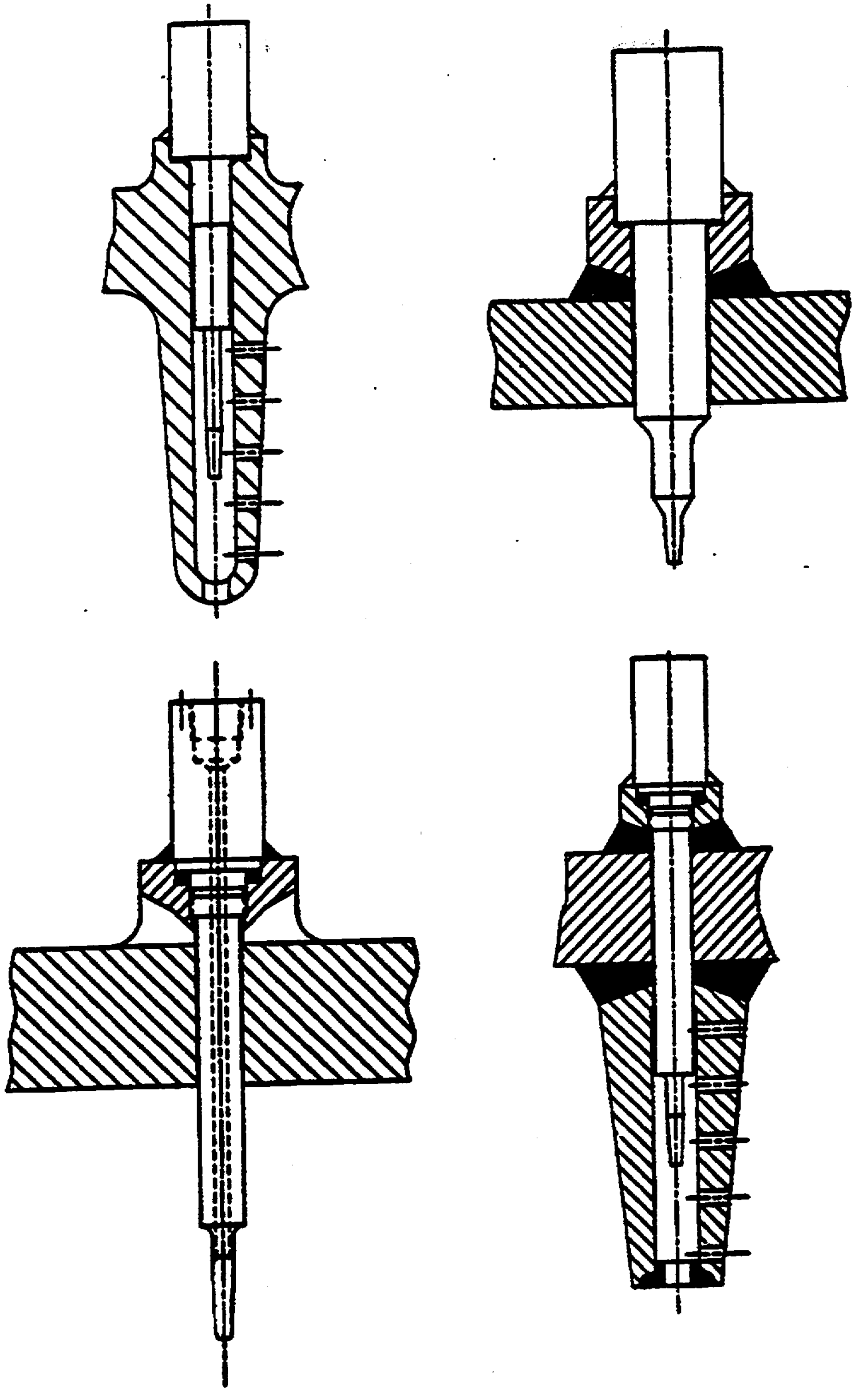


Figure 2-9 Representative Fast-Response Resistance Temperature Detector Thermowell Installations (for resistance temperature detector bypass elimination)

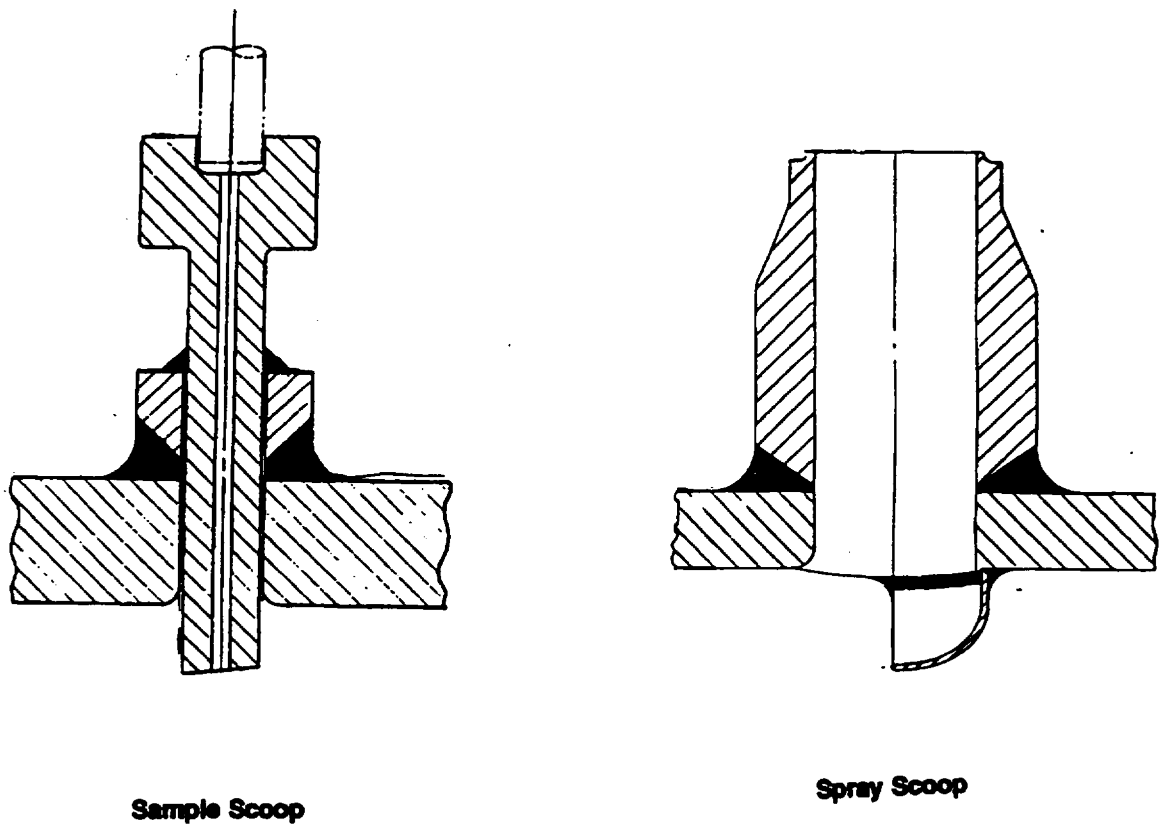


Figure 2-10 Sample and Spray Scoops

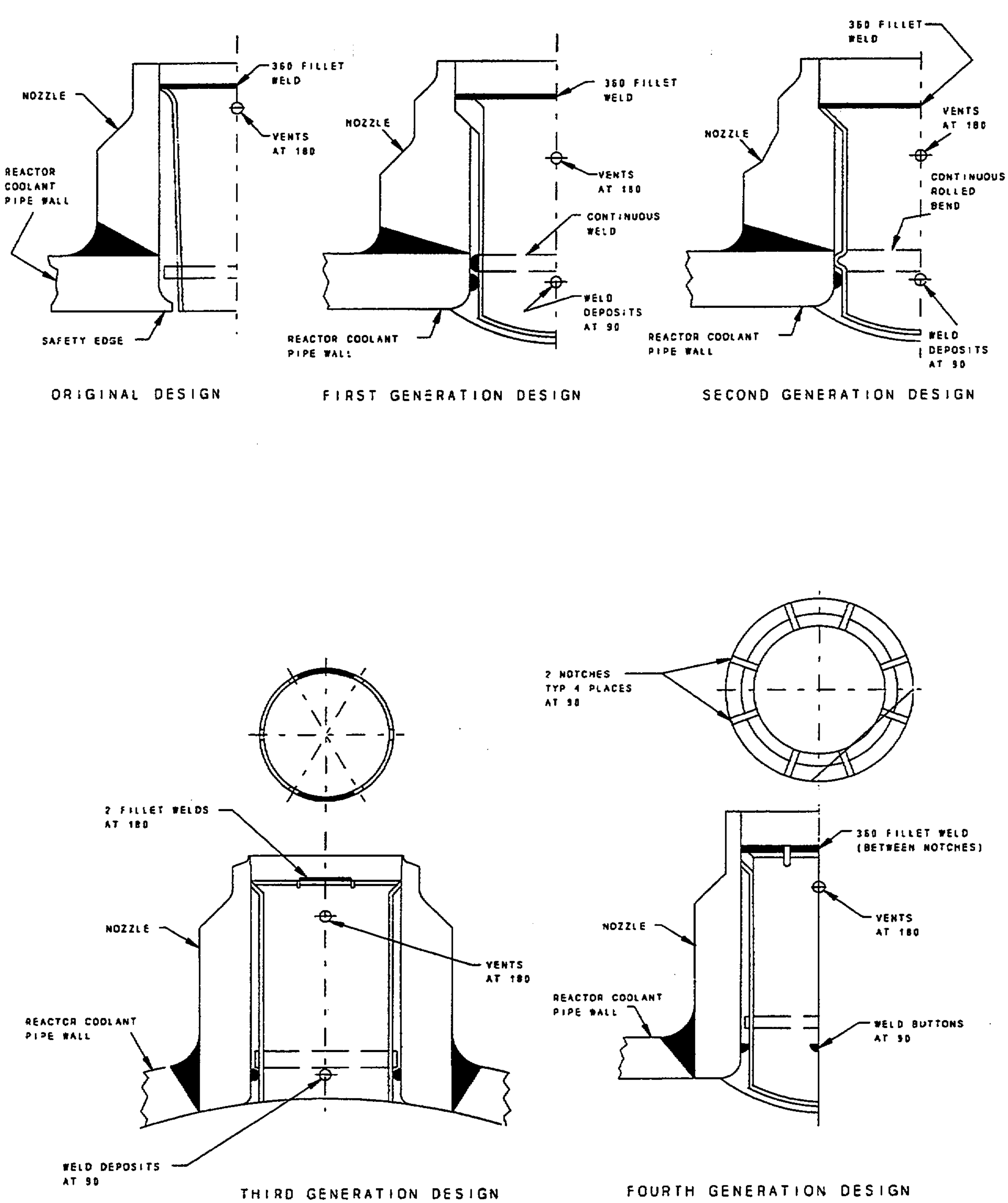


Figure 2-11 Representative Large Bore Nozzles with Thermal Sleeves (See Table 2-1 for identification of plant name to design generation number)

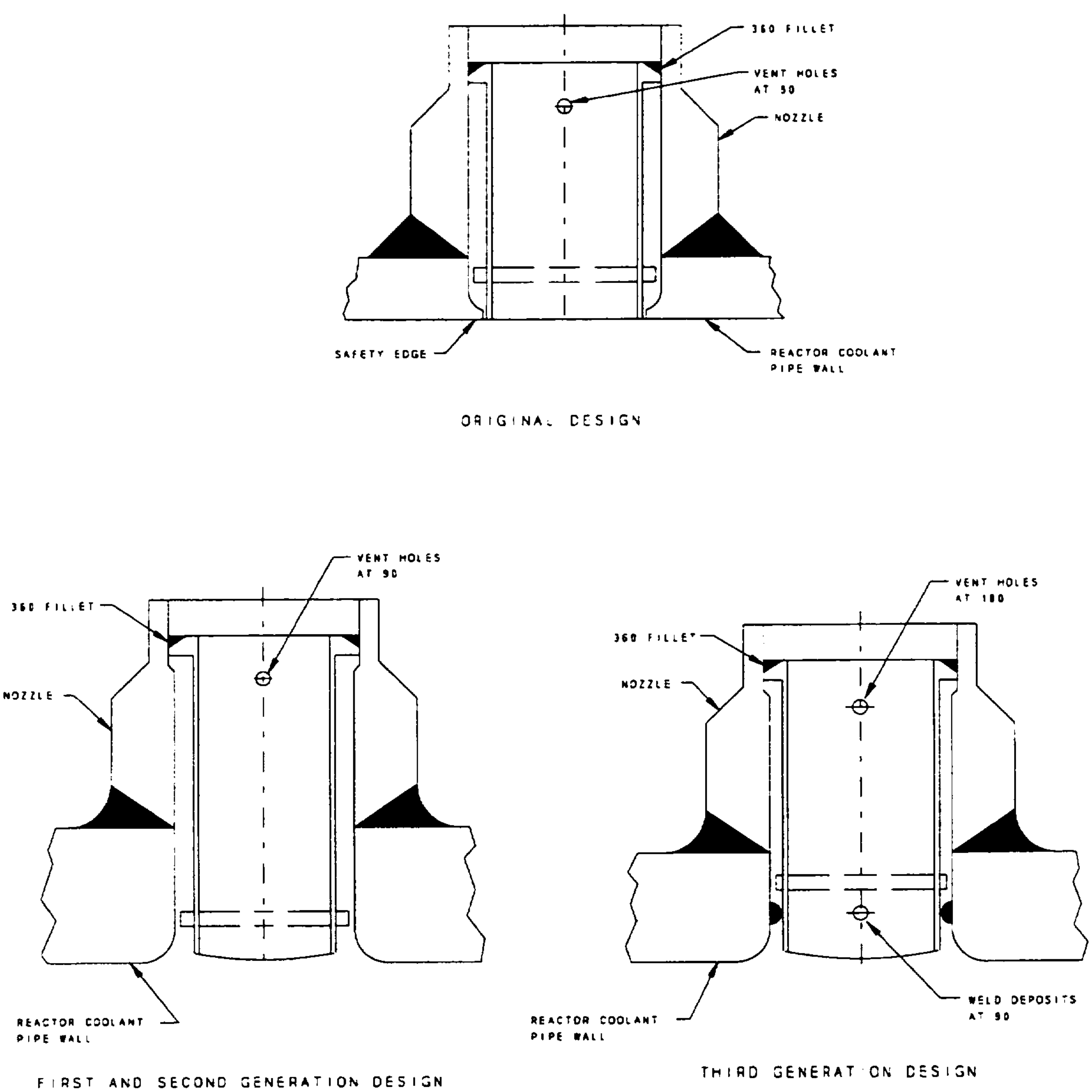


Figure 2-12 Representative Small Bore Nozzles with Thermal Sleeves (See Table 2-1 for identification of plant name to design generation number)

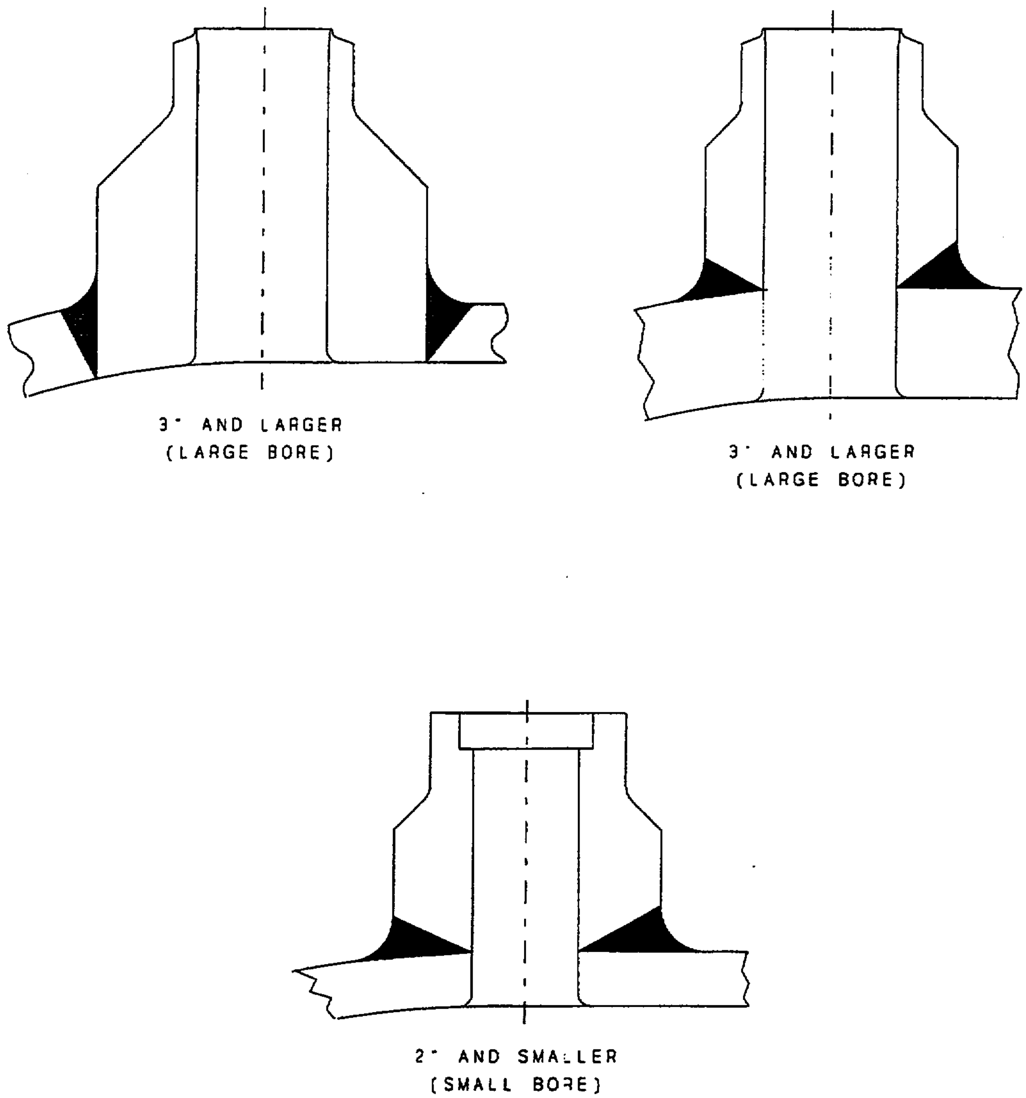


Figure 2-13 Representative Nozzles without Thermal Sleeves

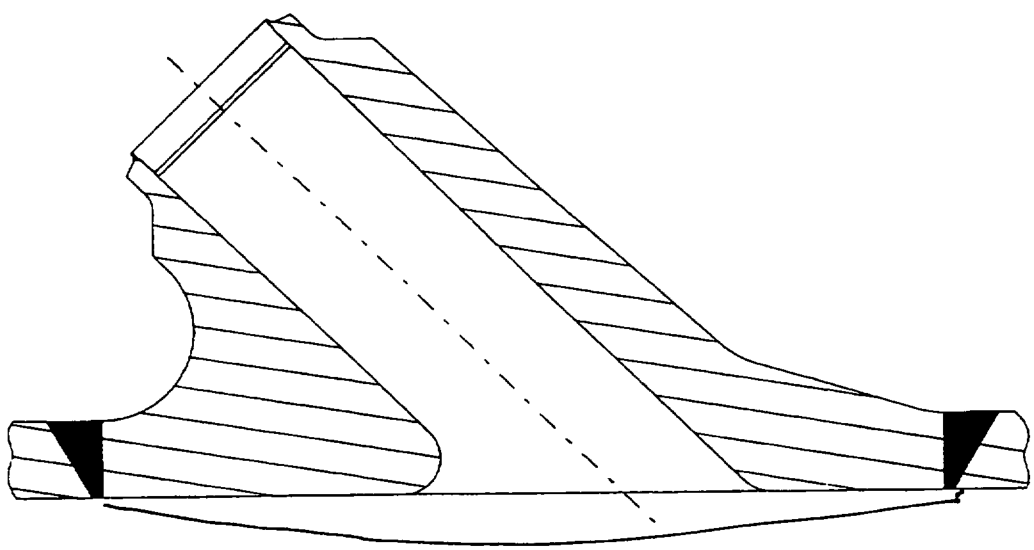


Figure 2-14 Representative 45-degree Accumulator Nozzle

From this statement, it can be concluded that any line connecting to the RCS can be downgraded from Safety Class 1 if the normal makeup system is capable of adding water to the RCS at the same rate it is being lost through a break in that line. The reactor coolant is assumed to be subcooled and at high pressure.

Depending on the charging system arrangement for a particular plant, the normal makeup system may be different. Typically, the normal makeup system is defined as having the capability to supply makeup water through the normal CVCS charging line to maintain plant water inventory so that the PZR water level is unaffected.

Based on a Westinghouse evaluation for several postulated breaks, a 3/8-inch diameter flow restrictor was sized for the subject branch lines and instrument connections.

2.3.2.3 Valves

The aging effect of the pressure boundary valve body is considered in this evaluation.

Representative valves are shown in Figures 2-15 and 2-16, including check valves, manual valves, pneumatic valves (air-operated valve), motor-operated block valves, solenoid-operated valves (typical RVHV), and safety valves.

For the valves described above, this evaluation considers the effects of aging on the pressure boundary functions of the valves and does not address other valve functions. Valves are normally expected to remain in service for the life of the plant, although periodic testing and inspection are required along with occasional maintenance (lapping or dressing of the seats, recalibration). Although the valves are classified as active components, this report will consider the long-lived passive pressure boundary function of these valves. All valve bodies are either statically cast or forged stainless steels, and the connections are welded (except for the PZR safety valve, which is a flanged connection).

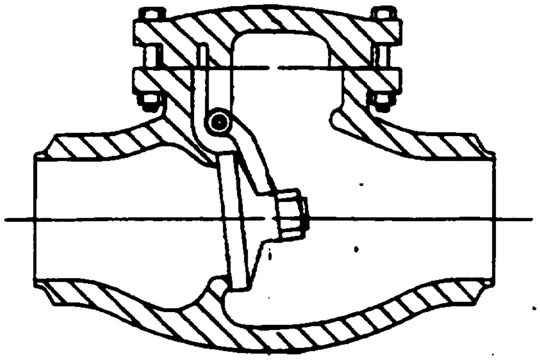
Valve bodies and bonnets that form part of the pressure boundary are classified as long-lived passive components and their pressure-retaining function will be addressed in this evaluation. Valve operators, discs, and seats are classified as active components and thus are not considered in this evaluation. The functions of valve operators, discs and seats are periodically tested to ensure their functions are maintained.

PZR Safety Valves

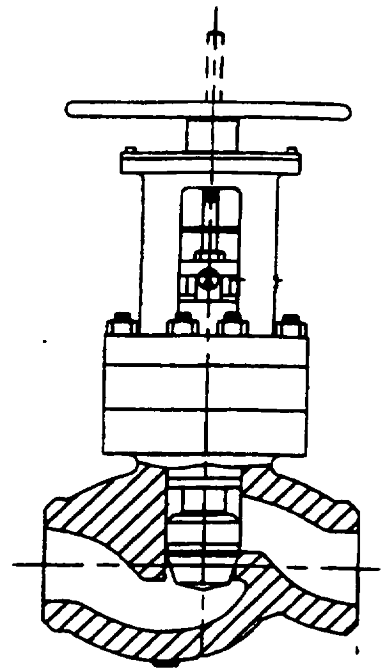
The PZR safety valves are of the totally enclosed pop type and are spring-loaded and self-actuating with backpressure compensation features. These valves provide overpressure protection for the RCS and are sized to limit system pressure to below 110 percent of the system design pressure. In addition, these valves are set to the system design pressure, which is typically 110 percent of the operating pressure. The boundary between the piping and the safety valve is a flanged connection.

Power-Operated Relief Valve (Air-Operated Valve)

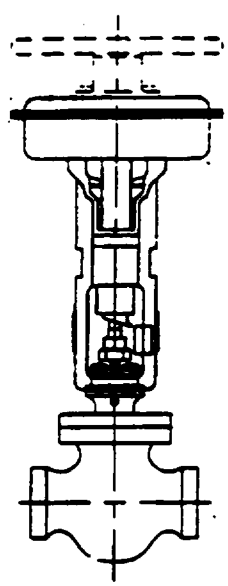
The power-operated (pneumatic) relief valve (PORV) limits system pressure during large system transients. The valves are operated automatically from a pressure sensing system or manually from the control room. The valves are designed to limit PZR pressure to a value



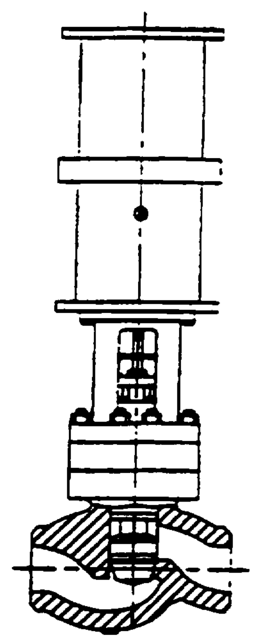
Check Valve



Manual Valve

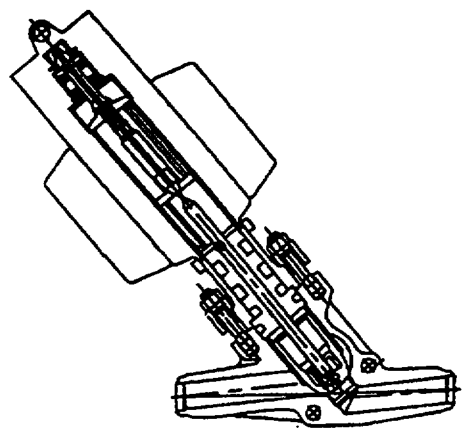


Pneumatic Valve
(Air Operated Valve)

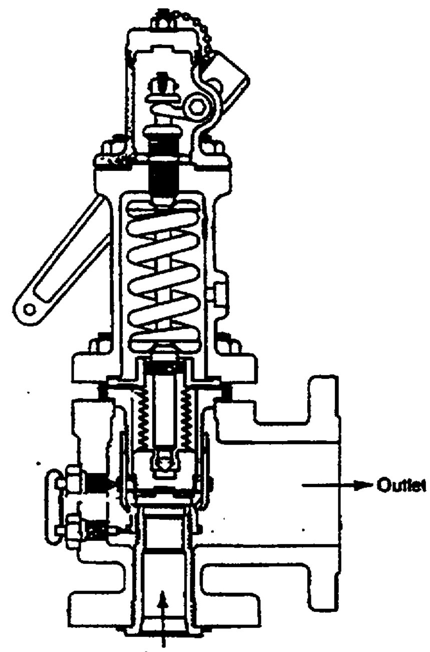


Moter Operated Valve

Figure 2-15 Representative Valves — Check, Manual, Pneumatic, and Motor-Operated



Solenoid Valve



Safety Valve

Figure 2-16 Representative Valves — Solenoid and Safety

below the high pressure trip setpoint for all design transients up to and including the design percentage step load decrease, with steam dump but without reactor trip. The valves are also used with the cold overpressure mitigation system to control pressure during cooldown.

The PORVs have two valves in parallel to ensure that either can perform the relief function.

Head Vent Valves

The solenoid reactor head vent valves are used to remove noncondensable gases or steam from the reactor vessel head to mitigate a possible condition of inadequate core cooling or impaired natural circulation resulting from the accumulation of noncondensable gases.

The RVHV lines have the following redundancies. There are two independent flow paths operated from different safety trains that ensure the venting function. Each independent flow path has two valves, each of which can be the pressure boundary and is thus a redundant assurance that the flow path can be closed.

Motor-Operated Block Valves

Motor-operated block valves are on lines that have intended flow out of the RCS, such as RHR suction, letdown, and PORVs (see air-operated valves). The arrangement consists of two valves in a series that stops flow by closing either valve. These valves provide a pressure boundary to prevent the outflow of fluid from the RCS.

Check Valves B Interconnecting Systems

Interconnecting system check valves are used to allow flow of fluid from systems required to operate in support of plant operations or an emergency situation and to prevent the backflow of reactor coolant into the support system. The check valve serves as a boundary by preventing flow out of the system.

Loop Stop Valves

Some RCL designs include loop isolation stop valves to isolate the RCLs, SG, and RCP from the RPV. During normal operation, these valves are in the open position. Although some plants have these valves, none are currently licensed to operate with the SG and RCP out of service.

2.3.2.4 Thermal Barrier and RCP Seals

The aging effect of the pressure boundary RCP casing is considered in this evaluation. Representative RCP models are shown in Figures 2-17 through 2-20.

In addition to the RCP casing being a part of the Class 1 pressure boundary, the tubes of the thermal barrier heat exchanger within the RCP are considered to be part of the pressure boundary. The aging processes affecting stainless steel tubes are essentially the same as the balance of the Class 1 piping and are discussed in that context of this report.

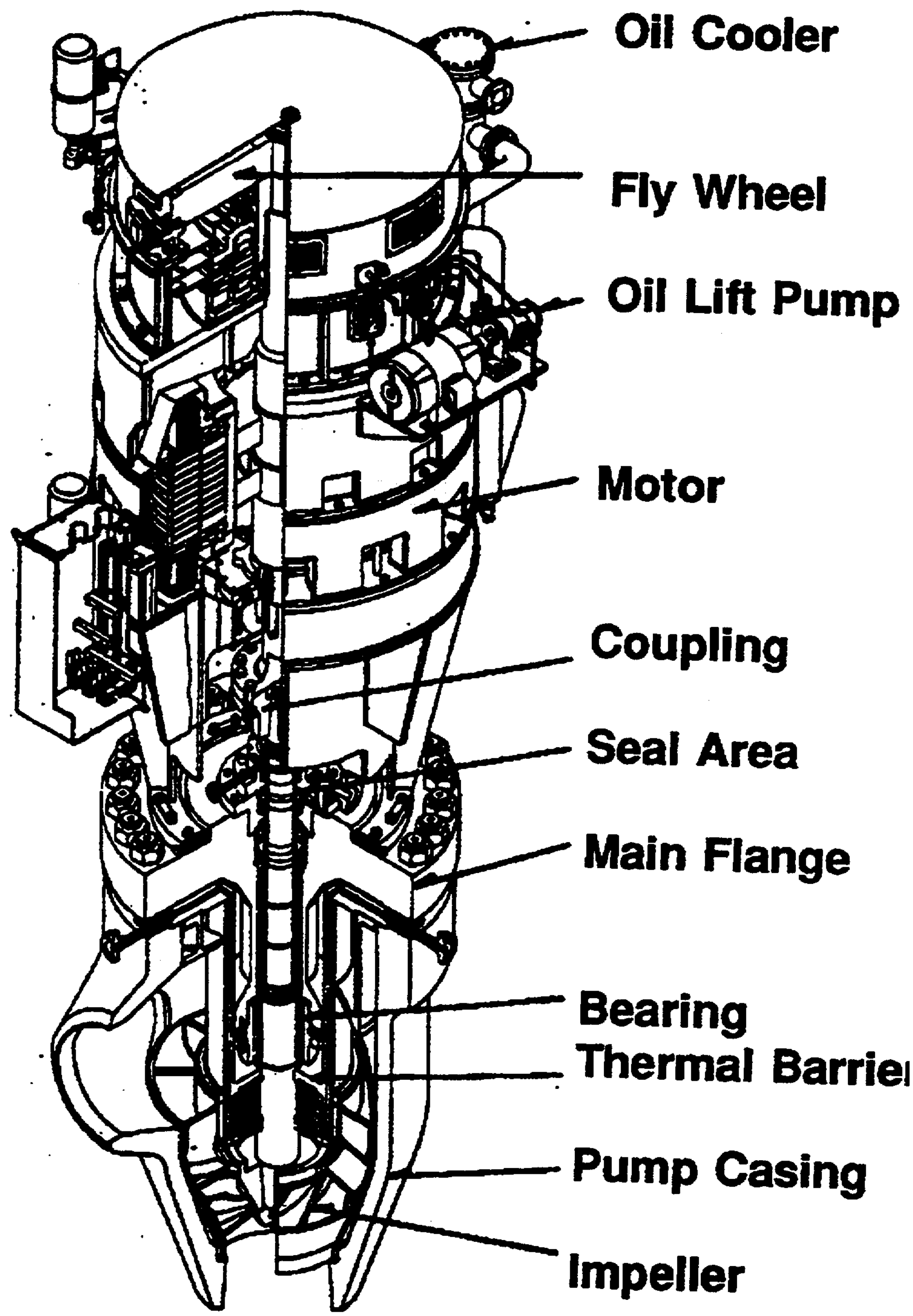


Figure 2-17 Representative Reactor Coolant Pump — Model 93A

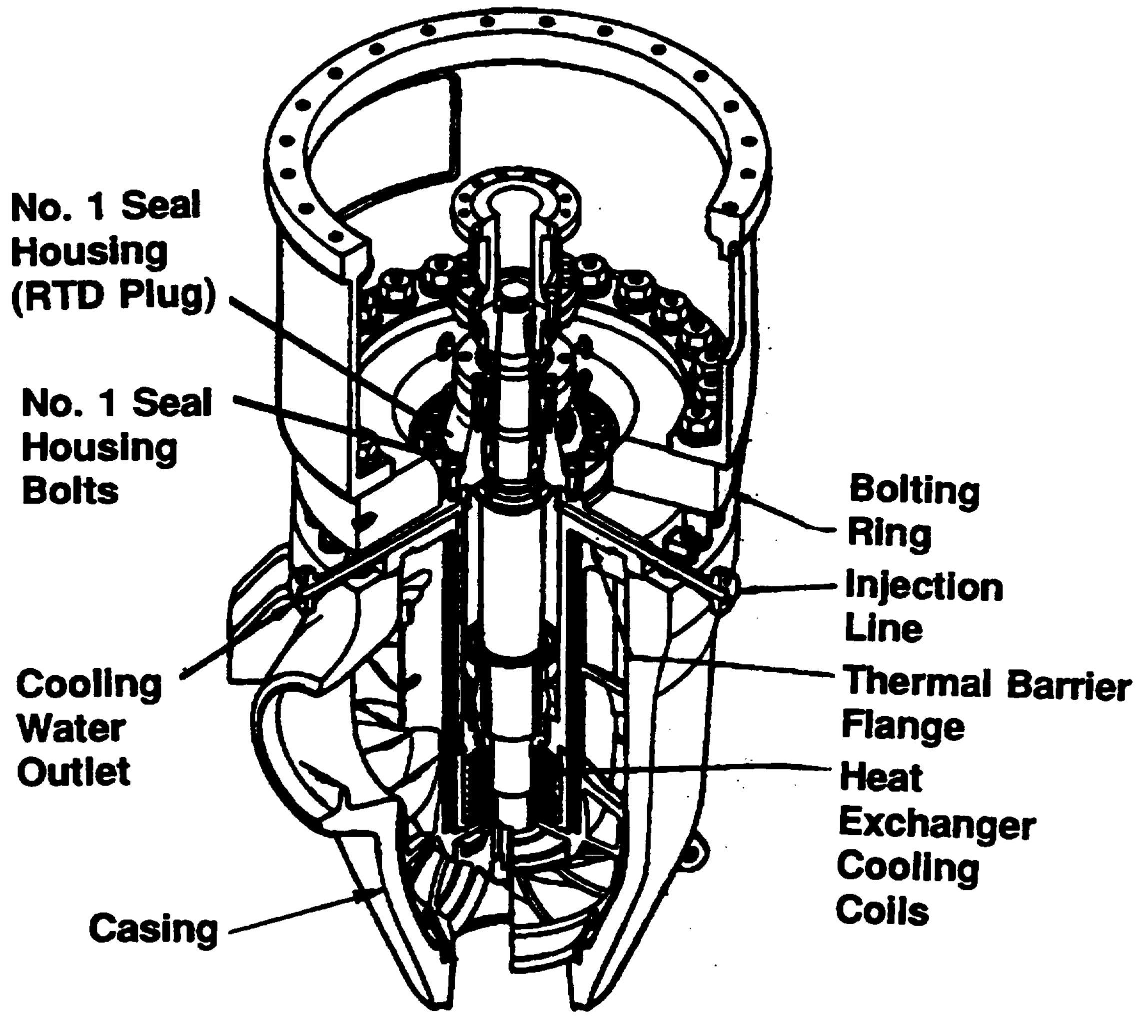


Figure 2-18 Representative Reactor Coolant Pump — Model 93A-1 Pressure Boundary Components

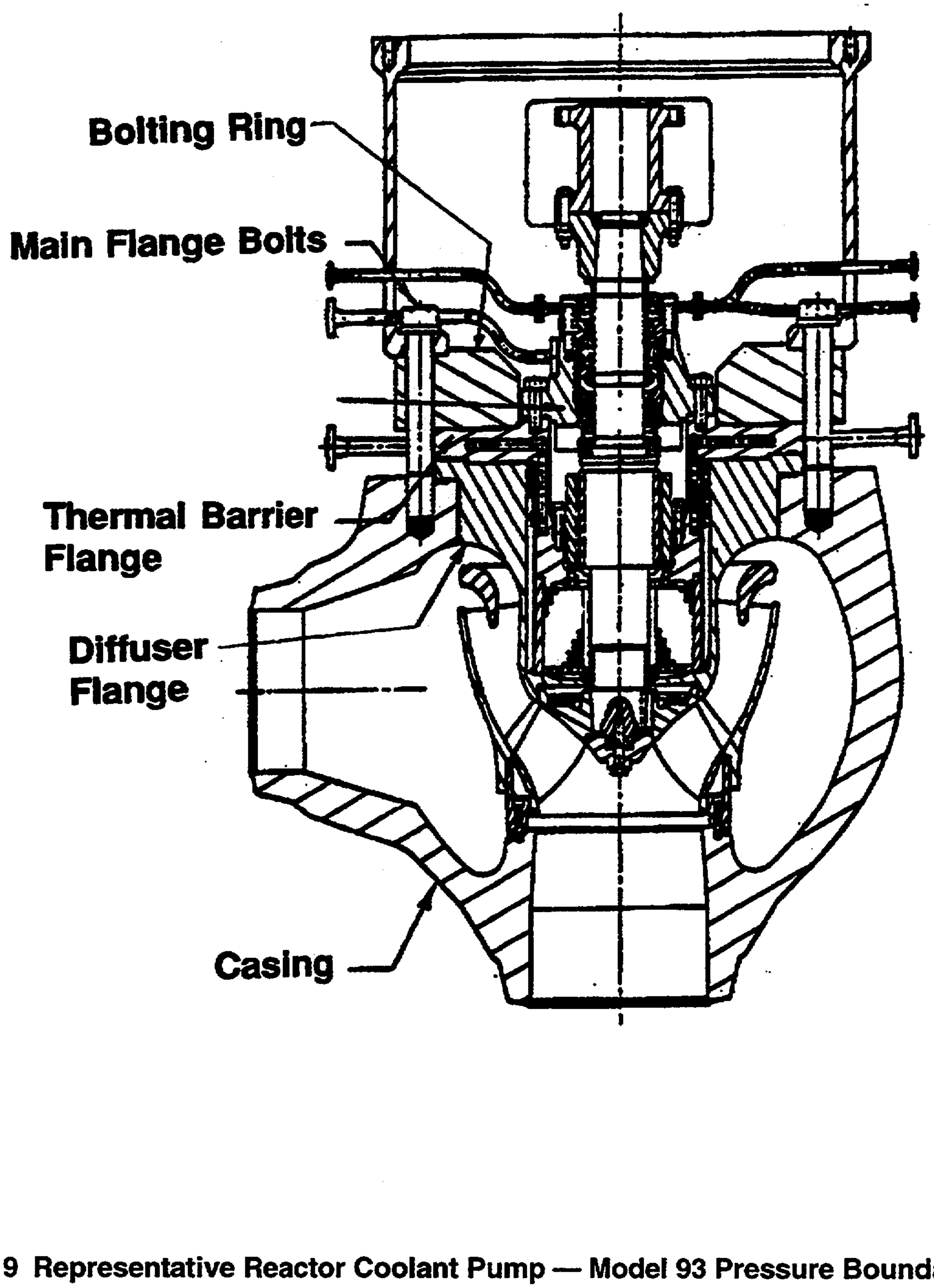


Figure 2-19 Representative Reactor Coolant Pump — Model 93 Pressure Boundary Components

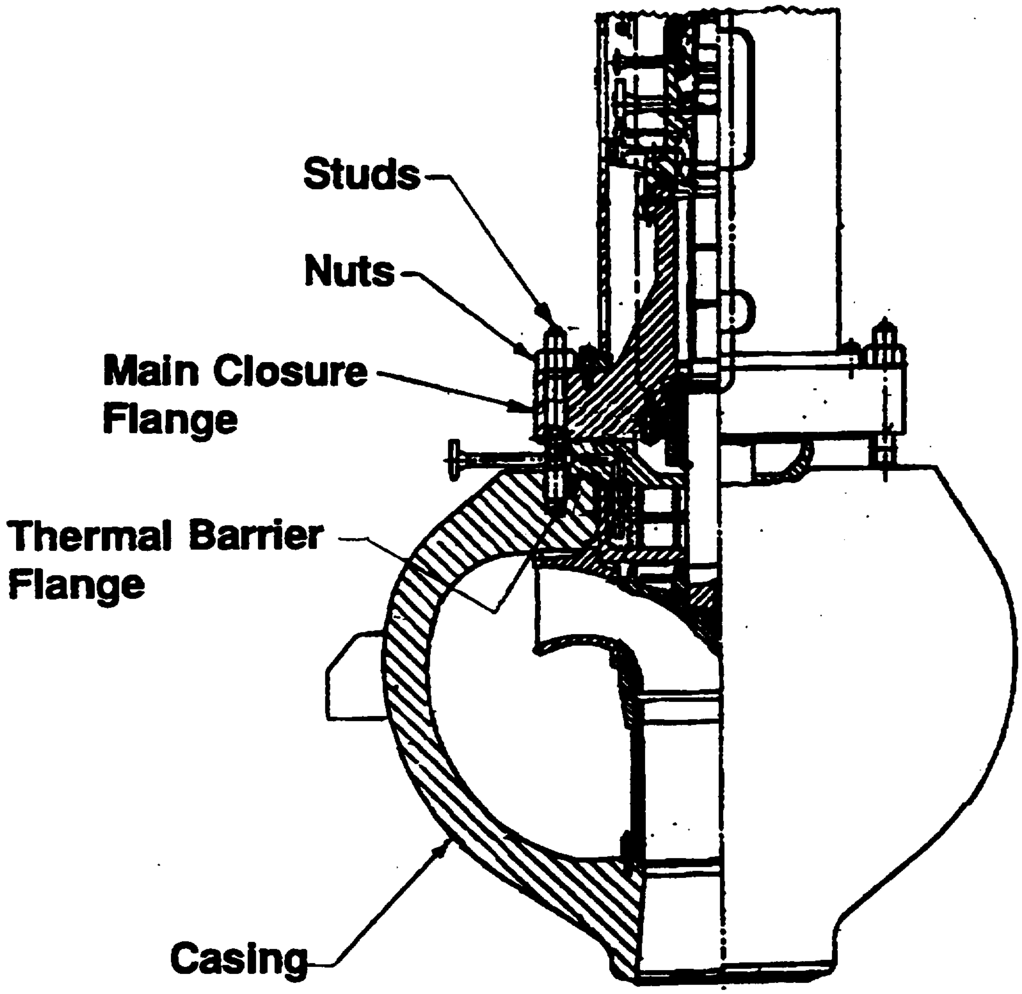


Figure 2-20 Representative Reactor Coolant Pump — Model M100 Pressure Boundary Components

RCP seals are also part of the pressure boundary. The seals are designed to leak during all operations. During normal operations, Class 1 seal water injection lines inject approximately 8 gpm into the no. 1 seal area. This flow splits, with 5 gpm going into the RCS and 3 gpm bypassing and cooling the no. 1 seals. In the event charging flow is lost and the thermal barrier heat exchanger is functioning, the seal will leak cool water at 3 gpm. However, this leak will be reactor coolant water rather than charging water. The 3 gpm is within the normal reactor coolant makeup capacity. If both the charging flow and component cooling flow are lost, the 3 gpm leakage will be hot water that will have a deleterious effect on the RCP seals. This combination of events is considered an operation event and not an aging event, and thus will not be discussed further in this report. For more detailed information, refer to WCAP-10542 [Ref. 9]. Also, the RCP seals are a replaceable component and, as such, are exempt from license renewal.

2.3.2.5 RTD Fast-Response Thermowells and RCS Wide-Range Thermowells

All thermowells connected directly to the RCS are designed and fabricated to accommodate the system flow, pressures, and temperatures attained under all expected modes of plant operation and anticipated system interactions. Materials of construction are specified to minimize erosion and corrosion and ensure material compatibility within the operating environment. Threaded connections are used on all of the RCS wide-range thermowells and on some of the RTD fast-response thermowells that were designed for installation into RCL piping without draining. The remainder of the RTD fast-response thermowells are welded connections, which are a socket weld design. Thermowells are classified as passive components and designed to remain in service for the life of the plant.

2.3.2.6 Splitter Elbows and Plenum Elbows

Early plant designs included splitter elbows to balance flow relocation into the RCP. Alternatively, plenum elbows were used for later plant designs (see Figure 2-21).

2.3.2.7 Nozzles and Thermal Sleeves

All nozzles connected directly to the RCS are designed and fabricated to accommodate the system pressures and temperatures attained under all expected modes of plant operation and anticipated system interactions. Construction materials are specified to minimize erosion and corrosion and to ensure material compatibility with the operating environment. Nozzles are classified as passive components and are designed to remain in service for the life of the plant.

Nozzles with thermal sleeves are designed to protect the nozzle from thermal shock. The stagnate layer of fluid captured between the thermal sleeve and nozzle protects the nozzle against sudden surges of cool or hot fluid. Junctions between the charging/safety injection nozzles and the cold leg are usually protected by thermal sleeves. For example, thermal sleeves may be used in the charging line nozzle and PZR surge nozzle. The PZR thermal sleeves, located in the surge line and spray line, are designed to minimize stresses on the PZR line nozzles. The thermal sleeves are classified as passive components and are designed to remain in service for the life of the plant.

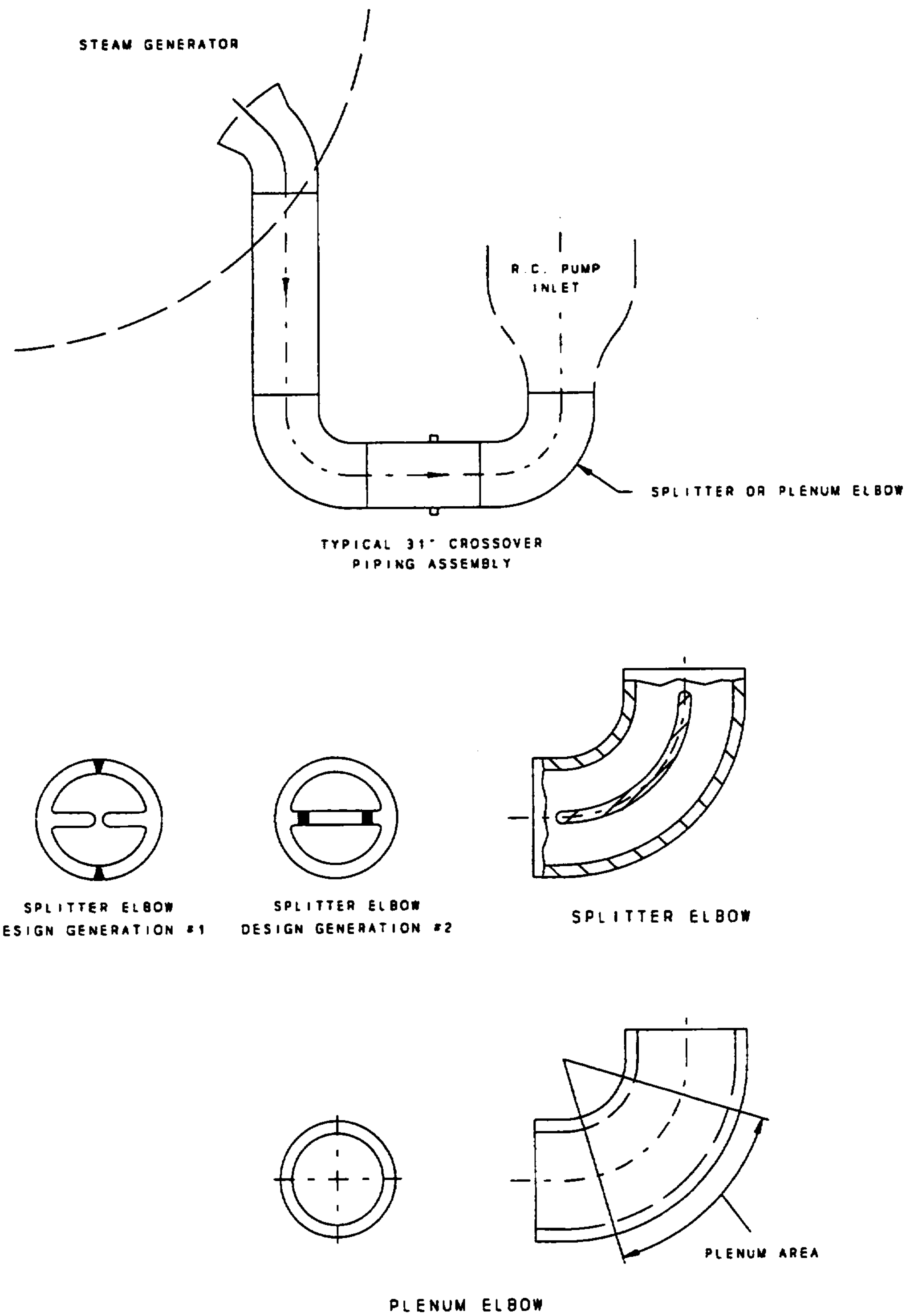


Figure 2-21 Reactor Coolant Loop Splitter and Plenum Elbows

2.3.2.8 Trunnions, Lugs, and Super-Stiff Clamps

Trunnions, lugs, and super-stiff clamps are designed to provide interface support from a component to some form of restraint or movement-limiting device. Lugs and trunnions are welded to the piping in accordance with the welding standard specified to match the piping size and material. Trunnions, lugs, and super-stiff clamps are designed to accommodate the thermally induced fatigue cycles associated with the system to which they are attached. These components are designed for the life of the plant and are passive. Super-stiff clamps depend on the friction developed by a bolted connection and are an exception to welded design.

In considering trunnions, lugs or welded attachments, and super-stiff clamps, this evaluation addresses only the aging effects these items may have on the piping pressure-retaining function; this evaluation does not address the support functions of the components. The support functions are addressed in a separate report [Ref. 2].

2.4 ENGINEERING AND DESIGN DATA

To start the aging management review, it is necessary to obtain plant data such as the engineering design and operational history of Class 1 piping systems. Data have been retrieved from existing Westinghouse files and supplemented by information from utilities. The following subsections discuss both Class 1 piping and the associated pressure boundary components.

Tables 2-2a and 2-2b present a summary listing of RCS piping dimensions and design parameters for Westinghouse commercial PWR units.

2.4.1 Materials

The reactor coolant piping and fittings that make up the loops are stainless steel. The piping is either seamless forged or centrifugally cast, and the fittings are statically cast without longitudinal welds or electroslag welds, except for some splitter elbow designs that may consist of a two-piece construction. The material specifications for the RCL piping are presented in Table 2-3.

The reactor coolant pump (RCP) casings and Class 1 valve bodies are also stainless steel. RCP casings are statically cast and some early designs may be two-piece welded construction. Class 1 valve bodies are either statically cast or forged. See Subsection 2.4.4 for material selection.

2.4.2 Design Specifications

Tables 2-4a and 2-4b summarize the design specifications for the RCL piping. Note that design specifications for analysis are not shown for all plants. Early B31.1 plant designs did not require analysis design specifications. For plants not within the Westinghouse scope, the analysis specifications were provided by utilities and their architects/engineers (AEs) and were not generally available to Westinghouse. Typically, RCS piping analysis design specifications reference the design specifications for Class 1 valves and RCPs. For some plants, additional specifications may be available for other portions of Class 1 piping.

TABLE 2-2a
REACTOR COOLANT PIPING DIMENSIONS

Reactor Inlet Piping, Inside Diameter, inches	27.5
Reactor Inlet Piping, Nominal Wall Thickness, inches	2.32
Reactor Outlet Piping, Inside Diameter, inches	29
Reactor Outlet Piping, Nominal Wall Thickness, inches	2.45
Coolant Pump Suction Piping, Inside Diameter, inches	31
Coolant Pump Suction Piping, Nominal Wall Thickness, inches	2.60
Pressurizer Surge Line Piping, Nominal Pipe Size, inches	14
Pressurizer Surge Line Piping, Nominal Wall Thickness, inches	1.406

Note:

The dimensions shown may vary depending on size of the plant, the plant code, and material selection.

TABLE 2-2b
REACTOR COOLANT PIPING SYSTEM DESIGN PARAMETERS

Line	System Design Pressure, psig	System Design Temperature, °F
Reactor Coolant Loop	2485	650
Pressurizer Surge	2485	680
Pressurizer Safety Valve Inlet	2485	680
Pressurizer Power-Operated Relief Valve Inlet	2485	680

Note:

System design pressure and temperature are ASME, Section III, B31.7, and B31.1 piping design terms and are the maximum values of pressure and temperature that the system is expected to experience during operating and accident conditions. (Hydrotesting pressure may exceed the design pressure limit.) Different components including flanges, fittings, and valve bodies may have design codes (i.e., B16.5, B16.11) with such expressions as a 2"-6000# nominal size. Designers must correlate the nominal size to the pressure and temperature system design limits for the selected material. For example, a 6000# flange might be the smallest acceptable nominal size in a piping system with a 2485 psi system design pressure at a 680°F system design temperature.

TABLE 2-3
REACTOR COOLANT LOOP PIPING MATERIALS

Plant Name	Pipe Material				Fitting Material	
	316	304N	CF8M	CF8A	CF8M	CF8A
Beaver Valley 1			✔		✔	
Beaver Valley 2				✔		✔
Braidwood 1		✔				✔
Braidwood 2		✔				✔
Byron 1		✔				✔
Byron 2		✔				✔
Callaway				✔		✔
Catawba 1				✔		✔
Catawba 2				✔		✔
Comanche Peak 1				✔		✔
Comanche Peak 2				✔		✔
Diablo Canyon 1	✔				✔	
Diablo Canyon 2	✔				✔	
Donald C. Cook 1			✔		✔	
Donald C. Cook 2			✔		✔	
Farley 1				✔	✔	
Farley 2				✔		✔
Ginna	✔				✔	
Indian Point 2	✔				✔	
Indian Point 3	✔				✔	

TABLE 2-3 (Continued)
REACTOR COOLANT LOOP PIPING MATERIALS

Plant Name	Pipe Material				Fitting Material	
	316	304N	CF8M	CF8A	CF8M	CF8A
Kewaunee			✔		✔	
McGuire 1				✔		✔
McGuire 2				✔		✔
Millstone 3				✔		✔
North Anna 1				✔	✔	
North Anna 2				✔	✔	
Point Beach 1	✔				✔	
Point Beach 2	✔				✔	
Prairie Island 1	✔				✔	
Prairie Island 2			✔		✔	
Robinson 2	✔				✔	
Salem 1	✔				✔	
Salem 2	✔				✔	
Seabrook		✔				✔
Sequoyah 1			✔		✔	
Sequoyah 2			✔		✔	
Shearon Harris		✔				✔
South Texas Project 1				✔		✔
South Texas Project 2				✔		✔

TABLE 2-3 (Continued)
REACTOR COOLANT LOOP PIPING MATERIALS

Plant Name	Pipe Material				Fitting Material	
	316	304N	CF8M	CF8A	CF8M	CF8A
Summer		✔				✔
Surry 1	✔				✔	
Surry 2	✔				✔	
Trojan (SHUTDOWN)			✔		✔	
Turkey Point 3	✔				✔	
Turkey Point 4	✔				✔	
Vogtle 1				✔		✔
Vogtle 2				✔		✔
Watts Bar 1				✔		✔
Watts Bar 2				✔		✔
Wolf Creek				✔		✔
Zion 1	✔				✔	
Zion 2	✔				✔	

TABLE 2-4a
REACTOR COOLANT PIPING SPECIFICATIONS —
COMPLIANCE WITH B31.1 POWER PIPING CODE

Plant Name	Pipe	F(1)/CC(2)	Fittings(3)	Shop Fab	Analysis
Beaver Valley 1	G-676580 Rev 2	CC	G-676342 Rev 4	G-676343 Rev 3	
Diablo Canyon 1	G-676341 Rev 0	F	G-676342 Rev 2	G-676343 Rev 1	952595 Rev 0
Diablo Canyon 2	G-676341 Rev 1	F	G-676342 Rev 4	G-676343 Rev 3	
Donald C. Cook 1	G-676580 Rev 2	CC	G-676342 Rev 4	G-676343 Rev 3	
Donald C. Cook 2	G-676580 Rev 2	CC	G-676342 Rev 3	G-676343 Rev 3	
Ginna	G-676341 Rev 0	F	G-676342 Rev 0	G-676343 Rev 0	
Indian Point 2	G-676341 Rev 0	F	G-676342 Rev 0	G-676343 Rev 0	
Indian Point 3	G-676341 Rev 0	F	G-676342 Rev 2	G-676343 Rev 1	
Kewaunee	G-676580 Rev 2	CC	G-676342 Rev 4	G-676343 Rev 3	
Point Beach 1	G-676341 Rev 0	F	G-676342 Rev 0	G-676343 Rev 0	
Point Beach 2	G-676341 Rev 0	F	G-676342 Rev 2	G-676343 Rev 1	
Prairie Island 1	G-676341 Rev 1	F	G-676342 Rev 2	G-676343 Rev 1	
Prairie Island 2	G-676580 Rev 2	F	G-676342 Rev 4	G-676343 Rev 3	
Robinson 2	G-676341 Rev 0	F	G-676342 Rev 0	G-676343 Rev 0	
Salem 1	G-676341 Rev 0	F	G-676342 Rev 2	G-676343 Rev 1	
Salem 2	G-676341 Rev 1	F	G-676342 Rev 4	G-676343 Rev 3	
Sequoyah 1	G-676580 Rev 2	CC	G-676342 Rev 3	G-676343 Rev 3	952768 Rev 0
Sequoyah 2	G-676580 Rev 2	CC	G-676342 Rev 3	G-676343 Rev 3	

TABLE 2-4a (Continued)
REACTOR COOLANT PIPING SPECIFICATIONS —
COMPLIANCE WITH B31.1 POWER PIPING CODE

Plant Name	Pipe	F[1]/CC[2]	Fittings[3]	Shop Fab	Analysis[4,5,6]
Surry 1	G-676341 Rev 0	F	G-676342 Rev 2	G-676343 Rev 1	
Surry 2	G-676341 Rev 0	F	G-676342 Rev 2	G-676343 Rev 1	
Trojan	G-676580 Rev 2 679076 Rev 0	CC	G-676342 Rev 4 679045 Rev 1	G-677387 Rev 2 679097 Rev 0	678856 Rev 4
Turkey Point 3	G-676341 Rev 0	F	G-676342 Rev 0	G-676343 Rev 0	
Turkey Point 4	G-676341 Rev 0	F	G-676342 Rev 2	G-676343 Rev 1	
Zion 1	G-676341 Rev 1	F	G-676342 Rev 3	G-676343 Rev 3	
Zion 2	G-676341 Rev 1	F	G-676342 Rev 3	G-676343 Rev 3	

<u>Notes</u>:

1. The pipe specification for forged pipe.
2. The pipe specification for centrifugally cast pipe.
3. All fittings are statically cast.
4. The analysis specifications listed are for the RCS. There may be additional specifications for other Class 1 piping systems.
5. Analysis specifications are not required for B31.1 plants.
6. Westinghouse did not provide all of the RCS analysis specifications. Additional RCS analysis specifications may have been provided by the architect engineer.

TABLE 2-4b
REACTOR COOLANT PIPING SPECIFICATIONS —
COMPLIANCE WITH ASME B&PV CODE OR B31.7

Plant Name	Pipe(7)	F(1)/CC(2)	Fittings(3,7)	Shop Fab(7)	Analysis(4,5,6)
Beaver Valley 2	G-678864 Rev 1-I1 679199 Rev 1	CC	G-678865 Rev 2 679147 Rev 2	G-678843 Rev 1 679191 Rev 2-I3	
Braidwood 1	G-678866 Rev 2 679107 Rev 3	F	G-678865 Rev 2 679145 Rev 3	G-678843 Rev 3 953396 Rev 0-I2	953456 Rev 1
Braidwood 2	G-678866 Rev 2 679107 Rev 3	F	G-678865 Rev 2 679145 Rev 3	G-678843 Rev 3 953396 Rev 0-I2	953456 Rev 0
Byron 1	G-678866 Rev 2 679107 Rev 3	F	G-678865 Rev 2 679145 Rev 3	G-678843 Rev 1 679190 Rev 2	953456 Rev 0
Byron 2	G-678866 Rev 2 679107 Rev 3	F	G-678865 Rev 2 679145 Rev 3	G-678843 Rev 1 679190 Rev 2	953456 Rev 1
Callaway	G-678864 Rev 4 953202 Rev 1	CC	G-678865 Rev 4 953065 Rev 1-I1	G-678843 Rev 3 953346 Rev 1-I1	
Catawba 1	G-678864 Rev 1-I1 679200 Rev 2	CC	G-678865 Rev 2 679163 Rev 3	G-678843 Rev 1 679189 Rev 1-I1	
Catawba 2	G-678864 Rev 1-I1 679200 Rev 2	CC	G-678865 Rev 2 679163 Rev 3	G-678843 Rev 1 679189 Rev 1-I1	
Comanche Peak 1	G-678864 Rev 4 953181 Rev 0	CC	G-678865 Rev 4 953063 Rev 1-I1	G-678843 Rev 3 953245 Rev 1	955125 Rev 1
Comanche Peak 2	G-678864 Rev 4	F	G-678865 Rev 4 953063 Rev 1-I1	G-678843 Rev 3 953245 Rev 1	955125 Rev 1
Farley 1	G-678864 Rev 1 679039 Rev 3	CC	G-676342 Rev 4 679075 Rev 0	G-678843 Rev 1 679098 Rev 1	952445 Rev 1
Farley 2	G-678864 Rev 1 679198 Rev 2	CC	G-678865 Rev 2 679139 Rev 1	G-678843 Rev 1 679195 Rev 2	955131 Rev 0
McGuire 1	G-678864 Rev 1 679015 Rev 2	CC	G-678865 Rev 2 679013 Rev 4	G-678843 Rev 1 679169 Rev 4	
McGuire 2	G-678864 Rev 1 679016 Rev 2	CC	G-678865 Rev 2 679013 Rev 4	G-678843 Rev 1 679169 Rev 4	
Millstone 3	G-678864 Rev 1-I1 679203 Rev 5	CC	G-678865 Rev 2 679161 Rev 3	G-678843 Rev 3 679194 Rev 2-I1	
North Anna 1	G-678864 Rev 1 952289 Rev 0	CC	G-676342 Rev 4 679041 Rev 0	G-677387 Rev 2	953100 Rev 0
North Anna 2	G-678864 Rev 1 952289 Rev 0	CC	G-676342 Rev 4 679041 Rev 0	G-677387 Rev 2	953100 Rev 0
Seabrook	G-678866 Rev 3 952313 Rev 2	F	G-678865 Rev 4 952310 Rev 2	G-678843 Rev 3 952318 Rev 1	953182 Rev 0
Shearon Harris	G-678866 Rev 3 679106 Rev 4	F	G-678865 Rev 2 679144 Rev 4	G-678843 Rev 1 679187 Rev 1-I1	

TABLE 2-4b (Continued)
REACTOR COOLANT PIPING SPECIFICATIONS — COMPLIANCE WITH ASME B&PV CODE OR B31.7

Plant Name	Pipe(7)	F(1)/CC(2)	Fittings(3,7)	Shop Fab(7)	Analysis(4,5,6)
South Texas Project 1	G-678864 Rev 4 953193 Rev 0	CC	G-678865 Rev 4 953064 Rev 1	G-678843 Rev 3 953231 Rev 1	953385 Rev 0
South Texas Project 2	G-678864 Rev 4 953193 Rev 0	CC	G-678865 Rev 4 953064 Rev 1-I1	G-678843 Rev 3 953231 Rev 1	953385 Rev 0
Summer	G-678866 Rev 2 679204 Rev 2	F	G-678865 Rev 2 679146 Rev 3	G-678843 Rev 1 679186 Rev 1	955136 Rev 0
Vogtle 1	G-678864 Rev 4 679202 Rev 1	CC	G-678865 Rev 2 679162 Rev 2	G-678843 Rev 3 679196 Rev 3	955118 Rev 0
Vogtle 2	G-678864 Rev 4 679202 Rev 1	CC	G-678865 Rev 2 679162 Rev 2	G-678843 Rev 3 679196 Rev 3	955118 Rev 0
Watts Bar 1	G-678864 Rev 1 679183 Rev 2	CC	G-678865 Rev 2 679148 Rev 3	G-678843 Rev 1 679170 Rev 4	
Watts Bar 2	G-678864 Rev 1 679183 Rev 2	CC	G-678865 Rev 2 679148 Rev 3	G-678843 Rev 1 679170 Rev 4	
Wolf Creek	G-678864 Rev 4 953202 Rev 1	CC	G-678865 Rev 4 953065 Rev 1-I1	G-678843 Rev 3 953346 Rev 1-I1	

Notes:

1. The pipe specification for forged pipe.
2. The pipe specification for centrifugally cast pipe.
3. All fittings are statically cast.
4. The analysis specifications listed are for the RCS. There may be additional specifications for other Class 1 piping systems.
5. Analysis specifications are not required for B31.1 plants.
6. Westinghouse did not provide all of the RCS analysis specifications. Additional RCS analysis specifications may have been provided by the architect engineer.
7. Interim revisions are designated by an "I" followed by the number.

All smaller piping that comprise part of the RCS, such as the PZR surge line, spray and relief lines, loop drains, and connecting lines to other systems, are also austenitic stainless steel. All joints and connections are welded, except for the PZR safety valves and RTD bypass lines where flanged joints are used.

Thermal sleeves are installed at points in the system where high thermal stresses could develop due to rapid changes in fluid temperature during normal operational transients. These points include:

- PZR surge line connection at the PZR
- PZR spray line connection at the PZR
- Charging/ECCS injection connections

Piping connections from auxiliary systems (see Figure 2-22) are usually made above the horizontal centerline of the reactor coolant piping, with the exception of:

- Residual heat removal pump suction lines, which are 45 degrees down from the horizontal centerline.
- This orientation is preferred for gravity feed and to prevent vortex shedding at the branch opening. This enables the water level in the RCS to be lowered in the reactor coolant pipe while continuing to operate the RHRS, should this be required for maintenance.
- Loop drain lines and the connection for temporary level measurement of water in the RCS during refueling and maintenance operation.
- The differential pressure taps for flow measurement, which are downstream of the steam generators (SGs) on the first 90-degree RCL elbow.
- The PZR surge line, which is attached at the horizontal centerline.
- The ECCS connections to the hot leg, for which inservice inspection requirements and space limitations dictate location on the horizontal centerline.

Penetrations into the coolant flow path are limited to the following:

- PZR spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the RCL flow adds to the spray driving force.
- Reactor coolant sample system taps protrude into the main stream to obtain a representative sample of the reactor coolant.
- RTD hot leg scoops extend into the reactor coolant at locations 120 degrees apart in the cross-sectional plane. In the original design, these scoops collected a representative temperature sample for the RTD manifold. In the current design, they provide a convenient location for the RCS fast-response RTD thermowells.

REACTOR COOLANT SYSTEM PENETRATIONS

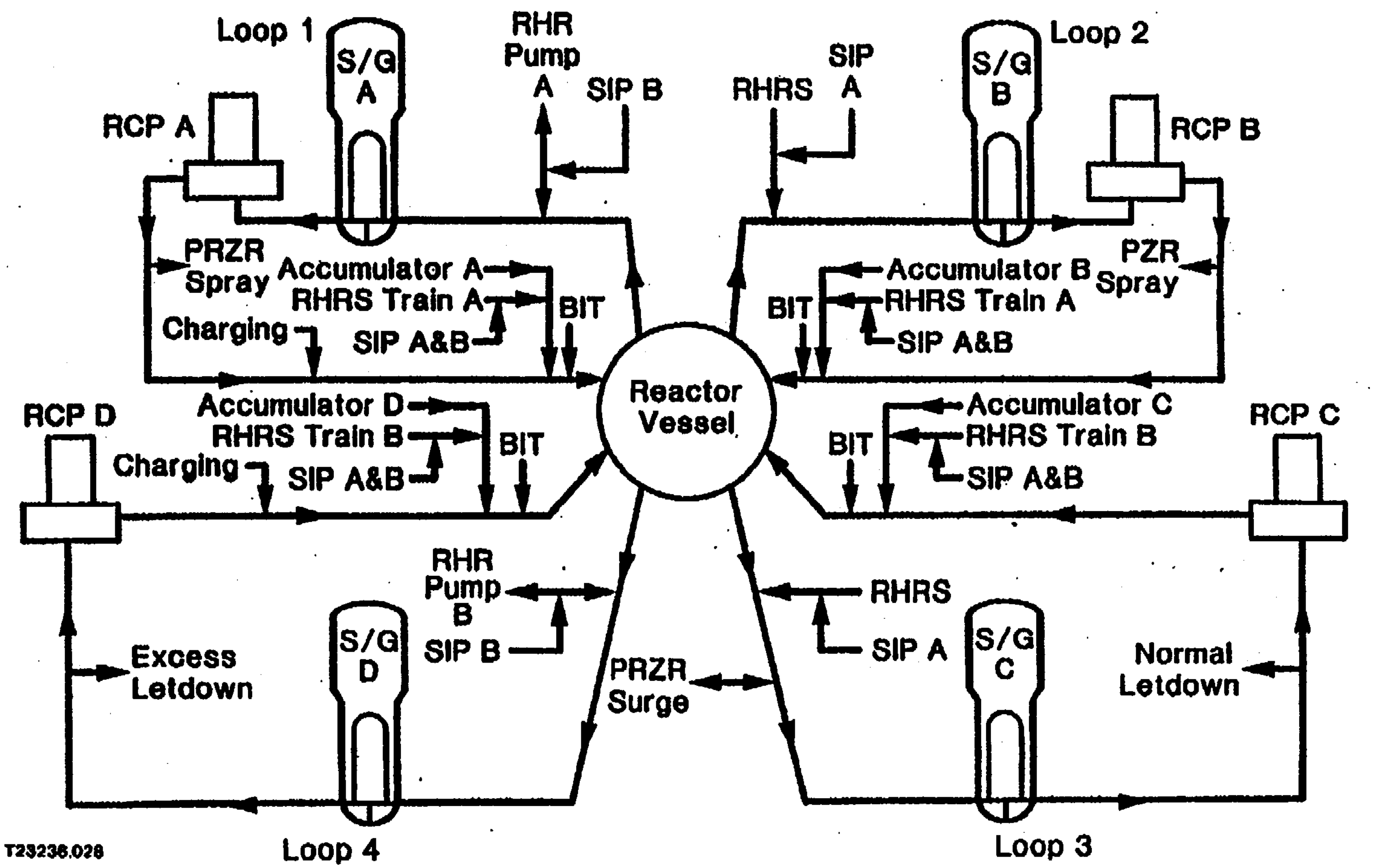


Figure 2-22 Reactor Coolant System Penetrations

- The hot and cold leg wide-range thermowells and the cold leg fast-response RTD thermowells extend into the reactor coolant pipes. For some plants, new penetrations on the hot and cold legs may be added for the RTD thermowells if the existing penetrations are not adequate.

2.4.3 Basic Sizing

The RCL piping inside diameter was sized to minimize the reactor coolant volume and hydraulic losses. The cold leg is cooler and thus contains fluid with a higher density. Therefore, it is slightly smaller (27.5-inch ID) than the hot leg (29-inch ID). The size of the crossover leg (31-inch ID) was determined from pump characteristics. Some plants have RCP inlet elbows with splitters to balance the velocity profile and prevent excess suction losses. Alternatively, plenum elbows were used (see Figure 2-21).

2.4.4 Material Selection

The materials are as shown in Tables 2-5, 2-6, and 2-7. These materials were selected from allowable code materials considering environmental requirements, inspection requirements, and cost. Environmental considerations include chemical properties of reactor coolant fluid and exposure to radiation.

The Class 1 piping and weld material data shown in Table 2-5 includes the Class 1 pipe fittings, nozzles, flanges, and welds that are discussed in this evaluation.

The RCP material data shown in Table 2-6 include the RCP casing and associated pressure boundary components that are discussed in this evaluation. This list does not cover all of the RCP pressure boundary components. Also not included are the auxiliary nozzles, all seal housing components and heat exchanger components that are welded or bolted to the pressure boundary components. The remainder of the RCP, other than the casing, can be replaced by a new pump assembly (see Subsection 3.3.5). There are four RCP models that are used in domestic Westinghouse NSSS plants (see Figures 2-7 through 2-20). The M93 RCP model is not ASME Code stamped.

Class 1 valve material data shown in Table 2-7 includes Class 1 valve body and associated pressure boundary components that are discussed in this report.

2.4.5 Welding Processes

Welding processes used to fabricate the RCL piping include gas tungsten arc welding (GTAW), shielded manual arc welding (SMAW), and submerged arc welding (SAW). The GTAW process was used for root closure welds and small (2-inch and smaller) girth welds. The GTAW process yields a high-quality weld with a slow deposition rate. During welding, argon gas flows over the weld to protect it. The SMAW process was used for large nozzles and for field butt welds. The SMAW process has a higher deposition rate. The SAW process was used on large girth shop welds for pipe-to-fitting welds as well as pipe-to-pipe welds. The SAW is an automatic process.

TABLE 2-5
CLASS 1 PIPING AND WELD MATERIALS

Piping Item	Material Specification
RCL Piping	
Wrought Seamless Pipe	A-376 TP 316, SA-376 TP 304N
Centrifugally Cast pipe	A-351 TP CF8M, SA-351 CF8A
Cast Fittings (Including 45° branch outlet)	A/SA-351 TP CF8M, SA-351 CF8A
Branch Nozzles	A/SA182 F316 SA-182 F316N, SA-182 F304N
Auxiliary Line Piping: same as above plus	
Wrought Seamless Pipe	SA-376 TP 304, TP-304L, A-376 TP304
Fittings	SA-351 F316, F316L, F304, F304L, SA182 F304, F316, A182 F304, A/SA403 WP304, WP316
Flanges	A/SA-182 F316
Welds	Type 308 weld filler metal

TABLE 2-6
REACTOR COOLANT PUMP MATERIALS

RCP Model	Component	Material Specification
M93	Casing	ASTM A-351-63T,CF8M (316 SST Casting)
	Thermal Barrier Flange	ASTM A-182,F304
	Main Closure Flange	ASTM A-351, CF8 or ASTM A-182,F304
	Bolts (Casing)	ASTM A-193, B7 or B16
	Nuts	ASTM A-194, Grade 4
M93A	Casing	SA-351, CF8 (304 SST Casting)
	Thermal Barrier Flange	SA-182,F304
	Main Closure Flange	SA-351, CF8
	Bolts (Casing)	SA-540 Grade B24 (Most are Class 4)
M93A-1	Casing	SA-351, CF8 (304 SST Casting)
	Thermal Barrier Flange	SA-182,F304
	Bolting Ring	SA-508, Class 2
	Bolts (Casing)	SA-540, Grade B24 Class 4
M100	Casing	SA-351, CF8 (304 SST Casting)
	Thermal Barrier Flange	SA-182,F304
	Diffuser Flange	SA-351, CF8
	Bolting Ring	SA-508, Class 2
	Bolts (Casing)	SA-540, Grade B24 Class 4

TABLE 2-7
CLASS 1 VALVE MATERIALS

Class 1 Valve Component	Material Specification
Body	A/SA-182 F316
	A/SA-351 CF8M
	A-182 F304
	Stellite on some bodies
Bonnet	A/SA-182 F316
	A/SA-351 CF8M
	SA-240 Type 304
Closure Bolting	A/SA-453 Gr 660
	A-193 Gr 6
	SA-194 Gr 6
	A-193 Gr B7
	A-194 Gr 2H

Quality assurance (QA) examinations applied to pipes, fittings, and welds include radiography (RT), liquid penetrant (LP), ultrasonic testing (UT), and visual examination (VT).

Similar welding and quality control measures are used for other components that make up the Class 1 pressure boundary discussed in the GTR.

The choices of materials, welding, and QA measures during fabrication minimizes the aging concerns on Class 1 piping and components. Specific discussions on aging can be found in Section 3.0.

2.4.6 Mechanical Design

The engineering design of the piping was made by applying the design philosophies and design rules of mechanical design codes. The ASME USAS B31.1 Power Piping Code was used for plants designed before 1969, the ASME USAS B31.7 Nuclear Power Piping and the ASME Pump and Valve Codes were used for plants designed between 1969 and 1971, and the ASME Boiler and Pressure Vessel Code, Section III was used for plants designed since 1971.

The first step in design, as required by either the B31.1, B31.7, or ASME Codes, is to determine the minimum wall thickness ™ from design pressure, design temperature, and material properties. The equations for these calculations are designed to provide conservative protection against pipe rupture under pressure. The equations are:

B31.1	Equation 2	$t_m = PD_o/(2[S_h + Py]) + A$
B31.7	Equation 1	$t_m = PD_o/(2[S_m + Py]) + A$
ASME Code	Equation 2	$t_m = PD_o/(2[S_m + Py]) + A$

where:

P	=	Design pressure of the piping system
D_0	=	Outside diameter of the piping
y	=	Empirical constant
A	=	Corrosion allowance of the piping
S_h and S_m	=	Code-allowable stresses based on material properties and design temperature

The features of the basic pressure design affected by aging include material properties and wall thickness. The aging effect issues of material properties in a PWR environment such as neutron embrittlement, thermal aging embrittlement, and stress corrosion cracking will be discussed in Section 3.0. The possible thinning of the pipe wall due to erosion, corrosion, and wear will also be discussed in Section 3.0.

The next stage of the engineering design is to determine piping stresses due to mechanical and thermal loads. Piping stresses are calculated as a function of mechanical loads (including seismic), thermal loads, and pipe geometry. The calculated stresses must be less than allowables based on material properties at operating temperatures. Again, the features of the basic mechanical design that are affected by aging include material properties and wall thickness.

2.4.7 Fatigue Design

The fatigue design approach taken in ANSI B31.1 differs from that contained in ANSI B31.7, Pump and Valve Codes, or ASME, Section III Class 1 rules. The B31.7 and the Pump and Valve Codes are considered to be equivalent to ASME Code, Section III [Ref. 10] for fatigue design methodology. The ANSI B31.1 fatigue design methodology is based on an implicit treatment of cyclic loadings, through a stress reduction factor applied to the stress allowables that depends on the number of equivalent full thermal loading cycles anticipated during the service life of the component. The B31.1 code does not directly address issues such as through-wall temperature distribution and material discontinuities. It indirectly addresses geometric discontinuities by applying stress intensification factors to calculated stresses where these stress intensification factors are empirical but based on cyclic research. As partial compensation for the simplicity of the implicit fatigue design approach, the ANSI B31.1 rules require that allowable stresses be based on material strength properties that are more conservative than those applicable to ASME, Section III Class 1 designs.

ASME Code, Section III Class 1 design rules directly address cyclic stresses, both mechanical and thermal, which can cause mechanical failure. These calculations are generally referred to as fatigue analyses.

The fatigue design approach taken in ASME Code, Section III for Class 1 components is based on an explicit treatment of the cyclic loadings, both thermal and mechanical, and the associated stresses that are anticipated during the service life of the component. The design basis transients are defined as pressure and temperature changes versus time plus the number of seismic events. In addition to pressure and temperature changes in the fluid, the overall piping systems respond to temperature and seismic events, which in turn cause mechanical loads to be considered in the fatigue analysis. Typical RCS piping has a prescribed number of design basis transients, as identified in Table 2-8. Examples of RCS transients are given in Figures 2-23 and 2-24. Typical auxiliary system transients are identified in Table 2-9. Typical surge line transients are shown in Table 2-10. An example of an auxiliary transient is given in Figure 2-25. The piping in each system is designed for these transients, and the nozzles that interconnect the systems are designed to both sets of transients.

The two features of fatigue design that are affected by aging are the same as for other loadings, namely material properties and wall thickness. In addition, the component design-basis transients must be reviewed, in comparison with component operating history, to determine whether sufficient conservatism exists in the number and severity of the design-basis transients when the actual operating transients are extrapolated through the end of the license renewal term. The fatigue analysis is a time-limited aging analysis (TLAA) discussed in detail in Section 3.3.

There is no essential difference between the construction and operation of Class 1 components, whether they are designed to ANSI B31.1 or ASME, Section III Class 1 requirements. Therefore, extending the service lives of B31.1 components should not require major changes to components. A similar conclusion is supported by the "Fatigue Comparison of Piping Designed to ANSI B31.1 and ASME, Section III Class 1 Rules" [Ref. 11].

TABLE 2-8
REACTOR COOLANT LOOP TRANSIENTS

Description	Occurrences
RCP startup and shutdown	4000
Plant heatup and cooldown	200
Unit loading and unloading between 0-15% of full power	500
Unit loading and unloading at 5% of full power/minute	13,200
Reduced temperature return to power	2000
Step load increase and decrease of 10% of full power	2000
Large step load decrease with steam dump	200
Steady-state fluctuations	4.5×10^6
Boron concentration equalization	26,400
Feedwater cycling	2000
Loop out of service	80
Refueling	80
Turbine roll test	20
Primary-side leakage test	200
Secondary-side leakage test	80
Upset Conditions Transients	
Description	**Occurrences**
Loss of load	80
Loss of power	40
Partial loss of flow	80
Reactor trip from full power	400
Case A - with no inadvertent cooldown	230
Case B - with cooldown and no SI	160
Case C - with cooldown and SI	10
Inadvertent RCS depressurization	30
Inadvertent startup of an inactive loop	10
Control rod drop	80
Inadvertent safety injection actuation	60
Excessive feedwater flow	30
Emergency Condition Transients	
Description	**Occurrences**
Small loss-of-coolant accident (LOCA)	5
Small steam line break	5
Complete loss of flow	5

TABLE 2-8 (Continued)
REACTOR COOLANT LOOP TRANSIENTS

Description	Occurrences
Reactor coolant pipe break (large LOCA)	1
Large steam line break	1
Feedwater line break	1
Reactor coolant pump locked rotor	1
Control rod ejection	1
Steam generator tube rupture	1
Simultaneous steam line/feedwater line break	1
Test Conditions Transients	
Description	**Occurrences**
Primary-side hydrostatic test	200
Secondary-side hydrostatic test	10
Tube leakage test	800

TABLE 2-9
AUXILIARY SYSTEM TRANSIENTS

Description	Occurrences
RCS and Pressurizer Heatup - Steam Bubble and Water Solid Modes	200
RCS and Pressurizer Cooldown - Steam Bubble and Water Solid Modes	200
Plant Heatup - Reduced Spray Operation H_1 - Hot Leg Surge Line Nozzle Transient	200
Plant Heatup - Reduced Spray Operation H_2 - Hot Leg Surge Line Nozzle Transient	200
Plant Heatup - Full Spray Operation H_6 - Hot Leg Surge Nozzle Transient	200
Plant Cooldown - Auxiliary Spray Operation C_6 Pressurizer Surge Nozzle and Hot Leg Surge Line Nozzle Transients	200
Plant Cooldown - Auxiliary Spray Operation C_7 - Pressurizer Surge Nozzle and Hot Leg Surge Line Nozzle Transients	200
RTD Manifold Return Nozzle ▯ Maintenance Operations	60
Charging Nozzle - Charging and Letdown Flow Shutoff and Return to Service (120 gpm system)	60
Charging Nozzle - Letdown Flow Shutoff with Prompt Return to Service (120 gpm system)	200
Charging Nozzle - Letdown Flow Shutoff with Delayed Return to Service (120 gpm system)	20
Charging Nozzle - Charging Flow Shutoff with Prompt Return to Service (120 gpm system)	20
Charging Nozzle - Charging Flow Shutoff with Delayed Return to Service (120 gpm system)	20
Charging Nozzle - Charging Flow Step Decrease and Return to Normal (120 gpm system)	24,000
Charging Nozzle - Charging Flow Step Increase and Return to Normal (120 gpm system)	24,000
Charging Nozzle - Letdown Flow Step Decrease and Return to Normal (120 gpm system)	2000
Charging Nozzle - Letdown Flow Step Increase and Return to Normal (120 gpm system)	24,000

TABLE 2-9 (Continued)
AUXILIARY SYSTEM TRANSIENTS

Description	Number of Occurrences
Charging Nozzle - Charging and Letdown Flow Shutoff and Return to Service (250 gpm system)	60
Charging Nozzle - Letdown Flow Shutoff with Prompt Return to Service (250 gpm system)	200
Charging Nozzle - Letdown Flow Shutoff with Delayed Return to Service (250 gpm system)	20
Charging Nozzle - Charging Flow Shutoff with Prompt Return to Service (250 gpm system)	20
Charging Nozzle - Charging Flow Shutoff with Delayed Return to Service (250 gpm system)	20
Charging Nozzle - Charging Flow Step Decrease and Return to Normal (250 gpm system)	24,000
Charging Nozzle - Charging Flow Step Increase and Return to Normal (250 gpm system)	24,000
Charging Nozzle - Letdown Flow Step Decrease and Return to Normal (250 gpm system)	2000
Charging Nozzle - Letdown Flow Step Increase and Return to Normal (250 gpm system)	24,000
Accumulator Nozzle, 212 Plant (2 sheets)	394
Accumulator Nozzle, 312 Plant	25
Accumulator Nozzle, 412 Plant (2 sheets)	394
Accumulator Nozzle, 414 Plant (2 sheets)	359
Safety Injection Nozzle, 312 Plant (2 sheets)	410
Reactor Vessel S.I. Nozzle, 212 Plant	110
Boron Injection Nozzle, 412 Plant	110
Charging and Letdown Flow Shutoff and Return to Service (120 gpm system)	60
Letdown Flow Shutoff with Prompt Return to Service (120 gpm system)	200
Letdown Flow Shutoff with Delayed Return to Service(120 gpm system)	20
Charging Flow Shutoff with Prompt Return to Service	20
Charging Flow Shutoff with Delayed Return to Service (120 gpm system)	20
Charging Flow Step Decrease and Return to Service (120 gpm system)	24,000
Charging Flow Step Increase and Return to Normal (120 gpm system)	24,000
Letdown Flow Step Decrease and Return to Normal (120 gpm system)	2000
Letdown Flow Step Increase and Return to Normal (120 gpm system)	24,000

TABLE 2-9 (Continued)
AUXILIARY SYSTEM TRANSIENTS

Plant Cooldown (120 gpm system) Plant Heatup (120 gpm system)	200 200
Component Cooling Water to Letdown HX Shutoff (120 gpm system)	200
Load Follow Boration Cycle for Letdown Reheat Heat Exchangers (120 gpm system)	24,000
Excess Letdown Heat Exchanger Transient During Maintenance Operations (120 gpm system)	100
Charging and Letdown Shutoff and Return to Service (250 gpm system)	60
Letdown Flow Shutoff with Prompt Return to Service (250 gpm system)	200
Letdown Flow Shutoff with Delayed Return to Service (250 gpm system)	20
Charging Flow Shutoff with Prompt Return to Service (250 gpm system)	20
Charging Flow Shutoff with Delayed Return to Service (250 gpm system)	20
Charging Flow Step Decrease and Return to Normal (250 gpm system)	24,000
Charging Flow Step Increase and Return to Normal (250 gpm system)	24,000
Letdown Flow Step Decrease and Return to Normal (250 gpm system)	2000
Letdown Flow Step Increase and Return to Normal (250 gpm system)	24,000
Plant Cooldown (250 gpm system)	200
Plant Heatup (250 gpm system)	200
Component Cooling Water to Letdown Heat Exchanger Shutoff (250 gpm system)	200
Load Follow Boration Cycle for Letdown Reheat Heat Exchanger (250 gpm system)	24,000
Excess Letdown Heat Exchanger Transient During Maintenance Operations (250 gpm system)	100

TABLE 2-10
SURGE NOZZLE TRANSIENTS

Transient Operation	Spray Rate (Normalized)	Spray Duration (Seconds)	T_{PRES} (°F)	T_{RCS} (°F)	ΔT (°F)
H_1-Reduced Spray	1.0	600	425	125	300
H_2-Reduced Spray	0.1	600	425	225	200
H_3-Full Spray	1.0	150	425	325	100
H_4-Full Spray	1.0	150	485	385	100
H_5-Full Spray	1.0	150	545	445	100
H_6-Full Spray	1.0	150	605	505	100
C_1-Full Spray	1.0	150	650	550	100
C_2-Full Spray	1.0	150	550	450	100
C_3- Full Spray	1.0	150	450	350	100
C_4-Reduced Spray	0.1	600	425	225	200
C_5-Reduced Spray	0.1	600	425	175	250
C_6-Auxiliary Spray	0.15	800	425	160	265
C_7-Auxiliary Spray	0.15	800	300	120	180

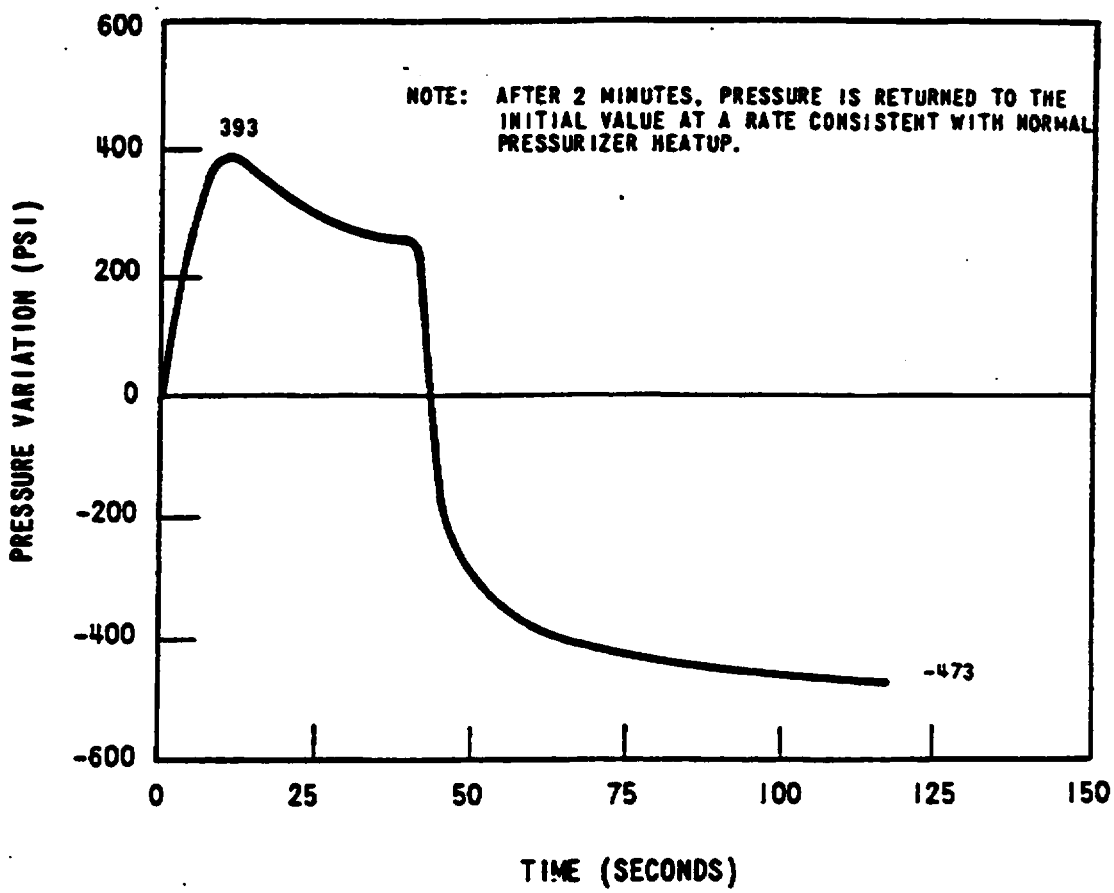
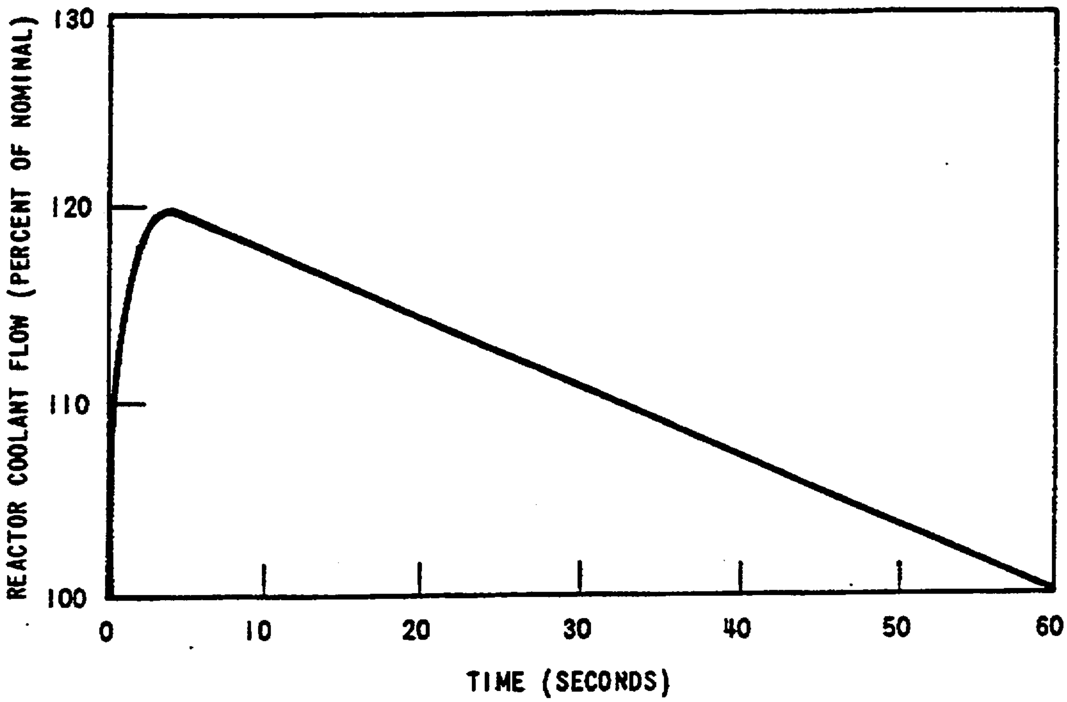


Figure 2-23 Loss of Load — Reactor Coolant Pressure Versus Time



NOTE: AFTER 60 SECONDS THE FLOW CONTINUES AT 100"
OF ITS NOMINAL VALUE.

Figure 2-24 Reactor Trip with No Inadvertent Cooldown — Reactor Coolant Flow Versus Time (Emergency Overspeed)

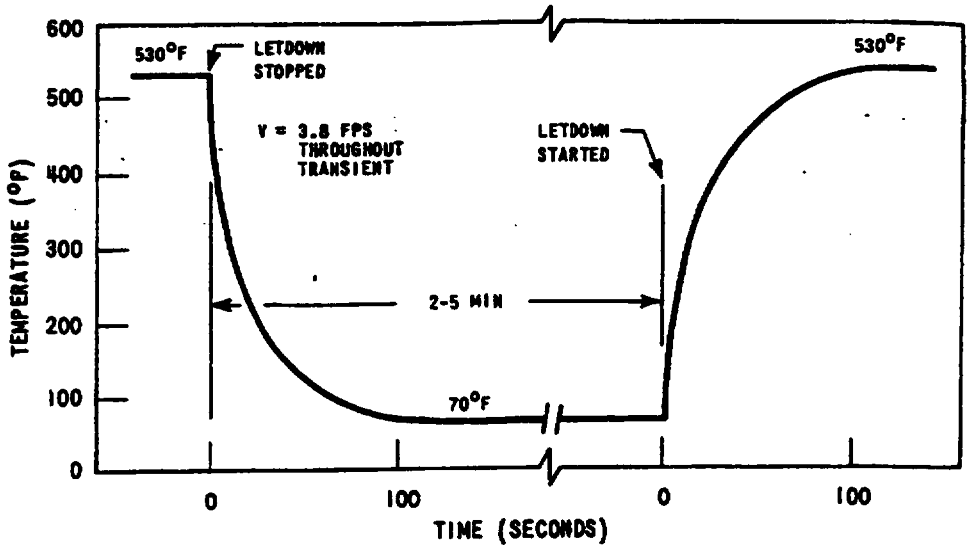


Figure 2-25a Charging Nozzle — Letdown Flow Shutoff with Prompt Return to Service (250 gpm system, 20 occurrences)

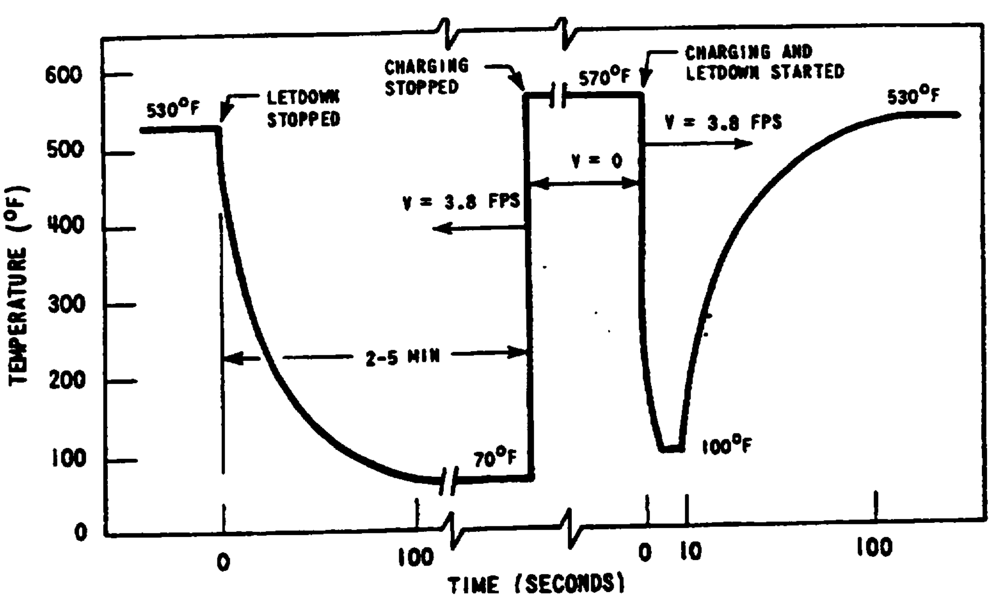


Figure 2-25b Charging Nozzle — Letdown Flow Shutoff with Delayed Return to Service (250 gpm system, 20 occurrences)

2.4.8 Leak-Before-Break Design

Leak-before-break (LBB) analysis is an additional TLAA that is applicable to piping systems to limit the severity of postulated accidents. This analysis is performed to show that any leaks that develop in the piping can be detected by plant monitoring systems before a crack causing the leak would grow to unstable proportions, leading to a potential double-ended guillotine break. The geometry of the pipe and materials are used in these analyses, and the aging effects on materials and geometry (e.g., erosion/corrosion or wear) must be considered.

2.5 TIME-LIMITED AGING ANALYSES

Time-limited aging analyses are those licensee calculations that:

- Involve the effects of aging
- Involve time-limited assumptions defined by the current operating term, for example, 40 years
- Involve systems, structures, and components within the scope of license renewal
- Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, or component to perform intended functions
- Were determined to be relevant by the licensee in making a safety determination
- Are contained or incorporated by reference in the current licensing basis

Based on the description of the engineering and design of Class I piping and associated components, the time-limited aging analyses satisfying all six criteria from the license renewal rule listed above are fatigue and leak before break. (See Table 2-11.)

2.6 AGING EFFECTS

Aging degradation refers to the time-dependent degradation of a material or component, which may result in a decrease in the ability of the material or component to perform its intended function. The mechanisms by which age-related degradation can occur may be driven by physical, mechanical, or chemical processes, i.e., by interaction of the material or component with its physical, mechanical, or chemical environment. The specific mechanisms selected for assessment are those that experience has shown to be significant or potentially significant to the performance of nuclear power plant components—pressurizers, steam generators, reactors—as well as those mechanisms recognized as being life-limiting during initial design of the Class 1 piping. The aging effects considered potentially significant for the Class 1 piping and associated components within the scope of this report are:

- Fatigue-related cracking for fatigue-sensitive items
- Cracking and material degradation due to corrosion/stress corrosion cracking

- Cracking due to irradiation embrittlement
- Thermal aging-related cracking of austenitic stainless steel static castings
- Material wastage due to erosion and erosion/corrosion
- Material loss caused by wear of RCP and Class 1 valve closure elements
- Loss of bolt preload due to creep or stress relaxation of bolted RCP and Class 1 valve closures

These aging effects can result in degradation of structural integrity.

Following a description of these mechanisms and the evaluation of their effects in Section 3.2, an assessment of the applicability and management of those aging effects to individual Class 1 piping and components is summarized in Section 3.4. Time-dependent analyses are discussed in Section 3.3. These discussions apply to both Class 1 piping and associated pressure boundary components.

TABLE 2-11
TIME-LIMITED AGING ANALYSES REQUIREMENTS FOR AGING MECHANISMS

Requirements	Fatigue	Corrosion	Irradiation Embrittlement	Leak-Before-Break and Fracture Mechanics Evaluation of RCP Casings (Thermal Aging)	Erosion	Wear	Creep and Stress Relaxation
Involve the effects of aging	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Involve time-limited assumptions defined by the current operating term (e.g. 40 years)	Yes	No	No	Yes	No	No	No
Involve SSCs within scope of license renewal	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Involve conclusions or provide the basis for conclusions related to the capability of the SSC to perform its intended functions	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Were determined to be relevant by the licensee in making a safety determination	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Are contained or incorporated by reference in the CLB	Yes	No	No	Yes	No	No	No

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3.0 TIME-LIMITED AGING ANALYSES AND AGING EFFECT EVALUATIONS

In this section, mechanisms are described to determine aging effects, and all identified aging effects are evaluated to identify potential degradation of the intended function of Class 1 piping and associated pressure boundary components. This section also evaluates time-limited aging analyses (TLAAs). All aging effects and TLAAs that require management during an extended period of operation are identified.

3.1 ASSESSMENT OF INDUSTRY ISSUES AND MAINTENANCE HISTORY

The Westinghouse RCS piping systems have experienced few operational and maintenance problems during more than 25 years of service. Historically, maintenance issues have been limited, and most issues are not design- or pipe-related. In more recent years, some concerns relating to aging management have been raised. In 1990, the Nuclear Energy Institute (NEI), then known as Nuclear Management and Resources Council (NUMARC), issued to the NRC for comment, Industry Report (IR) 90-07 on the RCS [Ref. 12]. This document addressed low-and high-cycle fatigue, corrosion, SCC, radiation effects, thermal aging, creep and stress relaxation, erosion, and wear age-related degradation mechanisms (see EPRI TR-104305 Rev. A [Ref. 13] for more details). Two major concerns on the piping are fatigue and thermal embrittlement of statically cast austenitic stainless steel (CASS). Class 1 valve and RCP pressure boundary age-related issues include those for piping plus stress relaxation of bolted closures, boric acid wastage on external surfaces, and wear of closure elements.

The following subsections apply to the Class 1 piping and associated pressure boundary components.

3.1.1 Reactor Coolant Piping

Industry experience with the RCS (see Table 3-1) has validated Class 1 piping design and integrity. The number of pipe failures has been limited and isolated to connections to the main coolant piping. This is reflected in several Information Notices (INs) and Licensee Event Reports (LERs). Several NUREG reports also address the probability of pipe failure in the reactor coolant loops (RCLs) of Westinghouse PWR plants. In addition, IN-92-36 Supplement 1 (see Table 3-1), addressing interfacing system loss-of-coolant accident (LOCA), identifies the use of IPE and PRA techniques for resolution of Generic Safety Issue (GSI) 105. The main areas of regulatory concern and scientific investigation in piping are failures from fatigue and thermal stress.

3.1.2 Fatigue

Since late 1991, there has been much attention given to the issue of fatigue qualification for nuclear power plants. Questions associated with this issue were originally raised in regard to plant license renewal. At that time, the NRC was developing a Branch Technical Position (BTP) [Ref. 14] that would include fatigue evaluation procedures. To account for NRC concerns regarding environmental effects on the fatigue life of PWRs and boiling water reactors (BWRs), the BTP procedures imposed significant penalties on the ASME Code fatigue calculations. The principal bases for these penalties were studies performed by Argonne National Labs (ANL) and documented in NUREG/CR-5999 [Ref. 15].

TABLE 3-1
INDUSTRY ISSUES AND MAINTENANCE HISTORY

Document ID	Date	Title	Author/Contributor	Comments
Regulatory				
IB 88-08	6/17/88	Thermal Stresses in Piping Connected to Reactor Coolant System	NRC	Identifies thermal stratification potentials for unisolatable portions of the RCS and advises utilities to review their designs for potential impact.
IB 88-11	12/20/88	Pressurizer Surge Line Thermal Stratification	NRC	Requires plants with operating licenses to perform a VT-3 inspection on pipe, supports, whip restraints, and anchor bolts to determine gross discernable distress or structural damage; and to evaluate the line to ensure that it meets the ASME Section III requirements, in particular high cycle fatigue and thermal fatigue.
GL 84-13	5/3/84	Thermal Stresses in Piping Connected to Reactor Coolant System	NRC	Provides revision to NRC Standard Technical Specification for snubbers.
GL 90-09	12/11/90	Alternative Requirements for Thermal Stresses in Piping Connected to Reactor Coolant System	NRC	Provides relief for visual inspection intervals based on snubber failure population, and states that functional testing provides a 95% confidence level that 90% to 100% of snubbers operate within specified acceptance limits.
GL 88-05	3/22/88	Boric Acid Corrosion of Carbon Steel Pressure Boundary Components in PWR Plants	NRC	The NRC requested licensees to procedurally control the corrosive effects of RCS leakage that could potentially affect the integrity of the reactor coolant pressure boundary.
IN 93-90	12/1/93	Unisolatable Reactor Coolant System Leak Following Repeated Application of Leak Sealant	NRC	Reactor coolant system integrity degradation caused by online leak sealing process using Furmanite.

TABLE 3-1 (Continued)
INDUSTRY ISSUES AND MAINTENANCE HISTORY

Document ID	Date	Title	Author/Contributor	Comments
IN 93-84	10/20/93	Determination of Westinghouse Reactor Coolant Pump Seal Failure	NRC	Monitoring of no. 2 seal leakage might not indicate the operability of no. 1 seal. Westinghouse issued Technical Bulletin NSD-TB-93-01-R0 on March 30, 1993, to affected PWR.
IN 94-55	8/4/94	Problems with Copes-Vulcan Pressurizer Power-Operated Relief Valves	NRC	Problems involving the cracking of plug material, severe wear plugs and cages, and stem misalignment and galling of C-V PORVs
IN 93-66	8/16/93	Switchover to Hot Leg Injection Following a LOCA in PWR	NRC	Identifies a single failure vulnerability for the switchover to hot leg injection for a medium and large hot-leg LOCA for Westinghouse PWRs.
IN 93-61	8/9/93	Excessive Reactor Coolant Leakage Following a Seal Failure in a Reactor Coolant Pump or Reactor Recirculation Pump	NRC	Reactor coolant pump seal improve performance monitoring and maintenance to replace obsolete parts
IN 93-02	1/4/93	Malfunction of a Pressurizer Code Safety Valve	NRC	Premature lift may have been caused by testing methods used to test the valve before installation, thus causing incorrect setpoint.
IN 92-86	12/24/92	Unexpected Restriction to Thermal Growth of Reactor Coolant Piping	NRC	Friction from sliding support prevented smooth movement of cross-over piping.
IN 92-36 Sup. 1	2/22/94	Intersystem LOCA Outside Containment	NRC	Identifies the use of IPE and PRA techniques to aid in resolving GSI 105.
IN 92-15	2/24/92	Failure of Primary System Compression Fillings	NRC	Maintenance/installation of 3/4-inch connection
IN 91-74	11/25/91	Changes in Pressurizer Safety Valve Setpoints Before Installation	NRC	Maintain closer control of maintenance, testing and operations performed on the valve after installation
IN 91-87	12/27/91	Hydrogen Embrittlement of	NRC	Concerns about use of Tinel in high-hydrogen,

TABLE 3-1 (Continued)
INDUSTRY ISSUES AND MAINTENANCE HISTORY

Document ID	Date	Title	Author/Contributor	Comments
		Raychem Craft Couplings		high-temperature environment
IN 88-30	5/25/88	Target Rock Two-Stage SRV Setpoint Drift Update	NRC	Continual problem on setpoint drift
IN 88-80	10/07/88	Unexpected Pipe Movement Attributed to Thermal Stratification	NRC	Provided information regarding unexpected movement of surge line
IN 82-30	4/21/82	Loss of Thermal Sleeves in RCS Piping at Certain Westinghouse PWR Power Plants	NRC	Design error
IN 82-14	6/11/82	TMI 1 Steam Generator/Reactor Coolant System Chemistry/Corrosion Problem	NRC	Personnel error
IN 86-108	4/16/87	Degradation of RCS Pressure Boundary Resulting From Boric Acid Corrosion	NRC	Poor maintenance
IN 87-046	9/24/87	Undetected Loss of Reactor Coolant	NRC	Procedural error
Reg Guide 1.45	5/31/73	Reactor Coolant Pressure Boundary Leak Detection System	NRC	–
INPO SOER 25-87	9/8/87	Surge Line Thermal Cycling Observed During Reactor Coolant System Pressurization Heatup and Cooldown	INPO	–
LER 92-002	3/25/92	Safety Relief Valve Actuation	Duke Power Co .– Oconee 2	Setpoint error

TABLE 3-1 (Continued)
INDUSTRY ISSUES AND MAINTENANCE HISTORY

Document ID	Date	Title	Author/Contributor	Comments
LER 91-016	1/8/92	PORV Stem to Wedge Assembly Failure	Duke Power Co. – Oconee 2	–
LER 91-026	11/29/91	Pressurizer Safety Valve Failure	Texas Utilities – South Texas Project	Setpoint out of tolerance
LER 88-044	2/15/89	Leakage from Safety Injection Check Valves	Duke Power Co. – Oconee 2	Caused by wear
LER 87-015	12/18/87	Backup Nitrogen Supply to PORV Inoperable	Consolidated Edison – Indian Point Unit 2	Caused by check valve failure
LER 84-012	6/26/84	Valve Disc to Stem Separation	PSE&G	Fabrication weld error
LER 84-010	5/1/84	RTD Bypass Valves Disc to Stem Separation	PSE&G – Salem 1	Excessive force from backseating on joints
LER 93-002-00	2/1/93	Identified Non-Conservatism in Heatup/Cooldown & Cold Overpressure Protection PORV Setpoint Curves	Commonwealth Edison Co. – Braidwood Station, Unit 1	Procedural problem
LER 84-006-00	8/6/84	Unidentified Reactor Coolant Leakage	Baltimore Gas & Electric Co. – Calvert Cliffs Nuclear Power Plant, Unit 2	Leak caused by inservice fatigue induced cracked weld.
LER 94-001-00	4/27/94	Unisolatable RCS Leak	Pacific Gas & Electric Co. – Diablo Canyon Nuclear Power Plant, Unit 2	RCS leak caused by inadequate weld penetration.
LER 89-002-00	3/13/89	Thermowell Leakage During Low Power Test	Carolina Power & Light Co. – H.B. Robinson Plant, Unit 2	Thermowell leak caused by fatigue failure. Reduced thermowell insertion length incorporated into replacement thermowells.

TABLE 3-1 (Continued)
INDUSTRY ISSUES AND MAINTENANCE HISTORY

Document ID	Date	Title	Author/Contributor	Comments
LER 94-003-00	2/17/94	RX Coolant System Sample Line Declared Inoperable & Isolated	Wisconsin Electric Power – Point Beach	–
LER 93-009-00	10/22/93	MSSV and Pressurizer Safety Valves "as Found" Relief Setting Out of Tolerance	Arizona Public Service Co. – Palo Verdi Unit 1	Setpoint drift.
LER 94-015-00	6/29/94	Determined that Postulated Pressure Transient Could Exceed Design Pressure Limit	Pacific Gas & Electric Co. – Diablo Canyon Nuclear Power Plant, Unit 1	Insufficient design basis information.
LER 87-010-01	4/11/88	Reactor Shutdown Due to Instrument Line Leakage	Georgia Power Co. – Joseph M. Farley Nuclear Power Plant, Unit 2	Defect in welded joint caused by high cycle fatigue.
LER 87-010-00	1/6/88	Reactor Shutdown Due to Instrument Line Leakage	Georgia Power Co. – Joseph M. Farley Nuclear Power Plant, Unit 2	Cracked joint due to fatigue in heat affected zone.
LER 89-012-01	3/4/91	Leaking Weld Attaching Vent to Loop Bypass Line	Yankee Atomic Power Co. – Yankee Rowe Nuclear Power Station	Weld leak caused by high cycle fatigue.
LER 90-008-00	6/8/90	Primary Coolant System Leakage at Drain Valve	Texas Utilities – South Texas Project, Unit 2	Caused by high cycle fatigue failure of weld.
LER 88-011-00	9/8/88	Declaration of Unusual Event Due to Sensing Line Failure	Arkansas Power & Light Co., – Arkansas Nuclear One, Unit 2	Sensing line failure caused by low stress, high cycle & weld fatigue failure.
LER 85-013-00	11/5/85	Unidentified RCS Leakage	Baltimore Gas & Electric Co. – Calvert Cliffs Nuclear Power Plant, Unit 1	Cracked weld between reactor coolant pump shaft seal and control bleedoff line.

TABLE 3-1 (Continued)
INDUSTRY ISSUES AND MAINTENANCE HISTORY

Document ID	Date	Title	Author/Contributor	Comments
LER 92-004-00	4/16/92	Out of Tolerance Main Steam Safety Valve and Pressurizer Safety Valve	Arizona Public Service Co. – Palo Verdi Unit 1	Caused by setpoint drift.
NPRDS	N/A	N/A	N/A	Failure resulted from the aging of the flow element flange gaskets.
NPRDS	N/A	N/A	N/A	Cracked weld on drain connection caused by poor design for not providing adequate support for the drain assembly.
NPRDS	N/A	N/A	N/A	Flow-induced vibrations caused the tack welds to fail.
NPRDS	N/A	N/A	N/A	Defective weld caused by excessive vibration.
NUREG CR-3982 R	11/30/84	Case Study of the Propagation of a Small Flaw under PWR Loading Conditions and Comparison with the ASME Code Design Life	N/A	Comparison of ASME Code, Sections III and XI
NUREG CR-3660	7/31/85	Probability of Pipe Failure in the Reactor Coolant Loops of Westinghouse PWR Plants	N/A	Volume 4: Pipe Failure induced by Crack Growth in West Coast Plants
NUREG CR-5195 R	12/31/88	Fatigue Strength of ASME SA 106-B Welded Steel Pipes in 288 Degree C Air Environments	N/A	–
NUREG CR-5490	10/90	Regulatory Instrument Review: Management of Aging of LWR Major Safety Related Components	PNL	–
NUREG CR-4999		Estimated Risk Reduction from improved PORV reliability in PWRs	N/A	–

TABLE 3-1 (Continued)
INDUSTRY ISSUES AND MAINTENANCE HISTORY

Document ID	Date	Title	Author/Contributor	Comments
NUREG CR-4234 V2	9/89	Aging and Service Wear of Electric Motor Operated Valves in Engineered Safety Feature Systems of Nuclear Power Plants	ORNL	-
NUREG CR-3660 R	3/13/85	Probability of Pipe Failure in the Reactor Coolant Loops of Westinghouse PWR Plants	N/A	
NUREG CR-3660 R	9/20/84	Probability of Pipe Failure in the Reactor Coolant Loops of Westinghouse PWR Plants	N/A	
NUREG CR-3660 R	5/85	Probability of Pipe Failure in the Reactor Coolant Loops of Westinghouse PWR Plants	N/A	Volume 1: Summary Report
NUREG CR-3483 R	1/31/84	A Study of the Regulatory Position on Postulated Pipe Rupture Location Criteria	N/A	-
REPORTS				
Topical Report	10/27/88	Technical Justification for Eliminating Large Primary Loop Pipe Rupture as Structural Design Basis for Beaver Valley Unit 2 after Reduction of Snubbers	N/A	Topical report issued for South Texas Units 1&2 pressurizer surge line and RHR line stratification
AEOD/T93-01	6/30/93	Primary System Integrity, Pressurized Water Reactors	NRC	-
CE Tech Report 85-01	2/13/85	Combustion Engineering Information bulletin Concerning Primary System Corrosion	Combustion Engineering	-

TABLE 3-1 (Continued)
INDUSTRY ISSUES AND MAINTENANCE HISTORY

Document ID	Date	Title	Author/Contributor	Comments
EPRI TR-102901		Fatigue Comparison of Piping Designed to ANSI B31.1 and ASME Section III, Class 1 rules	Structural Integrity Associates	–
Westinghouse Tech Report 85-039	10/8/87	Westinghouse Letter on Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion	Westinghouse	–
IN 93-84	10/20/93	Determination of Westinghouse Reactor Coolant Pump Seal Failure	NRC	Monitoring of No. 2 seal leakage might not indicate the operability of No. 1 seal. Westinghouse issued Technical Bulletin NSD-TB-93-01-R0 on March 30, 1993, to affected PWR.
IN 94-55	8/4/94	Problems with Copes-Vulcan Pressurizer Power Operated Relief Valves	NRC	Problems involving the cracking of plug material, severe wear plugs and cages, and stem misalignment and galling of C-V PORVs.
IN 93-66	8/16/93	Switchover to Hot-Leg Injection Following a LOCA in PWR	NRC	Identifies a single failure vulnerability for the switchover to hot leg injection for a medium and large hot-leg LOCA for Westinghouse PWRs.
IN 93-61	8/9/93	Excessive Reactor Coolant Leakage Following a Seal Failure in a Reactor Coolant Pump or Reactor Recirculation Pump	NRC	Reactor coolant pump seal improve performance monitoring and maintenance to replace obsolete parts.
IN 93-02	1/4/93	Malfunction of a Pressurizer Code Safety Valve	NRC	Premature lift may have been caused by testing methods used to test the valve before installation, thus causing incorrect setpoint.
IN 92-86	12/24/92	Unexpected Restriction to Thermal Growth of Reactor Coolant Piping	NRC	Friction from sliding support prevented smooth movement of cross-over piping.

TABLE 3-1 (Continued)
INDUSTRY ISSUES AND MAINTENANCE HISTORY

Document ID	Date	Title	Author/Contributor	Comments
IN 92-36 Sup. 1	2/22/94	Intersystem LOCA outside Containment	NRC	Identifies the use of IPE and PRA techniques to aid in resolving GSI 105.
LER 89-012-01	3/4/91	Leaking Weld Attaching Vent to Loop Bypass Line	Yankee Atomic Power Co. – Yankee Rowe Nuclear Power Station	Weld leak caused by high cycle fatigue.
LER 90-008-00	6/8/90	Primary Coolant System Leakage at Drain Valve	Texas Utilities – South Texas Project, Unit 2	Caused by high cycle fatigue failure of weld.
LER 88-011-00	9/8/88	Declaration of Unusual Event Due to Sensing Line Failure	Arkansas Power & Light CO. – Arkansas Nuclear One, Unit 2	Sensing line failure caused by low stress, high cycle & weld fatigue failure.
LER 85-013-00	11/5/85	Unidentified RCS Leakage	Baltimore Gas & Electric Co. – Calvert Cliffs Nuclear Power Plant, Unit 1	Cracked weld between reactor coolant pump shaft seal and control bleedoff line
LER 92-004-00	4/16/92	Out of Tolerance Main Steam Safety Valve and Pressurizer Safety Valve	Arizona Public Service Co. – Palo Verdi Unit 1	Caused by setpoint drift.
NPRDS	N/A	N/A	N/A	Failure resulted from the aging of the flow element flange gaskets.
NPRDS	N/A	N/A	N/A	Cracked weld on drain connection caused by poor design for not providing adequate support for the drain assembly.
NPRDS	N/A	N/A	N/A	Flow-induced vibrations caused the tack welds to fail.
NPRDS	N/A	N/A	N/A	Defective weld caused by excessive vibration.
NUREG CR-3660 R	5/85	Probability of Pipe Failure in the Reactor Coolant Loops of Westinghouse PWR Plants	N/A	Volume 1: Summary Report

TABLE 3-1 (Continued)
INDUSTRY ISSUES AND MAINTENANCE HISTORY

Document ID	Date	Title	Author/Contributor	Comments
NUREG CR-3483 R	1/31/84	A Study of the Regulatory Position on Postulated Pipe Rupture Location Criteria	N/A	–

In July, 1993, the NRC expanded their concern to the fatigue qualification of operating plants. The draft BTP had been withdrawn and was replaced by a generic technical Fatigue Action Plan (FAP) for operating plants [Ref. 16]. The FAP addressed three issues:

1. Do reactor coolant pressure boundary components of older vintage nuclear power plants that were designed to codes that did not require the explicit fatigue analysis required by the current ASME Code have adequate fatigue resistance?

2. Current test data show that the ASME design fatigue curves may not be conservative for nuclear power plant primary system environments. Is the decrease in fatigue life for components exposed to these environments significant enough to require licensees to use new fatigue curves that consider the environmental effects?

3. What are the appropriate actions to be taken when the calculated fatigue allowable limit has been exceeded (cumulative usage factor > 1)?

Results and conclusions of the NRC Fatigue Action Plan were documented in SECY-95-245 [Ref. 17]. To address issues 1 and 2, the NRC performed evaluations of selected components at seven operating plants to assess the degrees of conservatism in design fatigue evaluations and the impact of the more restrictive "interim fatigue curves" recommended in NUREG CR-5999 [Ref. 18]. Based on the component sample evaluations, the NRC concluded that no immediate licensee action was necessary since the ASME fatigue limit was not exceeded for most components for the current design life. It was also concluded that, with more detailed analyses and/or measured plant transient data, most of the remaining components could be shown to be within ASME limits for the current design life. Based on the U.S. NRC office of Nuclear Regulatory risk study, a backfit of environmental fatigue data for operating parts was not justifiable.

For operation beyond the current design life, the NRC concluded that FAP issues should be evaluated further, focusing mainly on components in the reactor coolant pressure boundary with high fatigue usage.

> The staff will consider, as part of the resolution of Generic Safety Issue GSI-166, 'Adequacy of Fatigue Life of Metal Components' [Ref. 19], the need to evaluate a sample of components with high fatigue usage, using the latest available environmental fatigue data, to ensure that RCPB components will continue to perform their intended functions and maintain a high level of reliability during the extended period of operation for license renewal. If GSI-166 has not been resolved before the issuance of a renewal license, the applicant would have to submit ... its technical rationale for concluding that the effects of fatigue are adequately managed for the extended period or until the resolution of GSI-166 becomes available. [Ref. 17]

In addressing issue 3 in SECY-95-245, the NRC recommended guidance from Generic Letter GL 91-18 [Ref. 20], as describing actions that a licensee can take to resolve the nonconforming condition. It also refers to a nonmandatory appendix being developed by ASME, Section XI Task Group on Operating Plant Fatigue Assessment that specifies actions to be taken if the CUF exceeds unity. When the appendix is published, the NRC will determine the acceptability of its approach.

In parallel with the NRC activities, the Pressure Vessel Research Council (PVRC), at the request of the ASME Board on Nuclear Codes and Standards (BNCS), is also examining the effects of RCS environments on existing ASME, Section III and Section XI fatigue curves. Results have indicated that the significance of the PWR and BWR environments is dependent on the combination of several variables: dissolved oxygen, temperature, material sulfur content, strain amplitude, coolant flow velocity, and loading strain rates. This work, which is still ongoing, is being addressed by a Steering Committee on Cyclic Life and Environmental Effects (CLEE), under which three working groups exist: Working Group on S-N Data Analysis, Working Group on da/dN Analysis and Working Group on Evaluation Methods.

Other industry studies have also continued on the fatigue issue. Nuclear Energy Institute (NEI) has worked with industry groups and the NRC through the NEI Fatigue Task Force. The task force documented its conclusions on the NRC fatigue concerns in the "Fatigue White Paper" [Ref. 21]. The task force reached conclusions similar to those of the NRC.

EPRI sponsored a project to evaluate piping systems designed to ANSI B31.1, by comparing the results of ASME Code, Section III, NB-3600 Class 1 detailed fatigue analyses to more simplified fatigue strength reduction factor analyses for these same piping systems [Ref. 11]. These results support the conclusion that both the ANSI B31.1 and ASME, Section III Class 1 fatigue design rules provide comparable piping component construction [Ref. 21].

With respect to license renewal for Westinghouse PWR components, including Class 1 piping and associated components, the following observations are considered to be significant:

- The conclusions of the FAP do not provide closure for the fatigue issues in the case of license renewal especially for environmental effects in fatigue.

- The resolution of GSI 166, "Adequacy of Fatigue Life of Metal Components" [Ref. 19], and GI 78, "Monitoring of Design Basis Transient Fatigue Limits for Reactor Coolant System" [Ref. 22], by the NRC should provide regulatory information regarding the need for additional component evaluations using environmental fatigue data. (Generic Issue 78 has been resolved with reference to the fatigue action plan for the transient monitoring concern).

- A request for license renewal before the resolution of GSI 166 will need to include technical rationale for concluding that the effects of fatigue are adequately managed for the extended period or until the resolution of GSI 166 becomes available.

Therefore, since fatigue is identified as a potentially significant degradation mechanism for the Class 1 piping and associated components, industry activities intended to resolve the fatigue issues identified in the U.S. NRC completion of the fatigue action plan should be evaluated relative to the fatigue management plan. Specific industry activities to evaluate include:

- Guidance from the NEI License Renewal Working Group and related NEI technical issue tracking efforts

- Recent developments on inservice inspection and flaw evaluation from ASME Code, Section XI bodies

- Recommendations to the ASME Code committees from the PVRC Steering Committee on Cyclic Life and Environmental Effects

3.1.3 Thermal Stress

In May of 1984, the NRC issued a Generic Letter 84-13 (see Table 3-1) addressing thermal stresses in piping connected to RCSs and the requirements for snubbers attached to the piping and components. The issuance of Information Notice (IN) 92-86 (see Table 3-1) in December 1992 focused the attention of thermal stress on the cross-over leg supports for the primary coolant system where the friction of the support was overcome in a step change, causing a noise event inside containment and a significant drop in pipe stress caused by thermal growth. Some of these restrictions may be removed following the application of leak-before-break criteria.

3.1.4 Safety and Relief Valves

The issues identified in this section are related to the active function and will not be subject to an aging management review. Another item to be considered is the ability of the safety relief and power-operated relief valves to perform their function. The main experience for these valves is the inability to seat after opening and drifting off the lift setpoint. IN 91-75 was issued following the investigation of the cause for excessive safety valve setpoint changes. This notice recognized the fact that historically, as identified in several LERs, over 40 percent of the as-found pressurizer safety valves have failed the setpoint test. IN 91-74 addressed Dresser valves; however, other valves performing a similar service are susceptible to the same problems. Industry experience with setpoint drift is identified in Table 3-1. In one instance, as identified in IN 93-02, dated January 4, 1993, the testing of the valve at a testing laboratory using a different environmental arrangement/conditions caused the setpoint to be incorrectly set.

In another instance, it has been identified that a problem exists with the cracking of plug material and severe wear of plugs and cages coupled with the misalignment and galling of stems in PORVs. The failures were attributed to stresses caused by differential thermal expansion. The solution to the problem in the case cited was to change plug material to type 316 stainless steel with stellite overlay and 17-4 PH stainless steel cages.

3.1.5 Check Valves

Several failures have been reported in check valves. These valves are the first pressure barrier between the RCS and the supporting system. Most of these failures have occurred due to the separation of stems from discs; in most cases, the integrity of the primary piping was not jeopardized. Loss of coolant from the system was minimal. The failure of the check valves to prevent loss of reactor coolant from the system provides the potential for intersystem LOCA (ISLOCA). This is addressed in References 23 through 26.

3.1.6 Pump Seals

The failure of the RCP seals has been identified as a concern in several NRC documents. IN 93-84 was issued on October 20,1993. The failure of a no. 1 seal is not always easy to identify by monitoring the flow from the no. 2 seal. Inadequacies in the instrumentation used to monitor leakage may not identify seal failure. Westinghouse Technical Bulletin NSD-TB-93-01 R0 [Ref. 27] was issued on March 30, 1993 to address this issue.

Several LERs have been issued concerning the fatigue-induced failures in several locations on instrument line and small bore piping attached to the RCS pipe. The failures tend to occur at the weld joint areas of small piping. Failure has occurred at several different plants. The pump seals are considered part of the overall active function of the pump. This issue is not a licensing renewal concern because pump seals are part of a preventive replacement program.

3.1.7 Primary System Material Interactions

The intrusion of material that may degrade the primary piping was identified in IN 93-90 issued December 1, 1993 (see Table 3-1). In this instance, a foreign substance, Furmanite, was introduced into the primary system through an online leak sealing operation. The effects of the introduction of Furmanite into the primary system environments and the long-term degrading effects on the piping system have not been assessed.

There has been a concern about hydrogen embrittlement of Raychem Craft couplings when they are used in a high hydrogen environment. There has been a limited amount of failure experience with trunnions and lugs on Class 1 systems, and none of these have created any failure mechanisms for the RCS.

3.2 AGING MANAGEMENT REVIEW

In this section, mechanisms are described to determine aging effects, and all identified effects are evaluated to identify potential degradation of the Class 1 piping and associated components intended function. Section 3.3 evaluates the time-limited aging analyses. All effects and time-limited aging analyses that require management during an extended period of operation are identified.

An aging effect is defined to be significant for a component if, when allowed to continue without an effective program, the capability of the component to perform its intended function throughout the license renewal term would be compromised. The potential significance of an aging effect was determined by examining the component design features (Section 2.4), the component design bases (Section 2.4), its operation and maintenance histories (Section 3.1), and its susceptibility to the aging effect being considered. If it can be shown that the component is either not susceptible or is susceptible to such a small degree that the component's safety function is maintained throughout the license renewal term, then the component/aging effect combination is not significant.

Effects of potentially significant age-related degradation mechanisms are examined in terms of the capability of effective programs for maintenance, inservice inspection, surveillance, testing, and analytical assessment to manage the effects. Combinations of effects and components for which generic program elements effectively manage the aging effects are provided in Section 4.0 of this report.

License renewal applicants intending to reference these generic conclusions are responsible for a review of plant-specific features, including appropriate current licensing basis (CLB) documents and information, to determine this report's limitations. This review should compare the design basis for particular components with the representative design bases given in Section 2.4. The component operation and maintenance histories should also be compared to the generic performance parameters described in Section 3.1. Finally, specific assumptions and criteria used in this section should be examined to ensure that they, or justified equivalents, apply to the component under consideration.

3.2.1 Fatigue

Mechanism Description

Fatigue is defined as the structural deterioration that can occur as a result of the periodic application of load or stress by mechanical, thermal, or combined effects. It has been recognized for many years that a metal subjected to a repetitive or fluctuating stress will fail at a stress much less than that required to cause fracture on a single application of load. The important factor in fatigue failure is stress repetition. The specific effects of fatigue are cracks in the material that may or may not be detected before mechanical failure. After repeated cyclic loading of sufficient magnitude, microstructural damage can accumulate, leading to macroscopic crack initiation at the most affected locations. Subsequent mechanical or thermal cyclic loading can lead to growth of the initiated crack.

Aging Effect Evaluations

Evaluations of fatigue analyses, which are considered to be time-limited aging analyses, are provided in Section 3.3.

<u>RCP Casings</u>

Typically, for ASME RCP casing designs, the fatigue analysis is not required because the limits on peak stress intensities as governed by fatigue are satisfied for the RCP casing by meeting all the conditions specified in NB-3222.4(d)(1) through (6).

Since the peak stress intensities are not a function of cycles, the fatigue waiver evaluations that were performed for the current licensing basis are valid for the license renewal term. Alternatively, in some cases, the procedures in NB-3200 were used to perform the detailed fatigue analysis of the RCP casing. The detailed fatigue evaluations were generally conservative. If the conservatisms were removed, the detailed fatigue evaluations should compare to the fatigue waiver evaluations. Similar to the B31.1 valves, no standard analysis method was available for older reactor coolant pump (RCP) designs; therefore, the requirements for the original RCP design basis could be satisfied without performing a fatigue analysis. In general, the Class 1 RCP casings for the B31.1 plant designs were designed to the intent of the ASME code for fatigue using a fatigue waiver. Therefore, the RCP casings are not considered to be fatigue-sensitive items.

Aging Effect Management

The potentially significant effects due to fatigue may occur at several fatigue-sensitive locations (see Table 4-4). Aging management options for fatigue will depend on the final NRC position for license renewal. Several options are described in Subsection 4.2.1 to manage the effects from fatigue. The first option is to demonstrate that fatigue effects anticipated for the license renewal term are bounded by the fatigue effects anticipated for the original service period as justified by the current licensing basis. The second and third options demonstrate that fatigue flaws will be detected before they can propagate to failure. The second option shows that an adequate inservice inspection program exists to detect and size flaws between inspection intervals. The third option, which is similar to the second, includes an analysis of the flaw in addition to inspections. Two types of analyses are considered. An ASME, Section XI [Ref. 28] type of flaw tolerance analysis will show that a postulated or actual flaw will not propagate to failure. Alternatively, a leak-before-break analyses could be used to show that a postulated through-wall flaw could occur, and the plant could safely shut down. Another approach, included in the third option, is to demonstrate that fatigue effects will not occur based on acceptable fatigue analyses in accordance with the reconstituted license renewal transients. The fourth option is to repair or replace the component.

3.2.2 Corrosion

Mechanism Description

Corrosion is the degradation of a material by chemical or electrochemical reaction with its environment. There are many forms or effects of corrosion depending on the material and environment. The extent of corrosion may be general or localized. General corrosion refers to a uniform attack over surfaces of the material and results in thinning of the material, usually at a slow rate. General or uniform corrosion can be managed by allowing sufficient excess material thickness to accommodate the amount of material expected to be lost during the service lifetime of the piping or component. Localized corrosion is usually more difficult to manage. The forms of localized corrosion include pitting, crevice corrosion, and stress corrosion cracking (SCC). Pitting corrosion is a microscopically localized form of corrosion associated with a specific chemical species in the environment or local conditions of the surface of the material. Crevice corrosion results from local environment conditions in the restricted region of a crevice being different and more aggressive than the bulk environment.

SCC is a localized nonductile failure caused by a combination of stress, susceptible material, and an aggressive environment. Microscopically, the SCC failure mode can be either intergranular (IG) or transgranular (TG). IGSCC is generally associated with a sensitized material. Sensitization of unstabilized austenitic stainless steel is characterized by a depletion of chromium at the grain boundaries with an accompanying precipitation of a network of chromium carbides. Because the depletion of chromium at or near grain boundaries is caused by the formation of carbides, the carbon content of the austenitic stainless steel is critical as to the susceptibility of the material to sensitization. If because of carbon content a given grade of austenitic stainless steel is considered susceptible to sensitization, it will not become sensitized unless cooled slowly through the sensitization temperature range, 482°C (900°F) to 927°C

(1700°F), during heat treatment. Sensitized austenitic stainless steel is susceptible to IGSCC in an oxidizing environment.

TGSCC is caused by aggressive chemical species, e.g., caustics or chlorides, especially if coupled with oxygen and combined with stresses approaching the yield strength or greater.

Aging Effect Evaluation

To date, operational experience in Westinghouse pressurized water reactor (PWR) plants has shown that general corrosion and stress corrosion are not a particular concern for the primary loop materials installed. Austenitic stainless steel is not susceptible to general corrosion in the benign PWR primary coolant because it passivates to form protective layers that mitigate the potential for corrosion degradation. This also holds true for the remaining Class 1 piping items and is mainly attributed to tight control of the water chemistry and low flow velocities in the system.

Since austenitic steels resist corrosive attack in a PWR environment by quickly oxidizing to form a protective film, all internal surfaces of PWR reactor coolant system (RCS) components fabricated from austenitic stainless steel are not subject to significant corrosive degradation. This resistance extends to crevice regions, where an aggressive environment has the potential to cause localized corrosion, even for film-forming materials. Hydrogen plays an important role in the control of crevice corrosion by minimizing the adverse effects of oxygen. The hydrogen overpressure in a PWR RCS provides adequate protection against crevice corrosion for the internal surfaces of RCS components.

Therefore, corrosion is nonsignificant for the internal surfaces of these components. No further evaluation is required with respect to general corrosion for the internal surfaces of Class 1 piping and components fabricated from austenitic stainless steel.

As a result of the protection afforded by austenitic stainless steel in a PWR environment, corrosion wastage of external surfaces of RCS components caused by leakage of borated water is the only potential concern related to corrosion for PWR RCS components. Leakage of PWR primary coolant through bolted closures, and the subsequent evaporation and re-wetting cycles, can lead to the presence of a concentrated boric acid slurry on the external surfaces of adjacent RCS components. These alternate wetting and drying cycles produce a low pH concentration that, in combination with an air atmosphere, can cause high corrosion rates. The corrosion rate is greatest at temperatures between 200B350EF, but potentially significant corrosion rates are possible at higher temperatures. Evaporative cooling of exposed components, associated with the flashing of leaking coolant into steam, can increase the corrosion rate of component external surfaces that are normally at temperatures where boric acid corrosion rates would be much lower [Ref. 12].

The only component external surfaces that may be exposed to leaking primary coolant are those adjacent to bolted joints, such as pump casings adjacent to bolted RCP covers, and Class 1 valve bonnet-to-body closures. The external surface of the RCS piping and associated components is potentially exposed to borated water if the event of a leak should occur.

Corrosion wastage may be the result of the exposure to a leak. Since current activities monitor for leakage of borated water and take corrective actions in a timely manner, corrosion would not be allowed to continue. Therefore, an aging effect (material wastage) could not occur that would prevent the performance of the RCS piping intended function.

These activities include the leakage monitoring program at a plant. Corrective actions would be taken based on the results of the leakage monitoring program. In addition to other activities, this program includes walkdowns of the RCS before, during, and after each refueling outage. Minor leaks would be found, inspected, and cleaned at this time. Based on the results of the inspections, repairs would be made as necessary, including post-maintenance inspections.

RCS bolting materials falling within the scope of this report are for the RCP casing-to-cover, the Class 1 valve body-to-bonnet, and the flanges in the safety valve and resistance temperature detector (RTD) bypass lines. Either A193/B7 or SA540/B24 Class 4 have been specified for the pump casing-to-cover and either A/SA193/B7 or SA453/660 for the Class 1 valve body-to-bonnet bolting. In addition, the bolting material specifications for the flanges in the safety valve and RTD bypass lines should be the same as for the Class 1 valve body-to-bonnet bolting.

Leakage of primary coolant or the interaction between joint lubricants/sealing compounds and water could provide the aggressive environment needed for SCC in bolting materials. For quenched and tempered low alloy steels used for closure bolting such as Alloy 4140 and 4340 steels (e.g. SA193/B7, SA540/B23, SA540/B24), material susceptibility to SCC is controlled by its yield strength. EPRI report NP5769 [Ref. 29] indicates that SCC should not be a concern for closure bolting such as Alloy 4140 and 4340 steels in nuclear power plant applications if the specified minimum yield strength is below 150 ksi. The specifications of SA193/B7 and SA540/B24, Class 4 require the minimum yield strength of 105 ksi and 120 ksi, respectively.

SA453/660 material has been used successfully for bolting applications in nuclear power plants, although there was a failure by SCC due to a high stress (>100 ksi) application in the primary coolant environment. SA453/660 bolting should not have any concern for SCC in closure bolting applications for Class 1 valves, safety valve flanges, and RTD bypass line flanges since the applied stress for the bolting of these items should be much less than 100 ksi.

Operating experiences and existing data indicate that SCC failure should not be a significant issue for the bolting materials of SA193/B7, SA540/B24 Class 4, and SA453/660 on pump cover-to-casing, Class 1 valve bonnet-to-body, safety valve flange and RTD bypass line flange applications. There is, however, a concern of boric acid wastage for the low alloy steels (SA193/B7 and SA540/B24 Class 4). Current activities and program attributes to manage the event-driven effect of potentially significant corrosion due to boric acid wastage for these bolting materials is included in the leakage monitoring program at the plant.

For IGSCC to occur in austenitic stainless steel, three things must be present: a susceptible material, stress approaching or exceeding the yield strength of the material, and an aggressive environment such as an oxidizing environment. In the absence of one of the three above conditions, IGSCC will not occur; however, intergranular attack (IGA) can occur without a high stress. As to a susceptible material, Westinghouse has a policy of prohibiting the use of sensitized austenitic stainless steel Class 1 piping and associated components. Sensitization can be prevented by reducing the exposure of susceptible materials to the sensitization temperature range, 900EB1700EF, to short times (to quench the material after solution annealing above the sensitization temperature range). Westinghouse recognizes that in construction of Class 1 piping and components, they must be subjected to welding. To minimize the time that the Class 1 piping and components were heated into the sensitization temperature range, 900oB1700EF, Westinghouse controls the heat input during welding. The maximum interpass temperature is limited to 350EF to avoid sensitization of Class 1 piping and associated components materials. Even though Westinghouse Class 1 piping and associated components materials are procured in the solution annealed conditions and the heat input is controlled during welding, Westinghouse requires that IGA tests be performed in accordance with ASTM A262. [Ref. 30]

In addition to the steps Westinghouse takes to eliminate or reduce the susceptibility of Class 1 piping and component materials to sensitization, Westinghouse prevents sensitized stainless steels from coming in contact with an aggressive environment. Westinghouse specifies that the reactor coolant be rigorously controlled, particularly with regards to oxygen, chlorides, and other halogens.

The efficiency of the above practice in the prevention of IGSCC and IGA has been demonstrated by years of operating experience without exhibiting IGSCC or IGA in Class 1 piping and associated components. Therefore, the aging effects from IGSCC and IGA do not degrade the Class 1 piping and associated pressure boundary components intended function. By eliminating sensitized austenitic stainless steel Class 1 piping and associated components materials, the potential occurrence of SCC due to any sulfate from demineralizer resins and the oxygen level prior to and during shutdown is minimized. In laboratory experiments, even in cases where severely sensitized austenitic stainless steel has been deliberately exposed to PWR coolant, no intergranular attack has been observed.

For Class 1 piping and components manufactured from austenitic stainless steel, the effects caused by SCC and IGA do not degrade the Class 1 piping and associated components intended function.

Aging Effect Management

The potentially significant effects of corrosion due to boric acid leakage may occur on the external surfaces of RCP casings near bolted pump covers, Class 1 valve bonnet-to-body closures, and the flanges in safety valve and RTD bypass lines. Because of the current activities described above, the leakage event would be detected and corrosion would not be established long enough for an aging effect to occur. Since no aging effect results due to corrosion are caused by borated water leakage, no aging management program is required.

3.2.3 Irradiation Embrittlement

Mechanism Description

The types of radiation relevant to the aging assessment of Class 1 piping and associated components are neutron and gamma radiation. Materials exposed to neutron radiation undergo changes in microstructure and properties. The extent of the changes depends on the particular material, the neutron flux (n/cm2-sec), flux spectrum, exposure time or fluence (flux x time, n/cm2), and temperature.

Exposure to high energy neutrons (neutron energies greater than 0.1 MeV) can cause changes in the properties of stainless steel. This neutron irradiation can produce changes in mechanical properties by increasing yield and ultimate strength and correspondingly decreasing ductility and fracture toughness. The reduced fracture toughness causes a reduction in the critical flaw size for the piping, which is defined as the flaw size which could lead to failure. The extent of irradiation embrittlement is a function of both the irradiation temperature, which is the thermal temperature of the material, and the neutron fluence. The nominal irradiation temperature for Class 1 piping and associated components is determined by the primary coolant temperature (550EFB650EF) and local gamma heating rates. Data from power reactor irradiation of Type 304 and Type 316 stainless steel are available from several studies [Refs. 31 and 32]. Embrittlement, as evidenced by increases in yield strength and decreases in uniform and total elongation, is common in these materials after irradiation at high levels. Studies [Refs. 32 and 33] have shown that embrittlement of stainless steel occurs at fluences greater than 1 x 1020 n/cm2 (E > 0.1 MeV). These same studies have shown that the rate of change in mechanical properties is reduced at fluences above 2 x 1022 n/cm2 (E > 0.1 MeV). Programs have been established to determine the properties of materials exposed to irradiation in operating PWRs.

The principal effect of gamma irradiation is to deposit energy in the material being irradiated, which increases the temperature of the material (gamma heating).

Aging Effect Evaluation

No instance of Class 1 piping and components degradation attributed to irradiation embrittlement has been recorded. For license renewal, the maximum end-of-life levels for Class 1 piping and components is less than 1.5×10^{16} n/cm^2 (E > 0.1 MeV). The Class 1 piping and components most susceptible to irradiation embrittlement are those that are nearest to the reactor core. These components will experience some neutron irradiation exposure while remotely located components will receive relatively little. Since the expected neutron fluence for Class 1 piping and components is much less than the approximate threshold level of 1×10^{20} n/cm^2 (E > .1 MeV), the changes in mechanical properties due to neutron exposure are insignificant.

Aging Effect Management

Due to the lack of a detrimental aging effect caused by irradiation embrittlement, there is no need for management of this aging effect during an extended period of operation.

3.2.4 Thermal Aging

Mechanism Description

The effect of thermal aging refers to gradual and progressive changes in the microstructure and properties of a material due to exposure at an elevated temperature for an extended period of time. There are many effects of thermal aging, and the changes that occur may be desirable or undesirable. The only significant effect of thermal aging with respect to degradation of Class 1 piping and component materials is embrittlement of duplex ferritic-austenitic stainless steel castings.

Cast austenitic stainless steels are duplex structures consisting of austenite and ferrite. At high temperatures, the ferrite undergoes complex phase changes, often resulting in hardening of the ferrite. This, in turn, usually produces a reduction in fracture toughness, often as much as an order of magnitude. This embrittlement is referred to as an effect of thermal aging. The reduced fracture toughness causes a reduction in the critical flaw size for the piping, which is defined as the size flaw that could lead to failure. The embrittlement is usually characterized by a period of time at a temperature for which little or no embrittlement occurs, followed by a dramatic exponential type reduction in toughness. This reduction has an Arrhenius character, that is, short time aging at a higher temperatures can be equated to long time aging at a lower temperatures [Ref. 34].

While it has been known for some time that cast austenitic stainless steels embrittle at temperatures of 750°F or above (noticeable embrittlement occurring in just a few hundred hours or less), it is only in the last decade that a significant effect of thermal aging has been observed for longer times at operating temperatures of light water nuclear power plant primary coolant loops (525°F to 620°F) [Refs. 34 and 35]. These observations have led to considerable concern for the cast austenitic stainless steel product forms in the primary coolant loops of Westinghouse type PWRs. Welds in the primary loop also thermally age but usually respond more slowly due to low ferrite [Ref. 35].

Aging Effect Evaluation

Evaluations for thermal aging in the leak-before-break evaluations of CASS and fracture mechanics evaluations of RCP casings, which are considered to be time-limited aging analyses, are provided in Section 3.3.

Aging Effect Management

The potentially significant effect from thermal aging embrittlement may occur on cast austenitic stainless steels. These effects can be managed by demonstrating that structural integrity is maintained based on acceptable leak-before-break evaluations and fracture mechanics evaluations of RCP casings as described in Subsection 4.2.2. Alternatively, the component can be repaired or replaced.

3.2.5 Erosion

Mechanism Description

Erosion is a combined action of abrasion and corrosion. Material wastage is the aging effect resulting from erosion. Erosive wear is characterized as an increased rate of deterioration or attack on metal because of the relative movement between a corrosive environment and the metal surface. Erosion is attributed to the removal of protective surface films on a metal by mechanical action of a fluid or particulate matter. Erosion/corrosion occurs when the fluid or particulate matter is also corrosive to the metal. General erosion occurs under high-velocity conditions, turbulence, and impingement. Geometrical factors are extremely important. Carbon steels and low alloy steels are most susceptible to erosion/corrosion. Higher alloy steels, nickel-base alloys, and stainless steels are considered resistant to erosion and erosion/corrosion in a PWR environment. A basic discussion of flow-accelerated corrosion is provided in Chapter 1 of EPRI NSAC 202L [Ref. 36].

Aging Effect Evaluation

All of the Class 1 piping and associated components considered in the scope of this report are constructed of austenitic stainless steel that is resistant to erosion in a PWR environment. The loss of material from erosion due to the flow of fluid in the piping has a low probability of occurring based on the following:

- There is a relatively low fluid flow velocity in the Class 1 piping and components.
- Water is filtered prior to injection into the primary system, minimizing erosion due to particles in the fluid.
- The operating pressures of a PWR preclude cavitation erosion.
- The inside diameter of the primary loop piping is 100 percent machined or ground.

Therefore, the effects from erosion are not considered to be significant for the Class 1 piping and associated components.

Aging Effect Management

Due to the lack of a detrimental aging effect caused by erosion, there is no need for management of this aging effect during an extended period of operation.

3.2.6 Wear

Mechanism Description

Mechanical wear is defined as damage to a solid surface caused by removal or plastic displacement of material by way of mechanical contact characterized by loss of material during relative motion or sliding. Wear occurs in parts that experience intermittent relative motion, in clamped joints where relative motion is not intended but may occur due to a loss of clamping force, or via flow-induced vibrations.

Wear that is the result of the contact of two surfaces due to vibration or sliding (e.g., flow-induced vibration) while the surfaces are in the presence of a corrosive environment is referred to as fretting wear. Another type of wear that may occur in PWRs and that is not related to flow-induced vibration is associated with the intentional displacement of adjacent components. Wear can result from either surface oxide removal or the direct removal of base material.

Aging Effect Evaluation

A limited number of the RCS component parts covered by this report are subjected to relative motion. RCP and Class 1 valve closure parts, such as the cover and bonnet flanges, the casing and body flanges, and the closure bolting, are subject to some degree of relative motion if preload is lost if infrequent disassembly and reassembly operations occur. Loss of material due to wear could cause leakage for these closure elements. Mechanical wear is nonsignificant for Class 1 piping and the associated pressure boundary component or component parts, with the exception of the RCP and Class 1 valve closure elements such as the cover and bonnet flanges, the casing and body flanges, and the closure bolting [Ref. 12]. Current activities and program attributes to manage the effect of potentially significant mechanical wear with respect to these component parts are provided in Section 4.1.

Aging Effect Management

The potentially significant effect from mechanical wear may occur on the RCP and Class 1 valve closure elements such as the cover and bonnet flanges, the casing and body flanges, and the closure bolting. These effects can be managed by following the current and effective programs of periodic inservice inspection and testing for the detection and evaluation-repair-replacement of the closures as described in Subsection 4.1.1. Alternatively, the component can be repaired or replaced.

3.2.7 Creep and Stress Relaxation

Mechanism Description

Creep is the plastic deformation that occurs over a period of time in a material subjected to a stress that is typically below the elastic limit. Creep occurs at elevated temperatures where continuous deformation takes place under constant strain. Creep is not a concern for austenitic alloys below 1000EF.

Stress relaxation is similar to creep, but it occurs under conditions of constant strain where part of the elastic strain is replaced with plastic strain.

The unloading of preloaded components due to stress relaxation is caused by long-term exposure of materials to elevated temperatures and/or neutron irradiation. Leakage due to loss of closure bolt preload is an aging effect resulting from creep or stress relaxation of bolts. A material loaded to an initial stress may experience a reduction in stress over a period of time at high temperatures. At temperatures well above operating temperatures, the thermal effect for stress relaxation is predominant. It has been determined, however, that the presence of fast neutron irradiation can result in stress relaxation even at normal operating temperatures. When the irradiation effect is dominant, the rate of neutron impingement controls the number of vacancies formed in the component material. The presence of vacancies increases the likelihood that the material will plastically deform, resulting in the relaxation effect. Stress relaxation is particularly important in the design of bolted connections.

Aging Effect Evaluation

The maximum temperature experienced by Class 1 piping and components during normal and upset conditions is approximately 650°F, except for certain localized areas in the surge line where temperatures can be as high as 680°F. Even with a maximum temperature to 680°F, these temperatures are well below the temperature of 1000°F at which creep is a concern for any of the austenitic stainless steel PWR Class 1 piping and associated components. Therefore, the effect from creep is not significant for any PWR Class 1 piping and associated pressure boundary components.

The only Class 1 piping and associated pressure boundary components that could be affected by stress relaxation are those with bolted closures [Ref. 12]. The prestress in the bolts (or studs) can relax at sufficiently high temperatures. Neutron irradiation will not lead to stress relaxation of the preloaded bolted closures due to the relatively low fluence levels. The components covered by this evaluation that incorporate bolted closures are the RCP casing-to-cover closure bolting, and the Class 1 valve body-to-bonnet closure bolting.

Factors affecting the rate of stress relaxation are the material type, time, temperature, and degree of initial prestress. The loss of prestress occurs at a decreasing rate, and the majority of the loss is within the first year. The amount of prestress loss significantly decreases with time to approach an asymptotic value. Therefore, the level of prestress with extended operation should be comparable to that at 40 years.

Loss of preload through stress relaxation could lead to associated damage in one of two ways. First, excessive loss of preload or even excessive variability of preload could cause leakage through the bolted closure. Second, if the excessive loss of preload is permitted to continue uncorrected, there is a potential for cyclic loads to be imposed on the bolting that could increase fatigue usage. Such cyclic loading amplitudes would have to be large, or of long duration, so that relative motion of the mating surfaces would lead to detectable leakage. As a result, fatigue damage estimates for bolting are dominated by the joint makeup/ detorquing cycle and not by fluctuating cycles superimposed on the preload stresses. Therefore, the aging effect under consideration here is leakage through the bolted closure caused by excessive loss of preload.

While the relaxation of bolting preloads in reactor pump casing-to-cover and Class 1 valve body-to-bonnet closures can occur, the magnitude of the preload is intended to compensate for some loss. In spite of this margin and the asymptotic behavior of preload loss, stress relaxation is considered to be potentially significant for the RCP and Class 1 valve bolted closures. With the exception of these bolted closures, stress relaxation does not cause a significant aging effect for Class 1 piping and the associated pressure boundary components. Current activities and program attributes to manage the effect of potentially significant stress relaxation on closure bolting for the RCPs and Class 1 valves are provided in Section 4.1.

Aging Effect Management

The potentially significant effect of leakage due to loss of bolt preload from stress relaxation can occur on the RCP and Class 1 valve bolted closures. These effects can be managed by following the current and effective programs of periodic inservice visual inspection and leakage testing as described in Subsection 4.1.2. Alternatively, the component can be repaired or replaced.

3.3 TIME-LIMITED AGING ANALYSES METHODOLOGY EVALUATION

Section 2.5 identifies time-limited aging analyses (TLAAs) related to those Class 1 piping and associated components for which a CLB analysis exists. This section evaluates the TLAAs to determine if management is required. Results from current TLAAs have been projected to an extended period of operation. When the projected results are not acceptable, options will be presented in Section 4.0 to manage the identified aging effects.

Fatigue, which was described in Subsection 3.2.1, is evaluated using a time-limited aging analysis. All of the explicit analysis requirements to evaluate fatigue are defined in ASME Code, Section III. These requirements may vary depending on the methodology used for the evaluation. The B31.1 Power Piping Code fatigue design methodology is based on an implicit treatment of cyclic loadings. This section discusses methodologies as they apply to components with a fatigue design basis and the two operational issues that need to be considered before evaluating fatigue. The degradation sustained from the effects of fatigue were determined to be potentially significant for the fatigue-sensitive Class 1 piping and piping components and the Class 1 valve and RCP pressure boundary components (see Tables 3-2 through 3-17).

The leak-before-break (LBB) evaluations and the fracture mechanics evaluations of the RCP casings per Code Case N-481 are time-limited aging analyses. Most plants were licensed for LBB for the CLB. And, instead of performing the volumetric inspections for the RCP casings, most RCP casings have a fracture mechanics evaluation per the requirements of Code Case N-481. For CASS materials, the degradation of material toughness properties due to thermal aging are considered in the LBB evaluations and the fracture mechanics evaluations of the RCP casings. Structural integrity for a component can be demonstrated by evaluating it to LBB or code case N-481 criteria. Therefore, revalidation of the LBB evaluation and the Code Case N-481 fracture mechanics evaluation of the RCP casing will manage the thermal aging effects by demonstrating structural integrity for the component.

3.3.1 Class 1 Piping Fatigue Methodology for ASME Code, Section III Piping Design

Class 1 piping and the associated pressure boundary components are subject to fluctuating loads with a variety of occurrences, ranging from relatively infrequent to relatively frequent. Components that undergo significant thermal and seismic events are potentially susceptible to low-cycle fatigue damage. Class 1 thermowells identified in this report are the only pressure boundary items that are subjected to a dynamic load associated with flow-induced vibration and are potentially susceptible to high-cycle fatigue damage. The design bases for many Class 1 piping associated components have included fatigue evaluations designed to the ASME Code, Section III, Subsection NB.

ASME Code, Section III fatigue design procedures use a design fatigue curve, which is a plot of alternating stress range (S_a) versus the number of cycles to failure (N). The design fatigue curve is based on the un-notched fatigue properties of the material, modified by reduction factors that account for various geometric and moderate environmental effects. The fatigue usage factor (u) is defined by Miner's Rule as the summation of the damage over the total number of design basis transient types (I), as given by the ratio of expected cycles of that type (n_i) to the allowable number of cycles (N_i) for the stress ranges associated with that transient:

$$u = \sum_{i=1}^{I} \frac{n_i}{N_i}$$

For ASME Code design acceptance, the usage factor calculated in this manner cannot exceed unity (1.0) for the design lifetime of the component.

The conservatism in ASME Code fatigue calculations stems from two sources. First, the design fatigue curves contain either a factor of 2 on stress range or a factor of 20 on the number of cycles to failure, depending upon which is controlling. Second, a substantial margin is also expected to exist because of conservatisms in the magnitude and frequency of occurrence assumed for various design basis transients.

3.3.2 Class 1 Piping Fatigue Methodology for B31.1 Piping Design

For earlier plant designs, the Class 1 piping was designed to the rules of the B31.1 Power Piping Code. In a B31.1 evaluation, the fatigue issue is addressed by (1) minimizing vibration and thus preventing high-cycle fatigue failures and (2) applying a factor (f) to the allowable stress (S_a) in the evaluation of thermal moment plus pressure stress range. This factor "f" is a function of the number of applied cycles. The B31.1 approach does not consider the stresses resulting from combinations of severe longitudinal or circumferential thermal gradients and severe geometric discontinuities (e.g., a carbon/stainless steel interface). The EPRI report [Ref. 11] compared the results of fatigue design evaluation methods for piping designed to the ANSI B31.1 Code to those of the ASME Code, Section III for Class 1 piping. Two representative Class 1 piping systems designed to the ANSI B31.1 piping code—a PWR charging line and a BWR recirculation system—were selected for comparison with the ASME Code, Section III, Class 1 design rules. The results showed that two exceptions occurred, both of which were on the BWR recirculation system. These locations, a pipe-to-valve weld and a carbon/stainless steel dissimilar metal weld, represented geometric discontinuities where stresses were amplified due to severe hypothetical thermal transients. Most piping system locations do not represent geometric discontinuities nor do they experience severe thermal transients, since the relatively thick reactor pressure vessel determines heatup and cooldown rates. Based on the successful operating history of fossil plants (using the B31.1 approach) and the high cost of evaluating these stresses with a detailed fatigue analysis, this was considered to be an acceptable approach for nuclear plants.

3.3.3 Class 1 Valve Body Fatigue Methodology

The following is a discussion of one method that can be used to evaluate fatigue on the Class 1 valve bodies.

Initially the following general information would be required to perform the evaluation:

- A list of the Class 1 valves (or would be designated as) including the valve sizes, manufacturer, and drawings. This will be used to group the valves to determine if evaluations are required.

- Determine the code or standard that was used for valve construction by reviewing the specification and/or valve data package. This will determine what type of valve information and analysis is available.

  - If the valves are built to the draft pump and valve code or the ASME Code, the fatigue would have been evaluated in accordance to the procedures in the Code. The evaluations would be performed using the transients in the original specification.

    - Valves 4 inches and less do not require evaluation if they conformed to the requirements in the code.

    - Valves greater than 4 inches are evaluated in accordance with the code.

    Note: Valves that require evaluation would have had a design report covering fatigue supplied as part of the original code requirements.

  - If the valves are not built to the draft pump and valve code or the ASME Code, then no report/evaluation was performed.

- Evaluating the fatigue on the Class 1 valve bodies for license renewal.

  - Valves that were built to the draft pump and valve code or the ASME Code would require re-evaluation using the new transients following the code procedures.

- Valves that are not built to the code would be broken down into groups according to size: 4 inches and less, and greater than 4 inches.

  - The valves $\leq$ 4 inches would be evaluated in accordance with the ASME Code to determine if any analysis is required. See Subsection 4.2.1 for more details.

  - The valves greater than 4 inches would require an analysis in accordance with the ASME Code. The transients used would be those in the specification or a set of standard transients. The transients can be modified if the plant's transients are different.

- Rules pertaining to the Class 1 valve body fatigue evaluation are specific for each plant.

3.3.4 Fatigue for Class 1 Valve Bodies

The valve bodies are subjected to many transients during the life of the plant, from normal operating conditions to faulted service conditions, as defined in paragraph NB-3113 of the ASME B&PV Code. Due to the valve body configuration, there are discontinuities between the valve body run and the neck region (crotch region), which can result in high stress concentrations. Because of the discontinuities, the transients, pressures, and temperatures will result in repeated stress cycles that, when summed, may be significant enough to result in crack initiation in the body run to neck region.

Prior to the Draft Pump and Valve Code, no standard analysis method was available; therefore, no formal analysis was generally performed. If transients were supplied in the specification, they were reviewed for their severity against transients for similar operating valves. With the issuance of Draft ASME Code for Pumps and Valves for Nuclear Power, Article 4, followed by the ASME Code, Section III, NB-3500, a standard methodology was developed. For valves having a nominal pipe size equal to or less than 4 inches, no fatigue evaluation was required, provided the ASME criteria for the design of small valves were met. To evaluate the acceptability of the valve body for valves having a nominal pipe size greater than 4 inches for the effects of internal pressure, pipe reaction loads, and thermal loads, an analytical method was provided. The evaluation consisted first of an analysis of the internal pressure and pipe loads with a thermal secondary stress. The thermal secondary stress results from a through-wall temperature gradient and wall thickness variation based on a continuous ramp change in fluid temperature at 100°F per hour. This evaluation is used to determine the acceptability of the stresses in the valve body crotch region and the acceptability of the valve body for 2000 operating cycles.

The second analysis consists of an evaluation of the cyclic transients that the valve will be subjected to during the life of the plant. This analysis uses a design fatigue curve that consists of a graph of alternating stress versus the number of cycles to material failure. The curves are based on the fatigue properties of the material with a reduction factor to account for various design configurations for which the curves can apply. The fatigue usage factor is the summation of damage that occurs from the transients seen by the valve body. The factor is the summation of the ratios of the number of cycles the transient occurs to the allowable number of cycles for the transient based on the calculated stresses for the transient. Both the Draft Pump and Valve Code and ASME Code, Section III state that the fatigue usage factor cannot exceed 1.0 for the design life of the valve.

In determining the fatigue usage factor, an evaluation of the transients is performed. The transients that the valve body is subjected to over the design life vary in severity, may have been included in previous evaluations, or may not have to be considered in the evaluation. For those transients that have small temperature and pressure changes, as defined in the codes, the alternating stresses are low and need not be considered. In addition, if the number of occurrences are small (no more than 5), or the transient is associated with startup/ shutdown at temperature changes less than 100EF per hour for less than 2000 cycles, they can also be excluded.

There are many conservatisms in the fatigue calculations for the valve body. The design fatigue curves have a factor of 2 on the stress range and a factor of 20 on the number of cycles to failure, depending on which governs. Another conservatism is that the valve transients provided in the specifications are based on the postulated events the valve may be subjected to over the 40-year design life of the plant. Because these transients were chosen to represent worst-case conditions, the actual number of transients or the severity of the transients, pressures, and temperatures, result in a usage factor smaller than actually experienced.

3.3.5 Reactor Coolant Pump Fatigue Methodology

RCP Parts Other Than Casing

For the remainder of the RCP other than the casing, there are the pump closure pressure boundary components, various internal components, and active components such as the seals and shaft. The RCP closures include the thermal barrier flange; and, depending on the pump model, the main closure flange, bolting ring, diffuser flange and the associated bolts, nuts, and studs. Attached to the thermal barrier flange are the seal injection and component cooling water nozzles. Some of these nozzles have high fatigue usage factors and are considered to be the highest fatigue-sensitive areas for the RCP closures.

All of these components can be replaced by a new pump assembly, thereby minimizing downtime, and no justification would be required for extended service since they would be new components. If a utility wants to extend the life of pressure boundary components other than the casing, the various closure flanges and seal housing components would have to be evaluated further on a plant-specific basis in accordance with the licensing basis for license renewal. The stainless steel components may need a fatigue cyclic analysis or a code fatigue waiver analysis to justify additional operation for license renewal. The carbon steel main closure bolts and the seal housing bolts may need replacement due to corrosion. The thermal barrier assembly also supports the pump bearing and the auxiliary nozzles. Older pump designs may require replacement due to the inadequacy be the graphite oyaring and/or auxiliary nozzle designs.

Redesign of some components may be required for certain plants. Inspection of closure components may reveal inelastic deformation or other damaging effects that make replacement the easiest solution. Certain closure flanges will need gasket replacements also. The thermal barrier flange supports the thermal barrier, or heat exchanger cooling coil assembly, which isolates the bearing and seals from the hot loop water. These areas have high thermal gradients/stresses and may require replacement also. Additional activities and program attributes to manage the effect caused by fatigue on RCP closures are provided in Section 4.0. Other RCP parts, some of which are safety related items, are considered expendable and would be handled by maintenance programs.

3.3.6 Operational Issues Related to Fatigue

Two operational issues need to be considered before evaluating fatigue:

- Thermal stratification, cycling, and striping (IEB 88-08 (U.S. NRC))
- Thermal stratification in the pressurizer surge line (IEB 88-11 (U.S. NRC); IEN 88-80 (U.S. NRC); INPO SER 87-25)

For license renewal, the plant-specific commitments for IEB 88-08 and IEB 88-11 need to be maintained.

3.3.6.1 Thermal Stratification, Cycling, and Striping

Thermal stratification describes the condition where there is a significant temperature gradient in a fluid (stagnant or in motion) with the hot fluid at the top of the pipe. For thermal stratification to occur, the flow must be low enough for turbulent mixing not to be dominant so that the hot and cold fluids within the pipe remain separated. Thermal stratification is particularly damaging if an effect exists to promote thermal cycling. Thermal cycling causes stress cycles that can eventually lead to fatigue cracking and through-wall leakage. Therefore, it is important to identify where stratification can occur and also to identify if cycling is possible. One type of cycling occurs as a result of operational flow or temperature changes. Another less obvious cause occurs when a stratified flow enters a region where sufficient turbulence exists to disturb or even mix the stratified fluids. In this region, fatigue cracking is a potential issue since a large number of stress cycles can accumulate over a short period of time.

Thermal striping is a unique effect associated with stratified flows. Under certain thermal-hydraulic conditions a well-defined stratification interface can exist, i.e., the transition from the hot to cold fluid occurs over a short distance. If there is sufficient turbulence in the fluid, the interface can become unstable and fluctuate rapidly, in a wave-like fashion. At the location of the fluctuation, thermal stresses can cycle, causing a localized fatigue loading during operation.

Thermal stratification, cycling, and striping (which causes metal fatigue) can result from several root causes as shown below:

- Operating a system with certain (generally low) flow rates and temperatures could result in stratification and possibly striping in horizontal piping sections. This root cause is associated with the stratification in the pressurizer (PZR) surge line and feedwater lines.

- A pipe section without flow during normal operation (dead-end) could be subject to stratification and cycling if a leak were to occur into the dead-end section. This has occurred in safety injection and residual heat removal (RHR) (decay heat removal) piping. Cases of in-leakage, out-leakage and cross-leakage are in this category.

- A pipe section without flow during normal operation could be subject to large temperature differences as a result of conductive or convective heating through a pressure boundary (such as a closed valve). This has been found to occur on safety injection piping, particularly where the isolation valve is located close to the main coolant pipe.

- A pipe section open to a steam environment can have steam condense and partially fill a horizontal section. This is potentially an issue in PZR spray lines in the region just above the PZR.

- A pipe section without flow during normal operation, connected to a pipe with high-temperature, high-velocity fluid, may be susceptible to temperature changes resulting from operational changes. This case is related to turbulent penetration length and is dependent on line layout, as explained later in this section.

Although this list of root causes is not exhaustive, most thermal stratification, cycling, and striping issues can be categorized within these definitions. It is possible that a pipe section be susceptible to more than one of the root causes noted above. It is important to know that thermal stratification, cycling, and striping can cause fatigue and lead to fatigue cracking and leakage from plant operation experiences.

3.3.6.2 Thermal Stratification in the Pressurizer Surge Line

For the PZR surge line thermal stratification, heatup and cooldown operations are of primary concern because of the temperature difference, termed as system ΔT, which occurs between the PZR and primary system hot leg. (Pipe stresses due to stratification are generally proportional to the top-to-bottom pipe ΔT which is, in turn, limited by the system ΔT.) There are two basically different methods of plant operation used in the heatup and cooldown of domestic Westinghouse PWRs: water solid and steam bubble. The maximum system ΔT is significantly different between these two methods.

Water Solid vs. Steam Bubble for Heatup and Cooldown Operation

All pressurized water reactor coolant systems (RCSs) operate with a steam bubble in the PZR. The steam bubble serves as a cushion to absorb and mitigate pressure transients caused by changes in the mass inventory or temperature of the RCS. The steam bubble also provides pressure control capability during power operation and helps maintain adequate pressure to prevent departure from nucleate boiling (DNB). Energizing PZR heaters increases bubble size and system pressure. Initiating PZR spray condenses some of the steam, reducing bubble size and system pressure.

Plants that use the steam bubble method of heatup form the bubble just prior to or shortly after startup of the first RCP. These pumps require approximately 350 psig system pressure before startup for the seals to function properly and to avoid suction voiding (flashing of water to steam) on the suction side of the pump. At this pressure, the water within the PZR must be heated to approximately 435°F to form steam. During and immediately following bubble formation, water in the RCS remains relatively constant at approximately 120°F to 180°F, depending on such factors as RHR usage, core decay heat, and reactor coolant pump (RCP) usage. It is this period of time when surge line stratification is most severe. The system ΔT is at a maximum and, with pressure still relatively low, the bubble is more sensitive to small pressure changes in the RCS. The result is an alternating insurge and outsurge, bringing cold and hot water, respectively, into the surge line. This action causes the surge line to become stratified in a varying (cyclic) manner.

Cooldown, using the steam bubble method, results in similar conditions. During cooldown, the reactor coolant temperature is reduced prior to collapsing the bubble. The bubble is collapsed after or shortly before shutdown of the last RCP. The period of time prior to bubble collapse, with high system ΔT and flexible steam bubble, is the most severe part of cooldown in regard to thermal stratification. System ΔT values during cooldown are generally less than during heatup because the reactor coolant temperature is generally warmer.

Plants that use the water solid method of heatup form the steam bubble later in the heatup process. The RCPs are started and used to heat the system (250°F to 350°F) before the bubble is formed. During this part of the heatup, PZR spray is used to circulate reactor coolant water through the PZR, keeping boron concentrations uniform and the system ΔT at or near zero. When the heaters are energized and the steam bubble is formed, (approximately 350 psig), the system ΔT is typically 200°F or less. Water solid cooldown results in similar conditions. The bubble is collapsed while still at relatively high reactor coolant temperatures.

Other Operational Effects on Surge Line Thermal Stratification

Although the most important operational effect on surge line stratification is water solid versus steam bubble methods, other practices also have significant effects.

Operation of the RCP, in the loop containing the surge line, influences both the severity and cyclic nature of surge line stratification. Monitoring data indicate that, while this pump is running, stratification is eliminated at the reactor coolant loop (RCL) nozzle and generally attenuated throughout the surge line. This is due to turbulent mixing in the nozzle region caused by reactor coolant flow. Starting and stopping this pump causes stratification to cycle. At the nozzle, this is due to the appearance and disappearance of turbulent mixing. In the surge line, this is due to associated pressure fluctuations that cause entry of cold (insurge) or hot (outsurge) water depending on whether the RCS pressure is increasing or decreasing. Both insurges and outsurges cause stratified conditions within the surge line when a significant system ΔT exists. Many other events can also cause pressure fluctuations and result in stratification cycling. However, monitoring data suggest that pump starts and stops cause the most severe stratification transients, especially with regard to the RCL nozzle.

Other events that can cause insurges and outsurges during periods of high system ΔT include charging and letdown mismatches, cyclic PZR spray operation, and sudden changes in residual heat removal (RHR) operation. Cyclic spray operation causes an insurge of cold water as the bubble condenses and an eventual outsurge of hot water as spray is terminated and the system returns to initial condition.

Fatigue Evaluation with Thermal Stratification in Surge Line

To evaluate the surge line fatigue usage factor, the thermal design transients are required to be modified to reflect the thermal stratification. The design transients for the surge line consist of two major categories:

- Heatup and cooldown transients
- Normal and upset operation transients

In the evaluation of surge line stratification, the definition of normal and upset design events and the number of occurrences of the design events remains unchanged.

The total number of current heatup-cooldown cycles (200) remains unchanged. However, subevents and the associated number of occurrences are defined to reflect stratification effects.

For all transients, the surge line fluid temperature distribution is modified from the original uniform temperature to a stratified distribution with the maximum temperature differentials and the associated PZR and hot leg nominal temperatures.

Operational Data Review

A review of historical operating records can be undertaken to determine the actual number of design transients accumulated, classify partial cycles (reactor trips from 30 percent as opposed to full power, for example) and incorporate any transients that have occurred but were not considered in the design basis. This operating transient set can then be used as a basis for fatigue evaluation. Operational data reviews of this kind are described in Refs. 15, 16, and 37.

A simplification of this process can be achieved if it can be established that existing component fatigue evaluations are influenced by certain transients that have occurred much less frequently or with substantially less severity than originally anticipated. Examples of this are postulated seismic events and load follow transients. The effects of seismic events and load follow transients that have not occurred during the design life can be eliminated from fatigue evaluations and appropriate projections made for the license renewal term. This kind of approach has been used in several studies [Refs. 38, 39, and 40].

Transient Monitoring

A program to monitor and record data can also be used to provide information on operating transients. Such a program can supplement an operational data review or can stand alone.

The final product of this process is a detailed transient data set on which realistic fatigue analyses of Class 1 piping and associated pressure boundary components can be based. Fatigue usage factors and, if necessary, fracture mechanics analyses can then be applied to the license renewal period with a minimum of excess conservatism.

3.3.7 Thermal Aging Effect Evaluation

The piping in the primary coolant loops of Westinghouse type PWRs may be forged (SA376 TP316 or SA376 TP304) or centrifugally cast (SA351 CF8M or SA351 CF8A). All the elbows are statically cast (SA351 CF8M or SA351 CF8A). The primary loop pump casings and Class 1 valve bodies are also static castings (SA351 CF8 or SA351 CF8M). Various combinations of these materials exist among many plants.

Thermal aging is considered to be potentially significant for Class 1 piping, RCP casing, and Class 1 valve bodies that are made of cast austenitic stainless steel (CASS) and also for the associated welds. Additional activities and program attributes to manage the effect of potentially significant thermal aging on the CASS Class 1 piping and pressure boundary components including the welds are provided in Section 4.2. The degradation sustained from thermal aging is nonsignificant to all other Class 1 piping and pressure boundary components covered by this report since they are not CASS, and, as such, are less susceptible to the effects of thermal aging due to their inherent low ferrite composition.

LBB Evaluations

The structural design basis for the primary loop piping and components required postulating nonmechanistic circumferential pipe breaks. This resulted in plant hardware (e.g., pipe whip restraints and jet shields) to mitigate dynamic consequences of pipe breaks. An LBB evaluation provides a mechanistic pipe break analysis method that can be used to establish that circumferential pipe breaks will not occur within the primary loop piping.

LBB evaluations have been performed for the primary loop in the majority of Westinghouse PWR plants. These evaluations follow the recommendations and criteria proposed in NUREG 1061, Volume 3 [Ref. 49]. The criteria and resulting steps of the evaluation procedure can be briefly summarized as follows:

- Calculate the applied loads and identify the location at which the highest stress occurs.
- Identify the materials and associated material properties.
- Postulate a surface flaw at the governing location and determine fatigue crack growth, showing that a through-wall crack will not result.

- Postulate a through-wall flaw at the governing location. The size of the flaw should be large enough so that the leakage is assured of detection with margin using the installed leak detection equipment when the plant is subject to normal operating loads. A margin of 10 is demonstrated between the calculated leak rate and the leak detection capability.

- Using the maximum faulted loads, demonstrate a margin of at least 2 between the leakage size flaw and the critical flaw size.

- Review the operating history to ascertain that operating experience has indicated no particular susceptibility to failure from the effects of corrosion, water hammer, or low- and high-cycle fatigue.

- For base and weld metals, provide the plant-specific material properties including toughness and tensile test data. Evaluate long-term effects such as thermal aging.

- Demonstrate margin on applied loads.

The LBB analyses, including the effect of thermal aging, is performed using the methodology described in Standard Review Plan 3.6.3 [Ref. 50].

Thermal Aging in LBB Evaluations

In 1983, the U.S. NRC requested that thermal aging degradation be addressed in demonstrating piping integrity by the LBB approach for all future LBB submittals by utilities. Westinghouse developed criteria for evaluating effects of thermal aging.

Integrity evaluations rest mainly on the application of elastic-plastic fracture mechanics and leak calculation methodologies. One of the primary inputs to an evaluation is the elastic-plastic fracture criteria in which the calculated applied fracture toughness values are compared against material fracture toughness values. In general, the J-integral approach has been applied with the following criteria:

$$J_{app} < J_{Ic}, \text{ or}$$

$$\text{If } J_{app} > J_{Ic}, \text{ then } T_{app} < T_{mat}$$

where T_{mat} is the material tearing modulus and the subscript "app" designates applied.

Additionally, the U.S. NRC has required the following condition to be satisfied as well:

$$J_{app} < J_{max}$$

where J_{max} does not exceed the maximum value of J determined from material test or chemistry information.

A correlation based on the chemistry of the cast material for estimating a room temperature Charpy U-notch (KCU) impact value has been developed in Reference 35. In addition, based on a 40-year operating license, a significant amount of fracture toughness data has been generated on a highly sensitive heat of cast stainless steel pipe material.

These models were developed from a large body of experimental data obtained by Fischer of Switzerland [Ref. 34]. The following equation was developed based on all available data.

for 10,000 hours aging at 752°F (400°C):

$$KCU\ (daJ/cm^2) = 52.5 - 2.19\ (Si + Cr + Mo) + 46/\ F$$

Where Si is silicon in percent weight, Cr is chromium in percent weight, Mo is molybdenum in percent weight, and F is percent ferrite determined by the Schoeffer Method.

This equation was determined by Slama et al. [Ref. 35] to result in Charpy values equivalent to the minimum Charpy values expected during service for CF8M cast stainless steel. This equation is applicable regardless of the temperature of operation of the piping (which will of course be different in the hot and cold legs). Slama, et. al. calculated using time-temperature equivalencies that the aging times at 752°F (400°C) corresponding to the total 32-year service life for CF8M ranged from 13,000 hours for the cold leg (554°F [290°C]) to 34,000 hours for the hot leg (608°F [320°C]). In studying the available data, however, they found that the minimum properties were obtained only after 10,000 hours and therefore this time was used.

The above equation is based on the actual ferrite percentages determined by Fischer on 15 heats of cast stainless steel, using magnetic measurement. Slama's verification of the model was accomplished using the Schoeffer diagram values of ferrite content, as normally reported on material test certificates. The ferrite levels determined in this manner were found to be within 1-3 percent of levels determined magnetically and by quantitative metallography, and the model predicted the behavior of Slama's additional heats well. The data base used to develop the model included ferrite contents ranging from 6-42 percent.

All the RCSs examined through the end of 1985 met the 40-year design life LBB criteria. However, isolated cases for heats of materials were found where the correlation based on chemistry did not produce the minimum required energy equivalent to the limiting material mentioned previously. Accordingly, general toughness criteria for thermally aged cast stainless steel were developed for calculated KCU impact values that could not be demonstrated to be as good as reference material.

As described previously, the thermal aging of austenitic-ferritic stainless steel occurs at RCL temperatures as a chromium rich phase, alpha prime (α'), precipitates in the ferritic phase. The precipitation of the α' stage is mainly responsible for the hardening and embrittlement experienced by the steel. A research program was conducted to determine quantitatively the influence of aging on material toughness and to determine what material properties can be useful in predicting end-of-life toughness of stainless steel components of the RCL [Ref. 35].

Through multiple linear regression analysis, the end-of-life material toughness was correlated to the material's silicon, chromium, and molybdenum contents and to the ferrite content.

Fracture Mechanics Analysis

Thermal Aging in Fatigue Mechanics Analysis of RCP Casings per Code Case N-481

In lieu of performing volumetric inspections of RCP CASS casing, a fracture mechanics analysis, supplemented by visual examinations, per the requirements of ASME Code, Section XI Case N-481, can be performed for the current term of operation. For the license renewal term, a similar fracture mechanics analysis can be used to assess the structural integrity of the RCP casings, since there is no specific lifetime limit in Section XI or its code cases.

Several elastic-plastic fracture mechanics methods are available for the integrity assessment. The most commonly used is the J approach involving a crack initiation toughness, J_{Ic}, and a ductile tearing resistance toughness usually stated in terms of the tearing modulus, T_{mat}. These toughness parameters are routinely measured for structural materials including stainless steel. Sophisticated analytical techniques are available for calculating the applied J and T for a variety of complex flawed structures.

The application of fracture mechanics is always associated with flaws (i.e., cracks) or potential flaws in a structure, and this is closely related to the reliability and accuracy of inspection methods and the accessibility of the component of interest. Should a flaw or crack-like defect be found or hypothesized, the evaluation for serviceability involves the elastic-plastic fracture analyses and the fracture criteria discussed above to evaluate the current integrity and a crack extension evaluation (usually fatigue crack growth) in concert with a fracture evaluation to assure integrity for continued service in the future.

The materials used in the RCP casings and the nature and extent of degradation during service factor into a fracture mechanics assessment. RCP casings made from cast stainless steel are subject to thermal aging embrittlement, which is a time-dependent phenomenon. Initially the stainless steels are tough and crack-resistant. Thermal aging has been demonstrated to produce significant reduction in fracture toughness in some heats of cast stainless steel in time periods approaching service life at the service temperatures.

The affected materials in the product forms of interest will retain some resistance to brittle fracture, with ductile tearing being a dominant mode of fracture. This suggests that elastic-plastic fracture mechanics methods should be used in the integrity evaluation of those components in which a flaw or crack-like inclusion may exist or is hypothesized.

Code Case N-481 Requirements

In ASME Code Case N-481 it is stated that the following requirements shall be met in lieu of performing the volumetric examination on the reactor coolant loop pump casings specified in Table IWB-2500-1, Examination Category B-L-1, Item B12.10:

- Perform a VT-2 visual examination of the exterior of all pumps during the hydrostatic pressure test required by Table IWB-2500-1, Category B-P.

- Perform a VT-1 visual examination of the external surfaces of the weld of one pump casing.

- Perform a VT-3 visual examination of the internal surfaces whenever a pump is disassembled for maintenance.

- Perform an evaluation to demonstrate the safety and serviceability of the pump casing. The evaluation shall include the following:

  - Evaluating material properties, including fracture toughness values

  - Performing a stress analysis of the pump casing

  - Reviewing the operating history of the pump

  - Selecting locations for postulating flaws

  - Postulating one-quarter thickness reference flaw with a length six times its depth

  - Establishing the stability of the selected flaw under the governing stress conditions

  - Considering thermal aging embrittlement and any other processes that may degrade the properties of the pump casing during service

- A report of this evaluation shall be submitted to the regulatory and enforcement authorities having jurisdiction at the plant site for review.

In performing the evaluation for the Westinghouse Owners Group, plants required by *d1* of the code case, the effects of thermal aging have been incorporated. The fracture mechanics evaluation is similar to an LBB evaluation. However, for the current term of operation, fracture toughness is estimated based on NUREG/CR-4513, Rev. 1 [Ref. 52].

3.4 AGING EFFECT EVALUATION SUMMARY

This section contains a summary of the aging effects investigated in this report. Those effects that could cause potentially significant degradation to Class 1 piping and associated pressure boundary components (Class 1 valve bodies and RCP casings) are covered by this evaluation.

The following aging effects have potential to degrade the intended function of Class 1 piping and associated components:

- Fatigue-related cracking for fatigue-sensitive items
- Thermal aging-related cracking of austenitic stainless steel static castings

- Material loss caused by wear of reactor coolant pump (RCP) and Class 1 valve closure elements
- Loss of bolt preload due to stress relaxation of bolted RCP and Class 1 valve closures

Potential aging effects were assessed for each of the typical piping components in each pipe line. Most of these degradation effects were not a major concern for the life extension and license renewal of the piping and piping components. To assess the effects, a matrix of mechanisms verses the piping components was developed for each pipe line. The effects were evaluated and given a rating or qualitative probability for the component in one of the following categories:

N =Components that were considered to not be an issue

I-M =Components that were considered to be possible issues but were manageable due to plant actions such as inservice inspection

I-RA =Components that were considered to be possible issues that would require an aging management program

The matrix of effects and ratings for the components is shown in Tables 3-2 through 3-17. Table 3-17 provides a summary that includes all Class 1 piping and associated components, and Tables 3-2 through 3-16 specifically address Class 1 piping and piping components.

The fatigue entries in Tables 3-2 through 3-16 were developed from the Westinghouse generic fatigue data base (GFDB), which is an accumulation of components and applicable usage factors for Class 1 piping systems that have been evaluated for fatigue. This data base is used to compare specific loads and component properties used in the GFDB against plant-specific loads and component properties. The fatigue qualification performed in the GFDB for each component is used for a plant-specific application as long as the plant-specific data is shown applicable.

The plant-specific data compared to the GFDB is listed as follows:

- Moment stress range is less than that in the GFDB.
- Material allowable stress is greater than that used in the GFDB.
- Geometry of the component is similar or discontinuities are enveloped by the GFDB.
- Transient definition and number of cycles are enveloped by the GFDB.
- ASME Class 1 stress indices are enveloped by those in the GFDB.

If the plant-specific data are not shown applicable, then further evaluations are required.

The moment stress ranges used in the GFDB are typically high enough so that most components can be shown acceptable without further evaluation. The material for most components in the GFDB is type 304 stainless steel, which is similar in most plants. The geometries for most components are similar to those in the GFDB. Transients and cycles for auxiliary lines and for most of the loop piping from plant to plant are similar to those used in the

GFDB, and most of the ASME Class 1 stress indices are the same as those used in the GFDB. Therefore, the GFDB was used as a reasonable conservative upper bound representation of the usage factors for a typical plant. Those components identified as not being fatigue sensitive are considered to not need further review for ASME, Section III Class 1, B31.7, and B31.1 piping designs. There are three categories that are used to identify the degree of sensitivity for each component. These three categories are N, I-M, and IRA. The N category identifies the components that are not fatigue-sensitive. The other two categories are both fatigue-sensitive. However, I-M refers to those components where added calculations are needed to show the component acceptable, and IRA will require either additional fatigue evaluation or an aging management program (AMP). Those components that are identified as an issue requiring a generic fatigue evaluation or an issue requiring an AMP will need to be addressed further. As many as half of the components can be removed from the list of fatigue-sensitive components by performing additional generic analysis. These generic analyses can be performed to show acceptability for 60 year life by any one or any combination of the following methods:

- Reduce the severity of the thermal transient or the piping loads to more closely represent plant operation
- Reduce the stress intensification factors used in the generic fatigue
- Reduce the number of cycles to more closely represent plant operation

The fatigue information in Tables 3-2 through 3-16 was determined for each system by listing all of the component types in the GFDB except for the trunnions, lugs, nonstandard components and super stiff clamps that must be evaluated on a case-by-case basis. The corresponding usage factor for each component in the GFDB was increased by 1.5 to account for 60 years of design cycles. If the usage factor was less than 1.0, then the component was not considered to be fatigue-sensitive. If the usage factor was greater than 1.0, then additional analysis is needed to show the component is acceptable for 60 years of design cycles.

The B31.1 code does not require a fatigue evaluation for valve bodies and RCP closures. In some cases, the B31.1 plant designs include Class 1 valve bodies that were designed to the ASME code for fatigue. Most of the plant-specific ASME fatigue evaluations for the valve bodies were performed by or were subcontracted by vendors and are maintained by the owners and vendors. Since the scope of the Westinghouse fatigue design basis for valve bodies does not include all of the fatigue evaluations, and since the B31.1 code does not require a fatigue evaluation for the valve bodies and RCP closures, the valve bodies and RCP closures were included in the list of fatigue sensitive items (see Table 3-17). A set of generic fatigue evaluations could be performed to qualify most of the ASME, draft pump and valve, and B31.1 valve bodies and RCP closures (with the possible exception of the seal injection and component cooling water nozzles attached to the thermal barrier flange) to the ASME fatigue requirements for the license renewal term.

3.4.1 Fatigue

Degradation sustained from the effects of fatigue was determined to be potentially significant for the fatigue-sensitive Class 1 piping and piping components, the Class 1 valve bodies greater than 4 inch nominal pipe schedule (NPS), and the RCP pressure boundary closure components (see Tables 3-2 through 3-17). This determination has its basis in analysis, test, and experience. A review of calculated usage factors for Class 1 piping and piping components designed to the ASME Code, Section III, Subsection NB, and a comparison of geometric and operating similarities served to identify the fatigue-sensitive Class 1 piping and piping components. Section 4.0 provides aging management program attributes to manage the effects caused by fatigue (AMP-3.3, 3.4, and 3.5). For all other Class 1 piping and associated components covered by this report, fatigue is nonsignificant.

3.4.2 Corrosion

Degradation sustained from the effects of corrosion was determined to be potentially significant for the external surfaces near the bolted connections of the RCP casing closures, the Class 1 valve body-to-bonnet closures, the flanged connections for the safety valve and RTD bypass lines, and the associated HSLA bolts that may be subject to boric acid corrosion from leaking primary coolant. Since current activities monitor for leakage of borated water and take corrective actions in a timely manner, corrosion would not be allowed to continue. Therefore, an aging effect (material wastage) could not occur that would prevent the performance of the RCS piping intended function. For all other Class 1 piping and components covered by this report, the effects of corrosion are nonsignificant.

The degradation sustained from the effects of stress corrosion cracking is nonsignificant to all Class 1 piping and associated components covered by this evaluation.

3.4.3 Irradiation Embrittlement

Degradation sustained from the effects of irradiation embrittlement is nonsignificant to all Class 1 piping and associated components covered by this evaluation.

3.4.4 Thermal Aging

Degradation sustained from the effects thermal aging was determined to be potentially significant for the cast austenitic stainless steel (CASS) piping and associated pressure boundary components. Section 4.0 provides aging management program attributes to manage the effects caused by thermal aging. For all other Class 1 piping and associated components covered by this evaluation, thermal aging is nonsignificant.

3.4.5 Erosion

Degradation sustained from the effects of erosion is nonsignificant to all Class 1 piping and components covered by this report.

3.4.6 Wear

Degradation sustained from the effects of wear was determined to be potentially significant for the RCP and Class 1 valve closure parts. Section 4.0 provides aging management program attributes to manage the effects caused by wear. For all other Class 1 piping and associated components covered by this evaluation, the effects of wear are nonsignificant.

3.4.7 Creep and Stress Relaxation

Degradation sustained from the effects of creep is nonsignificant to all Class 1 piping and associated components covered by this evaluation. Degradation sustained from the effects of stress relaxation was determined to be potentially significant for the RCP casing to cover closure bolting and the Class 1 valve body to bonnet closure bolting. Section 4.0 provides aging management program attributes to manage the effects caused by stress relaxation. For all other Class 1 piping and components covered by this report, the effects of stress relaxation are nonsignificant.

TABLE 3-2
PIPING DEGRADATION EFFECT EVALUATION
REACTOR COOLANT LOOP PIPING SYSTEM

Components	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue[1]		Stress Relaxation
								Design Basis and Operation Issues	Operation Issues	
Hot & Cold Leg Loop Stop Valve Transitions	N	N	N	N	N	N	N	N	N	N
Sample Nozzle	N	N	N	N	N	N	N	N	N	N
LFI Nozzles	N	N	N	N	N	N	N	N	N	N
Thermowell Bosses	N	N	N	N	N	N	N	N	N	N
CL Elbow	N	I-RA	N	N	N	N	N	N	N	N
XOL Elbows	N	I-RA	N	N	N	N	N	N	N	N
SG Inlet Elbow	N	I-RA	N	N	N	N	N	N	N	N
RPV Inlet Nozzle Safe End	N	N	N	N	N	N	N	I-RA	N	N
RPV Outlet Nozzle Safe End	N	N	N	N	N	N	N	I-RA	I-M	N
SG Inlet Nozzle Safe End	N	N	N	N	N	N	N	I-RA	I-M	N
SG Outlet Nozzle Safe End	N	N	N	N	N	N	N	I-RA	N	N
RCP Inlet Nozzle Safe End	N	N	N	N	N	N	N	N	N	N
RCP Outlet Nozzle Safe End	N	N	N	N	N	N	N	N	N	N
Straight Pipe (wrought)	N	N	N	N	N	N	N	N	N	N
Straight Pipe (centrifugally cast)	N	I-RA	N	N	N	N	N	N	N	N

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-3
PIPING DEGRADATION EFFECT EVALUATION
PRESSURIZER SURGE LINE SYSTEM

Components 12-inch and 14-inch Sch 140	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue[1]		Stress Relaxation
								Design Basis and Operation Issues	Operation Issues	
Pressurizer Nozzle	N	N	N	N	N	N	N	I-RA	I-RA	N
Thermowell Boss	N	N	N	N	N	N	N	I-RA	I-RA	N
Large Radius Elbow	N	N	N	N	N	N	N	I-RA	I-RA	N
Large Radius Bend	N	N	N	N	N	N	N	N	N	N
Butt Weld	N	N	N	N	N	N	N	I-M	I-M	N
Straight Pipe	N	N	N	N	N	N	N	N	N	N
Welded Attachments	N	N	N	N	N	N	N	I-RA	I-RA	N
RCL Branch Nozzle	N	N	N	N	N	N	N	I-RA	I-RA	N

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-4
PIPING DEGRADATION EFFECT EVALUATION
PRESSURIZER SPRAY LINE SYSTEM

Components 6-inch and 4-inch Sch 160	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue[1] Design Basis and Operation Issues	Fatigue[1] Operation Issues	Stress Relaxation
Pressurizer Nozzle	N	N	N	N	N	N	N	I-RA	I-RA	N
Thermowell Boss	N	N	N	N	N	N	N	N	N	N
Long and Short Radius Elbow	N	N	N	N	N	N	N	N	N	N
6-inch x 4-inch Reducers	N	N	N	N	N	N	N	N	N	N
4-inch Valve Transitions	N	N	N	N	N	N	N	N	N	N
Branch Pipes	N	N	N	N	N	N	N	I-RA	I-RA	N
Large Radius Bend	N	N	N	N	N	N	N	I-RA	N	N
Butt Welds	N	N	N	N	N	N	N	I-RA	N	N
Straight Pipe (steam or water filled)	N	N	N	N	N	N	N	N	N	N
Welded Attachments	N	N	N	N	N	N	N	I-RA	N	N
6-inch x 4-inch Non Standard Reducing Tee	N	N	N	N	N	N	N	I-RA	N	N
4-inch X 3/4-inch Branch Pipes	N	N	N	N	N	N	N	I-RA	N	N
6-inch X 2-inch Branch Pipes	N	N	N	N	N	N	N	I-RA	I-RA	N
RCL Branch Nozzles	N	N	N	N	N	N	N	N	N	N
Steam Filled Butt Welds, Tees, Transitions, and Branches	N	N	N	N	N	N	N	I-RA	I-RA	N

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-5
PIPING DEGRADATION EFFECT EVALUATION
AUXILIARY PRESSURIZER SPRAY LINE SYSTEM

Components 2-Inch Sch 160	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue[1]		Stress Relaxation
								Design Basis and Operation Issues	Operation Issues	
Socket Welds	N	N	N	N	N	N	N	I-RA	I-RA	N
2-inch Socket Welded Valve Transitions	N	N	N	N	N	N	N	I-RA	I-RA	N
Large Discontinuity Branch Pipes	N	N	N	N	N	N	N	I-RA	I-RA	N
Branch Pipes	N	N	N	N	N	N	N	N	I-RA	N
Socket Welded Elbow	N	N	N	N	N	N	N	N	I-RA	N
Butt Welds	N	N	N	N	N	N	N	N	I-RA	N
Straight Pipe	N	N	N	N	N	N	N	N	I-RA	N
Welded Attachments	N	N	N	N	N	N	N	I-RA	I-RA	N
Straight and Reducing Tees	N	N	N	N	N	N	N	I-RA	I-RA	N

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, and nonstandard components, super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-6
PIPING DEGRADATION EFFECT EVALUATION
PRESSURIZER SAFETY & RELIEF LINE SYSTEM

								Fatigue[1]		
Components 6-inch and 3-inch Sch 160	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Design Basis and Operation Issues	Operation Issues	Stress Relaxation
Pressurizer Nozzles	N	N	N	N	N	N	N	I-RA	N	N
3-inch Valve Transitions	N	N	N	N	N	N	N	I-RA	N	N
Branch Pipes	N	N	N	N	N	N	N	I-RA	N	N
Long and Short Radius Elbow	N	N	N	N	N	N	N	N	N	N
Butt Welds	N	N	N	N	N	N	N	N	N	N
6-inch Flange and Bolts	N	N	I-M*	N	I-M	N	N	I-RA	N	I-M
Straight Pipe	N	N	N	N	N	N	N	N	N	N
6-inch x 3-inch Reducers	N	N	N	N	N	N	N	I-RA	N	N

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-7
PIPING DEGRADATION EFFECT EVALUATION
ACCUMULATOR INJECTION LINE SYSTEM

Components 10-inch and 12-inch Sch 140 & 160	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue[1]		Stress Relaxation
								Design Basis and Operation Issues	Operation Issues	
45° Accumulator Injection Nozzles	N	N	N	N	N	N	N	I-RA	N	N
10-inch & 12-inch Valve Transitions	N	N	N	N	N	N	N	I-RA	N	N
Branch Pipes	N	N	N	N	N	N	N	I-RA	N	N
Long and Short Radius Elbow	N	N	N	N	N	N	N	N	N	N
Butt Welds	N	N	N	N	N	N	N	N	N	N
6-inch and 8-inch Straight and Reducing Tees	N	N	N	N	N	N	N	I-RA	N	N
Straight Pipe	N	N	N	N	N	N	N	N	N	N

<u>Notes:</u>

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, and nonstandard components, super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-8
PIPING DEGRADATION EFFECT EVALUATION
COLD LEG SAFETY INJECTION LINE SYSTEM

Components 2-Inch, 6-Inch, 8-Inch, and 10-Inch Sch 160	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue(1) Design Basis and Operation Issues	Fatigue(1) Operation Issues	Stress Relaxation
RCL Injection Nozzles	N	N	N	N	N	N	N	I-RA	I-RA	N
6-Inch Valve Transitions	N	N	N	N	N	N	N	I-RA	I-RA	N
Branch Pipes	N	N	N	N	N	N	N	I-RA	I-RA	N
Long and Short Radius Elbow	N	N	N	N	N	N	N	N	I-RA	N
Butt Welds	N	N	N	N	N	N	N	N	I-RA	N
Straight Pipe	N	N	N	N	N	N	N	N	I-RA	N
6-inch, 8-inch and 10-inch Straight and Reducing Tees	N	N	N	N	N	N	N	I-RA	I-RA	N
Socket Welds	N	N	N	N	N	N	N	I-RA	I-RA	N
Socket Welded Valves	N	N	N	N	N	N	N	I-RA	I-RA	N
Socket Welded Tees	N	N	N	N	N	N	N	I-RA	I-RA	N

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-9
PIPING DEGRADATION EFFECT EVALUATION
HOT LEG SAFETY INJECTION LINE SYSTEM

Components 2-inch and 6-inch Sch 160	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue[1] Design Basis and Operation Issues	Fatigue[1] Operation Issues	Stress Relaxation
RCL Injection Nozzles	N	N	N	N	N	N	N	N	N	N
6-inch Valve Transitions	N	N	N	N	N	N	N	N	N	N
Branch Pipes	N	N	N	N	N	N	N	N	N	N
Long and Short Radius Elbow	N	N	N	N	N	N	N	N	N	N
Butt Welds	N	N	N	N	N	N	N	N	N	N
Straight Pipe	N	N	N	N	N	N	N	N	N	N
Reducers	N	N	N	N	N	N	N	N	N	N
Socket Welds	N	N	N	N	N	N	N	N	N	N
Socket Welded Valves	N	N	N	N	N	N	N	N	N	N

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-10
PIPING DEGRADATION EFFECT EVALUATION
BORON INJECTION TANK INJECTION LINE SYSTEM

Components 1 1/2-Inch & 3-Inch Sch 160	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue(1)		Stress Relaxation
								Design Basis and Operation Issues	Operation Issues	
RCL Injection Nozzles	N	N	N	N	N	N	N	I-RA	I-RA	N
3-Inch Valve Transitions	N	N	N	N	N	N	N	I-RA	I-RA	N
Branch Pipes	N	N	N	N	N	N	N	I-RA	I-RA	N
Long and Short Radius Elbow	N	N	N	N	N	N	N	N	I-RA	N
Butt Welds	N	N	N	N	N	N	N	N	I-RA	N
Straight Pipe	N	N	N	N	N	N	N	N	I-RA	N
Reducers	N	N	N	N	N	N	N	I-RA	I-RA	N
Socket Welds	N	N	N	N	N	N	N	N	I-RA	N
Socket Welded Valves	N	N	N	N	N	N	N	N	I-RA	N
Socket Welded Tees and Components	N	N	N	N	N	N	N	I-RA	I-RA	N

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-11
PIPING DEGRADATION EFFECT EVALUATION
RESIDUAL HEAT REMOVAL LINE SYSTEM

Components 6-inch Sch 160 10-inch & 12-inch Sch 140	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue[1]		Stress Relaxation
								Design Basis and Operation Issues	Operation Issues	
RCL Nozzles	N	N	N	N	N	N	N	N	I-RA	N
10-inch & 12-inch Valve Transitions	N	N	N	N	N	N	N	N	I-RA	N
Branch Pipes	N	N	N	N	N	N	N	N	I-RA	N
Long and Short Radius Elbow	N	N	N	N	N	N	N	N	I-RA	N
Butt Welds	N	N	N	N	N	N	N	N	I-RA	N
Straight Pipe	N	N	N	N	N	N	N	N	I-RA	N
Straight and Reducing Tees	N	N	N	N	N	N	N	I-RA	I-RA	N

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-12
PIPING DEGRADATION EFFECT EVALUATION
CHARGING LINE SYSTEM

Components 3-Inch and 4-Inch Sch 160	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue(1)		Stress Relaxation
								Design Basis and Operation Issues	Operation Issues	
RCL Nozzles	N	N	N	N	N	N	N	I-RA	I-RA	N
3-inch Valve Transitions	N	N	N	N	N	N	N	I-RA	I-RA	N
Branch Pipes	N	N	N	N	N	N	N	I-RA	I-RA	N
Long and Short Radius Elbow	N	N	N	N	N	N	N	N	I-RA	N
Butt Welds	N	N	N	N	N	N	N	N	I-RA	N
Straight Pipe	N	N	N	N	N	N	N	N	I-RA	N

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, and nonstandard components, super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-13
PIPING DEGRADATION EFFECT EVALUATION
NORMAL LETDOWN LINE SYSTEM

Components 2-inch and 3-inch Sch 160	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue[1]		Stress Relaxation
								Design Basis and Operation Issues	Operation Issues	
RCL Nozzles	N	N	N	N	N	N	N	N	N	N
3-inch Valve Transitions	N	N	N	N	N	N	N	N	N	N
Branch Pipes	N	N	N	N	N	N	N	N	N	N
Large Radius Elbow	N	N	N	N	N	N	N	N	N	N
Butt Welds	N	N	N	N	N	N	N	N	N	N
Straight Pipe	N	N	N	N	N	N	N	N	N	N
Straight and Reducing Tee	N	N	N	N	N	N	N	N	N	N
Socket Welds	N	N	N	N	N	N	N	N	N	N
Socket Welded Valves	N	N	N	N	N	N	N	N	N	N
Socket Welded Tees and Components	N	N	N	N	N	N	N	N	N	N

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-14
PIPING DEGRADATION EFFECT EVALUATION
EXCESS LETDOWN / DRAIN LINE SYSTEM

Components 1-inch and 2-inch Sch 160	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue(1) Design Basis and Operation Issues	Fatigue(1) Operation Issues	Stress Relaxation
RCL Nozzles	N	N	N	N	N	N	N	N	N	N
Branch Pipes	N	N	N	N	N	N	N	N	N	N
Large Radius Elbow	N	N	N	N	N	N	N	N	N	N
Butt Welds	N	N	N	N	N	N	N	N	N	N
Socket Welds	N	N	N	N	N	N	N	I-RA	I-RA	N
2-inch Socket Welded Valve Transitions	N	N	N	N	N	N	N	I-RA	I-RA	N
Straight Pipe	N	N	N	N	N	N	N	N	N	N
Tee	N	N	N	N	N	N	N	N	N	N
1-inch Pipe	N	N	N	N	N	N	N	N	N	N
1-inch Components	N	N	N	N	N	N	N	N	N	N

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, and nonstandard components, super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-15
PIPING DEGRADATION EFFECT EVALUATION
RESISTANCE TEMPERATURE DETECTOR LINE SYSTEM

Components 1-inch, 2-inch & 3-inch Sch 160	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue(1)		Stress Relaxation
								Design Basis and Operation Issues	Operation Issues	
RCL Nozzles	N	N	N	N	N	N	N	I-RA	N	N
Socket Welds	N	N	N	N	N	N	N	I-RA	N	N
2-inch Socket Welded Valve Transitions	N	N	N	N	N	N	N	I-RA	N	N
Branch Pipes	N	N	N	N	N	N	N	I-RA	N	N
Large Radius Elbow	N	N	N	N	N	N	N	N	N	N
Butt Welds	N	N	N	N	N	N	N	N	N	N
Straight Pipe	N	N	N	N	N	N	N	N	N	N
Reducers	N	N	N	N	N	N	N	N	N	N
Straight and Reducing Tees	N	N	N	N	N	N	N	I-RA	N	N
1-inch Pipe	N	N	N	N	N	N	N	N	N	N
1-inch Components	N	N	N	N	N	N	N	N	N	N
2-inch Socket Welded flange and 3-inch Butt Welded Flange	N	N	I-M*	N	I-M	N	N	I-M	N	I-M

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-16
PIPING DEGRADATION EFFECT EVALUATION
SEAL WATER INJECTION LINE SYSTEM

Components 2-inch & 3-inch Sch 160	Radiation Effects	Thermal Aging	General Corrosion	SCC	Wear	Erosion	Creep	Fatigue[1] Design Basis and Operation Issues	Fatigue[1] Operation Issues	Stress Relaxation
RCL Nozzles	N	N	N	I-M	N	N	N	N	N	N
Socket Welds	N	N	N	I-M	N	N	N	N	N	N
2-inch Socket Welded Valve Transitions	N	N	N	I-M	N	N	N	N	N	N
Branch Pipes	N	N	N	I-M	N	N	N	N	N	N
Large Radius Elbow	N	N	N	N	N	N	N	N	N	N
Butt Welds	N	N	N	I-M	N	N	N	N	N	N
Straight Pipe	N	N	N	N	N	N	N	N	N	N
Tee	N	N	N	I-M	N	N	N	N	N	N

Notes:

N = Not an issue
I-M = Issue but manageable
I-RA = Issue requiring AMP

1. The list of fatigue-sensitive items identified as I-RA could be reduced if generic fatigue evaluations were performed. Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 3-17
COMPONENT/EFFECT DISPOSITION SUMMARY FOR EACH STRUCTURAL ASSEMBLY GROUPING

Key: No Significant Degradation
☐
Effective Program
☒
Plant-Specific Management
■

Class 1 Piping and Associated Pressure Boundary Components	Fatigue	Corrosion		Irrad. Embrit.	Thermal Aging	Erosion	Wear	Creep	
		General Corrosion	SCC					Creep	Stress Relaxation
Class 1 Piping(2)									
Forged	■☐(2)	☐	☐	☐	☐	☐	☐	☐	☐
Cast	■☐(2)	☐	☐	☐	■	☐	☐	☐	☐
Flange	■☐(2)	☒(1)	☐	☐	☐	☐	☒	☐	☐
Flange Bolts(5)	■☐(2)	☒	☐	☐	☐	☐	☒	☐	☒
RCP									
Casings	☐	☒(1)	☐	☐	■	☐	☒	☐	☐
Closure(3)	■	☒(1)	☐	☐	☐	☐	☒	☐	☐
Closure Bolting	☐	☒	☐	☐	☐	☐	☒	☐	☒
Class 1 Valves									
Forged Bodies	■☐(6)	☒(1)	☐	☐	☐	☐	☒	☐	☐
Cast Bodies	■☐(6)	☒(1)	☐	☐	■	☐	☒	☐	☐
Closure(4)	☐	☒(1)	☐	☐	☐	☐	☒	☐	☐
Closure Bolting	☐	☒	☐	☐	☐	☐	☒	☐	☒

Notes:

1. External surfaces near bolted closures may be susceptible to general corrosion due to leakage.
2. The fatigue significance of the Class 1 piping and piping components is summarized by system in Tables 3-2 through 3-16.
3. The RCP closure components consist of the thermal barrier flange, and, depending on the model, the main closure flange (applicable to models M93 and M93A), the bolting ring (applicable to models M93A-1 and M100), and the diffuser flange (applicable to model M100). In addition, the RCP Class 1 auxiliary nozzles for injection and cooling water are considered to be fatigue-sensitive.
4. The Class 1 valve closure consists of the bonnets.
5. Class 1 piping flange and bolts are used on the safety valve and RTD bypass lines.
6. Class 1 valve sizes greater than 4-inch NPS are categorized as fatigue-sensitive and require a fatigue evaluation.
   Class 1 valve sizes 4-inch NPS and less do not require a fatigue evaluation.

4.0 AGING MANAGEMENT ACTIVITIES AND PROGRAM ATTRIBUTES

This section provides options to manage aging effects during an extended period of operation. Since this report is generically applicable, only program attributes are given. Plant-specific details will be developed during the preparation of license renewal applications. The plant-specific programs developed by utilities will demonstrate that aging effects are managed. Therefore, the Class 1 piping and associated components intended function will be maintained during an extended period of operation.

Section 3.0 identifies aging effects that require management during an extended period of operation. Section 4.1 provides current industry practices, and Section 4.2 provides additional activities and attributes, including time-limited aging analyses (TLAAs), required to manage aging effects.

Details and implementation guidance are provided. Deviations from the attributes provided will require descriptions and justifications in plant-specific license renewal applications. Aging management attributes are summarized by aging management program (AMP) tables (see Table 4-1 for a description of AMP attributes). These tables summarize program attributes and activities that form the basis for programs implemented by utilities during an extended period of operation. All six attributes may not be necessary for a program.

A license renewal applicant intending to take credit for the effective program is responsible for the review/evaluation of their related plant-specific features, including appropriate CLB documents/information, to ensure that the program attributes used to manage the aging effects are committed for use at their plant. Programs to manage aging effects that are not part of this report will require plant-specific evaluations, analyses, and justifications.

On the basis of the reviews provided in Section 3.0 of this report, seven aging effects were resolved relative to license renewal considerations. The component aging effects were determined to be nonsignificant because: (1) either the component was not susceptible to the aging effect under consideration or was susceptible to such a small degree that the component's intended function would be maintained throughout the license renewal period; or (2) current established regulations, tests, inspections, and analytical procedures (acceptable programs) were able to manage the aging effect so that the component's intended function would be maintained throughout the license renewal term.

Aging effects caused by fatigue and thermal aging will require additional activities and attributes.

TABLE 4-1
AGING MANAGEMENT PROGRAM ATTRIBUTES

Attribute	Description
Scope	Structures, components, or subcomponents and applicable aging effects.
Surveillance Techniques	Monitoring, inspection, and testing techniques used to detect aging effects.
Frequency	Time period between program performance or when a one-time inspection must be completed. Programs for event-driven effects should perform periodic inspections for the event. Inspection for the effect will take place when an event has occurred.
Acceptance Criteria	Qualitative or quantitative criteria that determine when corrective actions are required.
Corrective Actions	Actions to further analyze, prevent, or correct the consequences of the effect. Corrective actions should include evaluation of failures to determine where similar aging effects may occur and actions, if practical, to mitigate or eliminate the effect from occurring.
Confirmation	Post-maintenance test or other techniques to confirm that the actions have been completed and are effective.

4.1 CURRENT ACTIVITIES AND PROGRAM ATTRIBUTES

Inspection activities being performed and maintenance management programs being pursued to meet current licensing and industry issue requirements need to continue. It is not necessary to modify existing maintenance and inspection programs for the effects addressed in this section because the aging management reviews for license renewal have not resulted in any new requirements for utilities. The options provided in this section should already be part of a utility maintenance program that follows ASME inspection and examination requirements for the nuclear facility. A utility should provide the basis for deviation during an extended period of operation if:

- Their aging management activities are different from the methods given in this report.
- Their plant falls outside the parameter ranges that bound this report.
- The procedures required to address industry issues are not followed.

Maintenance programs follow the ASME Code recommendations. The regulations and rules that govern the inspection of Class 1 piping and associated pressure boundary components begin at the top level with the Code of Federal Regulations. Document 10 CFR 50.55a references ASME Code, Section XI. Requirements are given in the following ASME, Section XI, 1989 edition, subsections for the Class 1 piping and associated pressure boundary components that are within the scope of this evaluation:

- IWB-1000, "Scope and Responsibility"
- IWB-2000, "Examination and Inspection"
- IWB-3000, "Acceptance Standards"
- IWB-4000, "Repair Procedures"
- IWB-7000, "Replacements"

Industry issues that are potentially significant during an extended period of operation have been identified in Section 3.1. Issue resolution has included development of program attributes to manage the issues. A utility determines the significance of each industry issue to their plant(s) to determine if the program should be implemented.

Technical issues are addressed through specific actions required by U.S. NRC directives. The programs are followed by utilities during the current term. For example, if the current utility commitments in response to an industry issue are adequate to manage aging, a utility can extend their existing commitments into an extended period of operation, as required by 10 CFR 54.33(c). If a utility decides to modify this commitment, a utility must address this in the plant-specific license renewal application.

Maintenance programs to manage the potentially significant effects for the wear of closures, and stress relaxation of bolts are given in the following sections. A license renewal applicant intending to take credit for these effective programs is responsible for the review/evaluation of plant-specific documentation, to ensure that the program elements required to manage these aging effects, or their justified equivalent, are committed for use at their plant. All records generated by corrective actions and inspections shall be maintained, as defined by Reference 41.

4.1.1 Aging Management Program for Wear of Closures (AMP-3.1)

Mechanical wear affects reactor coolant pump (RCP) and Class 1 valve bolted closure elements, such as closure flanges and bolting, due to relative motion caused by loss of bolt preload or by infrequent disassembly and reassembly.

The effects of such potential degradation are managed by current programs of periodic inservice inspection and testing, in accordance with the requirements of ASME Code, Section XI, Subsection IWB, and ASME/ANSI OM standards, Parts 1 through 10. The aging management program attributes for the wear of closures are shown in Table 4-2.

TABLE 4-2
AGING MANAGEMENT ACTIVITIES AND PROGRAM ATTRIBUTES - AMP-3.1 FOR WEAR OF CLOSURES

Attribute	Description
Scope	RCP and Class 1 valve closure flanges and bolting.
Surveillance Techniques	• Visual inspection (VT-1) of associated flange surfaces surrounding bolt or stud and for RCP and Class 1 valve bolts and studs 2-inches or less in diameter. • Volumetric inspection for RCP and Class 1 valve bolts and studs greater than 2-inches in diameter. • System leakage and hydrostatic testing with associated visual (VT-2) inspection. • Pump and valve inservice and functional testing to ensure the operability of pressure boundary parts in accordance with Part 1 (Class 1 valves) and Part 6 (pumps) of the ASME/ANSI OM Standard.
Frequency	Each inspection interval of the plant's inservice inspection program or at each refueling outage in the case of system leakage tests.
Acceptance Criteria	Per ASME Code, Section XI and ASME/ANSI OM standard
Corrective Actions	• Repair or refurbish per ASME Code, Section XI • Replace per ASME Code, Section XI
Confirmation	Preserve examinations consisting of: IWA-4700, pressure test following repair by welding, is performed prior to return of the system to service IWB-2420, Successive Inspections IWB-2430, Additional Examinations

Table IWB-2500-1 of Section XI prescribes inservice inspection coverage and frequency for RCP and valve bolted closure parts. For example, Examination Category B-G-1 requires volumetric examination for pump and valve closure bolts and studs greater than 2 inches in diameter and visual (VT-1) examination for the associated flange surfaces. These examinations include a 1-inch wide annular surface of the flange surrounding each bolt or stud. Examination Category B-G-2 provides visual (VT-1) examination requirements only for the pump and valve studs and bolts that are less than 2 inches in diameter. Guidance for the establishment of ultrasonic examination procedures that may not be contained in plant-specific inservice programs is provided by the supplements to mandatory Appendix VIII of the 1989 edition of the ASME Code, Section XI. This component-specific guidance offers details on numbers, sizes, and types of both natural and artificial defects that can be used in the preparation of ultrasonic examination qualification test specimens.

Acceptance criteria for the B-G-1 and B-G-2 periodic visual inservice inspections are provided by the relevant conditions of IWB-3517.1, any detection of which is cause for a determination that the component is unacceptable for continued service, unless justified by supplemental examinations, analytical evaluations, corrective measures or repairs, or component replacement. The relevant conditions for B-G-1 and B-G-2 visual examinations include: (1) degradation of protective coatings on bolt surfaces; and (2) evidence of coolant leakage near bolting.

In addition, Examination Category B-P provides for system leakage and hydrostatic testing, with associated visual (VT-2) inspection, at periodic intervals. Such testing is carried out in conjunction with pump and valve inservice and functional testing, intended to ensure the operability of pressure boundary parts (including internals that comprise a portion of the pressure boundary), in accordance with Part 1 (Class 1 valves) and Part 6 (pumps) of the ASME/ANSI OM standard. Relevant conditions for these VT-2 examinations are provided in IWB-3522.1 and include (1) leakage from noninsulated components; and (2) discoloration or accumulated residue on surfaces of components, insulation, or floor areas that may be evidence of borated water leakage.

These examinations and tests are carried out at each inspection interval of the plant's inservice inspection program or at each refueling outage in the case of system leakage tests. The current practice is expected to remain effective during the license renewal term because the rate of degradation is not expected to change.

Therefore, current and effective programs are adequate to detect and manage the effects of potentially significant mechanical wear through evaluation/repair/replacement for RCP and Class 1 valve closure elements [Ref. 12].

4.1.2 Aging Management Program for Stress Relaxation of Bolts (AMP-3.2)

Loss of preload and subsequent leakage caused by stress relaxation have been identified in Section 3.0 as a potentially significant aging effects for RCP and Class 1 valve closure bolting. Periodic inservice inspections, in accordance with the ASME Code, Section XI, Subsection IWB are capable of managing these effects during the license renewal term. The aging management program attributes for stress relaxation of RCP and Class 1 valve bolts are shown in Table 4-3. Table IWB-2500-1 provides the coverage and frequency of examination requirements for the bolts of concern. For example, Examination Category B-G-1 requires volumetric examination for pump and valve closure bolts and studs greater than 2 inches in diameter and visual (VT-1) examination for the associated flange surfaces. These examinations include a 1-inch wide annular surface of the flange surrounding each bolt or stud. Examination Category B-G-2 provides visual (VT-1) examination requirements only for pump and valve bolts and studs that are less than 2 inches in diameter.

Guidance for the establishment of ultrasonic examination procedures that may not be contained in plant-specific inservice programs is provided by the supplements to mandatory Appendix VIII of the 1989 edition of the ASME Code, Section XI. This component-specific guidance offers details on numbers, sizes, and types of both natural and artificial defects that can be used in the preparation of ultrasonic examination qualification test specimens. Acceptance criteria for the B-G-1 and B-G-2 periodic visual inservice inspections are provided by the relevant conditions of IWB-3517.1, any detection of which is cause for a determination that the component is unacceptable for continued service, unless justified by supplemental examinations, analytical evaluations, corrective measures or repairs, or component replacement. The relevant conditions for B-G-1 and B-G-2 visual examinations include: (1) missing or loose bolts, studs, nuts, or washers; and (2) fractured bolts, studs, or nuts. In addition, Examination Category B-P of Subsection IWB provides for visual (VT-2) examination associated with system leakage and hydrostatic testing. Relevant conditions for these VT-2 examinations are provided in IWB-3522.1 and include: (1) leakage from noninsulated components; (2) leakage in excess of permissible levels defined by the owner from components provided with leakage limiting devices; (3) leakage from insulated components or inaccessible components that will require location of the leakage source; and (4) discoloration or accumulated residue on surfaces of components, insulation, or floor areas that may be evidence of borated water leakage. Corrective measures are included in Subsection IWA-5250 for detected leakage at bolted location, in accordance with the acceptance criteria of IWB-3142. The current practice is expected to remain effective during the license renewal term because the rate of degradation is not expected to change.

These VT-2 visual inservice examinations are supplemented by plant commitments in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." These commitments address the potential for primary coolant leak rates less than Technical Specification limits, especially at locations where such leaks could potentially affect the integrity of the pressure boundary by boric acid corrosion.

Therefore, current programs are adequate to manage potentially significant effects of stress relaxation (loss of preload and subsequent leakage) for RCP and Class 1 valve closure bolting.

TABLE 4-3
AGING MANAGEMENT ACTIVITIES AND PROGRAM ATTRIBUTES - AMP-3.2 FOR STRESS RELAXATION OF BOLTS

Attribute	Description
Scope	RCP and Class 1 valve closure bolting
Surveillance Techniques	• Visual inspection (VT-1) of bolting surfaces • System leakage and hydrostatic testing with associated visual (VT-2) inspection
Frequency	Each inspection interval of the plant's inservice inspection program or at each refueling outage in the case of system leakage tests.
Acceptance Criteria	Per ASME Code, Section XI
Corrective Actions	• Repair or refurbish per ASME Code, Section XI • Replace per ASME Code, Section XI
Confirmation	Preservice examinations consisting of: • IWA-4700, pressure test following repair by welding, is performed prior to return of the system to service • IWB-2420, Successive Inspections • IWB-2430, Additional Examinations

4.2 ADDITIONAL ACTIVITIES AND PROGRAM ATTRIBUTES

This section describes the aging management activities and program attributes (AMAPA) for fatigue and thermal aging.

4.2.1 Aging Management Program for Fatigue (AMP-3.3 to AMP-3.5)

This discussion covers fatigue-sensitive components, including components designed to ANSI B31.1 as well as components designed to ASME Code, Section III fatigue requirements. Fatigue is defined as the structural deterioration that can occur as a result of periodic application of load or stress by mechanical, thermal, or combined effects. Specific effects of fatigue are cracks in the material that may or may not be detected before mechanical failure. After repeated cyclic loading of sufficient magnitude, microstructural damage can accumulate, leading to macroscopic crack initiation at the most affected locations. Subsequent mechanical or thermal cyclic loading can lead to growth of the initiated crack.

Acceptable aging management options for fatigue will be dependent on the final NRC position on fatigue in license renewal. In this report, the attributes for a fatigue aging management program are presented based on a number of options. One alternative providing a number of options is the EPRI Proposed Industry Position on Fatigue Evaluation for License Renewal [Ref. 42], which is currently being developed. This source provides a broad-based, general approach to fatigue management to which the specifics of Class 1 piping and associated components may be applied. Figures 4-1 and 4-2 show the overall flowchart presented by the current industry position for ASME and B31.1 designs, respectively. Using these flow charts as a basis, the following paragraphs describe the characteristics of a general application of the proposed industry position to Class 1 piping and associated components.

The objectives of the fatigue management program are to:

(1) Maintain the CLB for fatigue for the current and license renewal terms by justifying that existing fatigue analyses are valid or extending the period of evaluation of the analyses so that they are valid.

or

(2) Justify that the effects of fatigue will be adequately managed for the license renewal term if the applicant cannot or chooses not to justify or extend the existing fatigue analyses.

For Class 1 piping and associated components, the CLB includes:

- Fatigue design basis: ASME Code, Section III Class 1 explicit fatigue design or B31.1 Code fatigue strength reduction factor design
- Fatigue operating basis: Cyclic duty commitments and ASME Code, Section XI ISI commitments
- Regulatory oversight process commitments

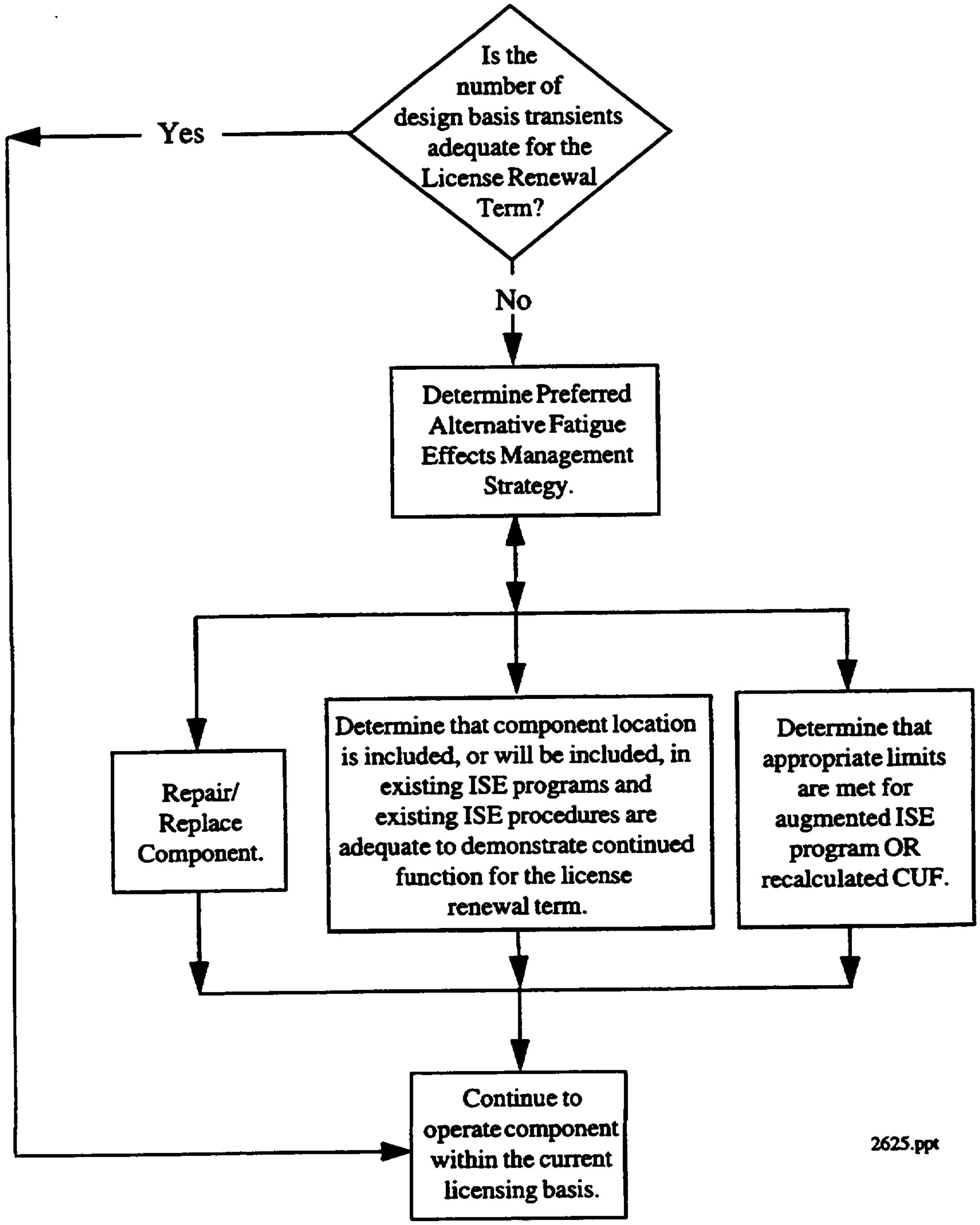

Figure 4-1 Flow Chart for Fatigue, AMP-3.3 (for ASME designs) [Ref. 42]

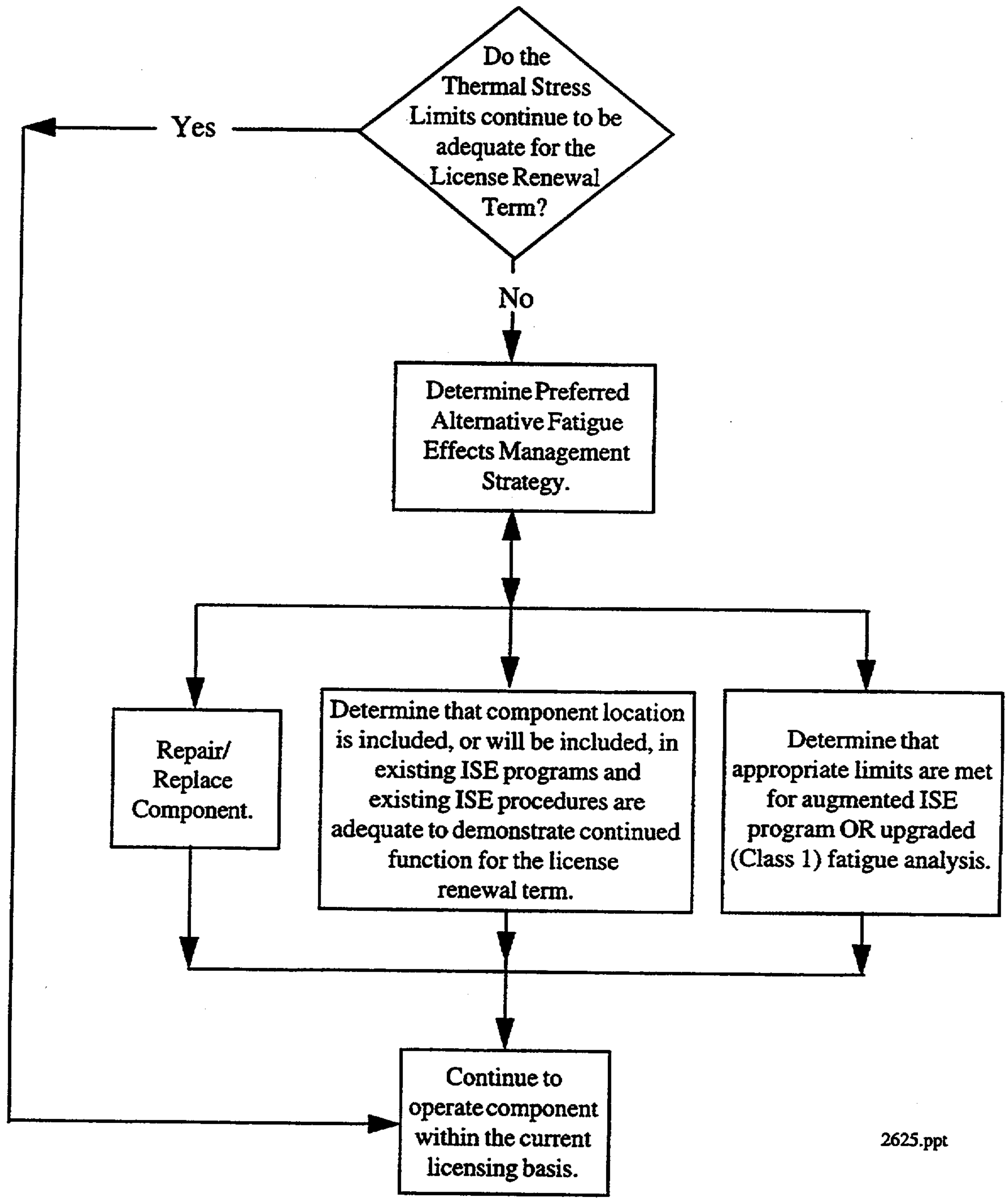

Figure 4-2 Flow Chart for Fatigue, AMP-3.4 and AMP-3.5 (for B31.1 designs) [Ref. 42]

In Section 3.0, fatigue was evaluated for various components (see Tables 3-2 through 3-17). Based on this review, components that had relatively high usage factors were identified as fatigue-sensitive. The fatigue-sensitive components identified in Tables 3-2 through 3-17 are summarized in Table 4-4. The large number of fatigue-sensitive components is based on an envelope that includes conservative design considerations (see Section 3.4). This list could be reduced if fatigue re-evaluations were performed or if data from fatigue monitoring programs was used. Re-evaluations can take advantage of inherent conservatisms in the ASME Code procedures and in the definition of the fatigue design basis transients. A key aspect of the fatigue re-evaluation is the use of a statistical distribution of stress amplitude peaks for some loading cycles, reflecting actual operating transient data. Another consideration is the evaluation of high-cycle fatigue prior to the inclusion in the 1983 ASME Code of design fatigue curves for numbers of loading cycles beyond 10^6.

The industry process is four alternatives, or steps, that are applied as follows for fatigue-sensitive locations. The corresponding AMPs, which summarize the options that manage aging effects for fatigue-sensitive locations, are provided in Tables 4-5 through 4-7.

Step 1. Determine if the current and projected transients for the license renewal term are within the CLB. For fatigue-sensitive locations, step 1 has two approaches that address ASME and B31.1 designs. The remaining steps are applicable to both ASME and B31.1 designs. Generic fatigue analyses can be generated on the Class 1 valve bodies, RCP casings, and certain piping and piping components for license renewal.

Step 1A, which is applicable for ASME, Section III, B31.7, and Draft Pump and Valve Code designs, is to assess the adequacy of CLB transient cycles for the current and license renewal terms. The notes to the first step in the position provide for virtually any justifiable manner of comparison to the CLB transients, from simple event-related cycle identification to partial cycle counting or fatigue usage recalculation. Subsections 4.2.1.1 and 4.2.1.2 provide options: (1) for methods to address cyclic adequacy of CLB, ranging from simple manual cycle counting to reclassification of plant transients and recalculation of fatigue usage and (2) to qualify fatigue for B31.1 designs. In this process, it is important that the fatigue operating basis transients are used for comparison. For Class 1 piping and associated components, this includes:

- Original design transients
- Evaluated plant-specific "off-normal" transients possibly including NRCB 88-08 reconstitution of other piping load transients
- NRCB 88-11 reconstitution of surge line piping load transients

TABLE 4-4
ENVELOPE OF FATIGUE-SENSITIVE ITEMS FOR CLASS 1 PIPING AND COMPONENTS

Component	Subcomponent	Fatigue	
		Design Basis and Operation Issues	Operation Issues
Reactor Coolant Loop	RPV Inlet Nozzle Safe End	I-RA	N
	RPV Outlet Nozzle Safe End	I-RA	I-M
	SG Inlet Nozzle Safe End	I-RA	I-M
	SG Outlet Nozzle Safe End	I-RA	N
	RCP Closure[1]	I-RA	N
Pressurizer Surge Line (12-inch and 14-inch Sch 140)	Pressurizer Nozzle	I-RA	I-RA
	Thermowell Boss	I-RA	I-RA
	Large Radius Elbow	I-RA	I-RA
	Butt Weld	I-M	I-M
	Welded Attachments	I-RA	I-RA
	RCL Branch Nozzle	I-RA	I-RA
Pressurizer Spray Line (6-inch and 4-inch Sch 160)	Pressurizer Nozzle	I-RA	I-RA
	Branch Pipes	I-RA	I-RA
	Large Radius Bend	I-RA	N
	Butt Welds	I-RA	N
	Welded Attachments	I-RA	N
	6-inch x 4-inch Nonstandard Reducing Tee	I-RA	N
	4-inch x 3/4-inch Branch Pipes	I-RA	N
	6-inch x 2-inch Branch Pipes	I-RA	I-RA
	Steam Filled Butt Welds, Tees, Transitions, and Branches	I-RA	I-RA

N = Not an issue

I-M = Issue but manageable if additional calculations are performed to show the component acceptable

I-RA = Issue requiring AMP or generic fatigue evaluations

Note: Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

1. The Class 1 RCP closure components consist of the thermal barrier flange, main closure flange (applicable to Models M93 and M93A), the bolting ring (applicable to Models M93A-1 and M100) and the diffuser flange (applicable to Model M100). In addition, the RCP Class 1 auxiliary nozzles for injection and cooling water are considered to be fatigue-sensitive.

TABLE 4-4 (Continued)
ENVELOPE OF FATIGUE-SENSITIVE ITEMS FOR CLASS 1 PIPING AND COMPONENTS

Component	Subcomponent	Fatigue: Design Basis and Operation Issues	Fatigue: Operation Issues
Auxiliary Pressurizer Spray Line System (2-inch Sch 160)	Socket Welds	I-RA	I-RA
	2-inch Socket Welded Valve Transitions	I-RA	I-RA
	Large Discontinuity Branch Pipes	I-RA	I-RA
	Branch Pipes	N	I-RA
	Socket Welded Elbow	N	I-RA
	Butt Welds	N	I-RA
	Straight Pipe	N	I-RA
	Welded Attachments	I-RA	I-RA
	Straight and Reducing Tees	I-RA	I-RA
Pressurizer Safety & Relief Line (6-inch and 3-inch Sch 160)	Pressurizer Nozzles	I-RA	N
	3-inch Valve Transitions	I-RA	N
	Branch Pipes	I-RA	N
	6-inch Flange & Bolts	I-RA	N
	6-inch x 3-inch Reducers	I-RA	N
Accumulator Injection Line (10-inch and 12-inch Sch 140 & 160)	45° Accumulator Injection Nozzles	I-RA	N
	10-inch & 12-inch Valve Transitions	I-RA	N
	Branch Pipes	I-RA	N
	6-inch & 8-inch Straight & Reducing Tees	I-RA	N

N = Not an issue
I-M = Issue but manageable if additional calculations are performed to show the component acceptable
I-RA = Issue requiring AMP or generic fatigue evaluations

Note: Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 4-4 (Continued)
ENVELOPE OF FATIGUE-SENSITIVE ITEMS FOR CLASS 1 PIPING AND COMPONENTS

Component	Subcomponent	Fatigue	
		Design Basis and Operation Issues	Operation Issues
Cold Leg Safety Injection Line (2-inch, 6-inch, 8-inch, and 10-inch Sch 160)	RCL Injection Nozzles	I-RA	I-RA
	6-inch Valve Transitions	I-RA	I-RA
	Branch Pipes	I-RA	I-RA
	Long & Short Radius Elbows	N	I-RA
	Butt Welds	N	I-RA
	Straight Pipe	N	I-RA
	6-inch, 8-inch, and 10-inch Straight & Reducing Tees	I-RA	I-RA
	Socket Welds	I-RA	I-RA
	Socket Welded Valves	I-RA	I-RA
	Socket Welded Tees	I-RA	I-RA
BIT Injection Line (1 1/2-inch and 3-inch Sch 160)	RCL Injection Nozzles	I-RA	I-RA
	3-inch Valve Transitions	I-RA	I-RA
	Branch Pipes	I-RA	I-RA
	Long & Short Radius Elbows	N	I-RA
	Butt Welds	N	I-RA
	Straight Pipe	N	I-RA
	Reducers	I-RA	I-RA
	Socket Welds	N	I-RA
	Socket Welded Valves	N	I-RA
	Socket Welded Tees & Components	I-RA	I-RA

N = Not an issue
I-M = Issue but manageable if additional calculations are performed to show the component acceptable
I-RA - Issue requiring AMP or generic fatigue evaluations

Note: Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 4-4 (Continued)
ENVELOPE OF FATIGUE-SENSITIVE ITEMS FOR CLASS 1 PIPING AND COMPONENTS

Component	Subcomponent	Fatigue: Design Basis and Operation Issues	Fatigue: Operation Issues
RHR Line (6-inch Sch 160 and 10-inch and 12-inch Sch 140)	RCL Nozzles	N	I-RA
	10-inch & 12-inch Valve Transitions	N	I-RA
	Branch Pipes	N	I-RA
	Long & Short Radius Elbows	N	I-RA
	Butt Welds	N	I-RA
	Straight Pipe	N	I-RA
	Straight & Reducing Tees	I-RA	I-RA
Charging Line (3-inch and 4-inch Sch 160)	RCL Nozzles	I-RA	I-RA
	3-inch Valve Transitions	I-RA	I-RA
	Branch Pipes	I-RA	I-RA
	Long & Short Radius Elbows	N	I-RA
	Butt Welds	N	I-RA
	Straight Pipe	N	I-RA
Excess Letdown / Drain Line (1-inch and 2-inch Sch 160)	Socket Welds	I-RA	I-RA
	2-inch Socket Welded Valve Transitions	I-RA	I-RA
RTD Line (1-inch, 2-inch, and 3-inch Sch 160)	RCL Nozzles	I-RA	N
	Socket Welds	I-RA	N
	2-inch Socket Welded Valve Transitions	I-RA	N
	Branch Pipes	I-RA	N
	Straight & Reducing Tees	I-RA	N
	2-inch Socket Welded Flange and 3-inch Butt Welded Flange	I-M	N
All Class 1 lines	6-inch & Larger Valve Bodies	I-RA	N

N = Not an issue
I-M = Issue but manageable if additional calculations are performed to show the component acceptable
I-RA = Issue requiring AMP or generic fatigue evaluations

Note: Trunnions, lugs, nonstandard components, and super-stiff clamps must be evaluated individually for fatigue.

TABLE 4-5
AGING MANAGEMENT ACTIVITIES AND PROGRAM ATTRIBUTES – AMP-3.3 FOR FATIGUE

<table>
<tr><th>Attribute</th><th colspan="2">Description</th></tr>
<tr><td>Scope</td><td><u>Component</u>
ASME Class 1 piping valve bodies 6-inches and larger, and RCP closure fatigue-sensitive locations
(see Table 4-4)</td><td><u>Effect</u>
Crack initiation in pressure-retaining elements due to cyclic loading
• Potential fluid loss caused by through-wall cracking
• Reduced structural capacity caused by crack stress concentration</td></tr>
<tr><td>Surveillance Technique</td><td colspan="2">1. Evaluation of CLB transients versus previous and anticipated transients, including:
• Comparison showing CLB transients envelop the current and anticipated license renewal term transients;
• Reclassify actual transients to meet cyclic duty commitments;
• Use of actual temperature/strain measurements to show cyclic duty is met; or
• Remove conservatism from the design basis fractions and recalculate CUF
(See Subsection 4.2.1.1)
<u>OR</u>
2. Qualify ISI procedures to adequately detect and size flaws. Examine fatigue-sensitive location per Subsection IWB, Requirements For Class 1 Components of Light-Water Cooled Power Plants (Examination Category B-F, Volumetric & Surface; Category B-J, Volumetric & Surface; Category B-K-1, Volumetric or Surface; Category B-M-1, Volumetric; Category B-M-2, Visual VT-3; Category B-P, Visual VT-2),
• IWB-2500, Examination and Pressure Test Requirements and Table IWB-2500-1
<u>OR</u>
3 Account for all significant fatigue loads by recalculating the fatigue analysis
<u>OR</u>
4. Evaluate location using LBB methodology
<u>OR</u>
5. Perform flaw tolerance evaluation with local inspections</td></tr>
<tr><td>Frequency</td><td colspan="2">1. One-time evaluation for components showing acceptable usage for current and license renewal terms or continuous transient cycle monitoring for components requiring fatigue usage calculation for actual plant transients.
2. Inspection: IWB-2410, Inspection Program - Table IWB-2412-1
• Follow Inspection Program B, 1st Interval, 10 year inspection plan
3. One-time qualification
4. One-time qualification
5. Follow inspection interval based on flaw tolerance evaluation</td></tr>
<tr><td>Acceptance Criteria</td><td colspan="2">1. The number and classification of CLB transients envelopes those associated with the license renewal term, or
CUF < 1.0 (unless justified otherwise)
2. IWB-3410, Acceptance Standards - Table IWB 3410-1 Acceptance Standards
3. CUF < 1.0 (unless justified otherwise)
4. Margin > 2 between leakage flaw size and critical flaw size, and
Margin > 10 between calculated leak rate and leak detection capability
5. Per ASME, Section XI nonmandatory appendix criteria</td></tr>
</table>

TABLE 4-5 (Continued)
AGING MANAGEMENT ACTIVITIES AND PROGRAM ATTRIBUTES – AMP-3.3 FOR FATIGUE

Attribute	Description
Corrective Actions	1. Replace per IWB-7000 rules 2. Replace per IWB-7000 rules 3. Cycle monitoring/counting and replace per IWB-7000 rules 4. Replace per IWB-7000 rules 5. Repair per IWB-4000 rules or replace per IWB-7000 rules
Confirmation	Preservice examinations consisting of: • IWA-4700, pressure test following repair by welding, is performed prior to return of the system to service • IWB-2420, Successive Inspections • IWB-2430, Additional Examinations

TABLE 4-6
AGING MANAGEMENT ACTIVITIES AND PROGRAM ATTRIBUTES – AMP-3.4 FOR FATIGUE

<table>
<tr><th>Attribute</th><th colspan="2">Description</th></tr>
<tr><td rowspan="2">Scope</td><td><u>Component</u></td><td><u>Effect</u></td></tr>
<tr><td>B31.1 Class 1 pipe fatigue-sensitive locations (see Table 4-4)</td><td>Crack initiation in pressure-retaining elements due to cyclic loading
• Potential fluid loss caused by through-wall cracking
• Reduced structural capacity caused by crack stress concentration</td></tr>
<tr><td>Surveillance Technique</td><td colspan="2">1. Evaluation of CLB transients using fatigue strength reduction factor methodology with additional criteria to address: 1) combined geometric and loading discontinuities, 2) high stress concentrations, and 3) significant through-wall temperature gradients. (see Subsection 4.2.1.2)
<u>OR</u>
2. Qualify ISI procedures to adequately detect and size flaws. Examine fatigue-sensitive location per Subsection IWB, Requirements For Class 1 Components of Light-Water Cooled Power Plants (Examination Category B-F, Volumetric & Surface; Category B-J, Volumetric & Surface; Category B-K-1, Volumetric or Surface; Category B-M-1, Volumetric; Category B-M-2, Visual VT-3; Category B-P, Visual VT-2),
• IWB-2500, Examination and Pressure Test Requirements and Table IWB-2500-1
<u>OR</u>
3. Account for all significant fatigue loads by recalculating the fatigue analysis
<u>OR</u>
4. Evaluate location using LBB methodology
<u>OR</u>
5. Perform flaw tolerance evaluation with local inspections</td></tr>
<tr><td>Frequency</td><td colspan="2">1. One-time evaluation
2. Inspection: IWB-2410, Inspection Program - Table IWB-2412-1
– Follow Inspection Program B, 1st Interval, 10 year inspection plan
3. One-time qualification
4. One-time qualification
5. Follow inspection interval based on flaw tolerance evaluation</td></tr>
<tr><td>Acceptance Criteria</td><td colspan="2">1. The number and classification of CLB transients envelopes those associated with the license renewal term, or
CUF < 1.0 (unless justified otherwise)
2. IWB-3410, Acceptance Standards - Table IWB 3410-1 Acceptance Standards
3. CUF < 1.0 (unless justified otherwise)
4. Margin > 2 between leakage flaw size and critical flaw size, and
Margin > 10 between calculated leak rate and leak detection capability
5. Per ASME, Section XI nonmandatory appendix criteria</td></tr>
<tr><td>Corrective Actions</td><td colspan="2">1. Replace per IWB-7000 rules
2. Replace per IWB-7000 rules
3. Cycle monitoring/counting and replace per IWB-7000 rules
4. Replace per IWB-7000 rules
5. Repair per IWB-4000 rules or replace per IWB-7000 rules</td></tr>
</table>

TABLE 4-6 (Continued)
AGING MANAGEMENT ACTIVITIES AND PROGRAM ATTRIBUTES – AMP-3.4 FOR FATIGUE

Attribute	Description
Confirmation	Preservice examinations consisting of • IWA-4700, pressure test following repair by welding, is performed prior to return of the system to service • IWB-2420, Successive Inspections • IWB-2430, Additional Examinations

TABLE 4-7
AGING MANAGEMENT ACTIVITIES AND PROGRAM ATTRIBUTES - AMP-3.5 FOR FATIGUE

<table>
<tr><th>Attribute</th><th colspan="2">Description</th></tr>
<tr><td>Scope</td><td>Component
B31.1 valve bodies 6-inches and larger and RCP closure</td><td>Effect
Crack initiation in pressure-retaining elements due to cyclic loading
• Potential fluid loss caused by through-wall cracking
• Reduced structural capacity caused by crack stress concentration</td></tr>
<tr><td>Surveillance Technique</td><td colspan="2">1. Evaluation of CLB transients using fatigue strength reduction factor methodology with additional criteria to address (see Subsection 4.2.1.2 parts C and D)
OR
2. Qualify ISI procedures to adequately detect and size flaws. Examine fatigue-sensitive location per Subsection IWB, Requirements For Class 1 Components of Light-Water Cooled Power Plants (Examination Category B-K-1, Volumetric or Surface; Category B-L-1, Volumetric; Category B-L-2, Visual VT-3; Category B-P, Visual VT-2),
• IWB-2500, Examination and Pressure Test Requirements and Table IWB-2500-1
OR
3. Account for all significant fatigue loads by recalculating the fatigue analysis
OR
4. Evaluate location using LBB methodology
OR
5. Perform flaw tolerance evaluation with local inspections</td></tr>
<tr><td>Frequency</td><td colspan="2">1. One-time evaluation
2. Inspection: IWB-2410, Inspection Program - Table IWB-2412-1
• Follow Inspection Program B, 1st Interval, 10 year inspection plan
3. One-time qualification
4. One-time qualification
5. Follow inspection interval based on flaw tolerance evaluation</td></tr>
<tr><td>Acceptance Criteria</td><td colspan="2">1. The number and classification of CLB transients envelopes those associated with the license renewal term, or
CUF < 1.0 (unless justified otherwise)
2. IWB-3410, Acceptance Standards - Table IWB 3410-1 Acceptance Standards
3. CUF < 1.0 (unless justified otherwise)
4. Margin > 2 between leakage flaw size and critical flaw size, and
Margin > 10 between calculated leak rate and leak detection capability
5. Per ASME Section XI nonmandatory appendix criteria</td></tr>
<tr><td>Corrective Actions</td><td colspan="2">1. Replace per IWB-7000 rules
2. Replace per IWB-7000 rules
3. Cycle monitoring/counting and replace per IWB-7000 rules
4.
Replace per IWB-7000 rules
5.
Repair per IWB-4000 rules or replace per IWB-7000 rules</td></tr>
</table>

TABLE 4-7 (Continued)
AGING MANAGEMENT ACTIVITIES AND PROGRAM ATTRIBUTES FOR FATIGUE (AMP-3.5)

Attribute	Description
Confirmation	Preservice examinations consisting of: • IWA-4700, pressure test following repair by welding, is performed prior to return of the system to service • IWB-2420, Successive Inspections • IWB-2430, Additional Examinations

The essential goal of this step is to show that the design basis evaluations encompass the effects of fatigue that will be experienced by the component through the end of the license renewal term. The design basis transients are intended to be a conservative estimate of the number, types, and severity of events that can occur in the plant. However, actual operating transients determine the true fatigue damage in components. Operating experience indicates that when a plant is operated by procedures in accordance with the design basis, the actual events are often fewer in number and less severe than postulated by the design transients. Options for methods to address cyclic adequacy of the design basis range from simple manual cycle counting to reclassification of plant transients and recalculation of fatigue usage.

Examples of transient comparison methods to accomplish this first step are:

- Transient cycles - Based on a period of actual plant operations sufficient to characterize operations during the license renewal period, determine a more representative number of total transient cycles for comparison to that assumed in the design basis fatigue evaluation.

- Transient severity - Based on operating experience for a number of plant heatup/cooldown cycles that are representative of operations anticipated during the license renewal period, determine a more representative loading for controlling transients for comparison to those assumed in the design basis fatigue evaluation.

- Transient fatigue effects - Effects of transients determined to be more representative of actual operations during the license renewal period may be compared based on the stress or partial usage effect produced for the subcomponent. Comparisons of this type should also include consideration of existing conservatisms in the design analysis.

Any of these options may also include incorporation of future transient tracking to further reduce conservatisms in the assessment of fatigue effects in a subcomponent.

Successful implementation of this step will adequately manage the effects of fatigue by demonstrating that the CLB fatigue evaluation is valid for the license renewal period, based on the transient loadings considered. The CLB cumulative usage factor acceptance criterion is designed to preclude fatigue cracking, and therefore will demonstrate that the Class 1 piping and associated components intended function will be maintained throughout the license renewal period.

If step 1A is unsuccessful, then go to steps 2, 3, or 4.

Step 1B, which is applicable for B31.1 Code designs, is to assess the thermal stress limits for the license renewal term. Alternatively, actual stresses can be used if available from instrumentation.

The process to consider when qualifying fatigue to B31.1 requirements for the current and license renewal terms is provided in Subsection 4.2.1.2.

If step 1B is unsuccessful, then go to steps 2, 3, or 4.

Step 2. Determine if component is or can be included in existing or enhanced Section XI ISI program with ISI procedures adequate to detect and size flaws that can be shown to not propagate to failure between inspection intervals.

For Class 1 piping and associated components, use the applicable plant-specific Section XI ISI program and include any future risk-based methodologies to the existing ISI program. This requires utilities to follow current Section XI activities on risk-based inspections and regulatory acceptance of those results.

Since manifestation of excessive fatigue damage is expected to be fatigue crack initiation and/or growth, which could ultimately result in a through-wall crack and leakage, the ISI program will detect the effects of significant fatigue damage.

If component is not or cannot be included in an adequate ISI program, then go to steps 1, 3, or 4.

Step 3. Evaluate Class 1 piping and associated components for the license renewal term based on an augmented inspection program or recalculate fatigue usage for reconstituted license renewal transients.

The position notes that augmented ISI may consider flaw tolerance plus local inspection procedures as prescribed by the ASME, Section XI nonmandatory appendix for evaluation of fatigue in operating plants. This option is subject to final regulatory acceptance of the nonmandatory appendix.

The proposed position also notes that risk-based evaluations may be used to determine the frequency and extent of coverage of the augmented inspections. This requires utilities to follow current Section XI activities on risk-based inspections and regulatory acceptance of those results.

Alternatively, a leak-before-break analysis (see Subsection 4.2.2) for the license renewal term may be applied to demonstrate structural integrity similar to the flaw tolerance approach. Similarly, for the RCP casings, a fracture-mechanics-based integrity evaluation in compliance with code case No. 481 may be applied [see Subsection 4.2.2].

The other alternative of the position is to recalculate fatigue usage for license renewal term transients. For Class 1 piping and associated components, conservatisms in the CLB fatigue analyses that may be removed are described in detail in Section 3.3 of this report.

Because of the conservatism in the initial design analysis, the fatigue reanalysis to current criteria is expected to show that the design life objectives can be extended for license renewal term for all "fatigue-sensitive" items, except for the possibility of a few isolated plant-specific items.

The position also notes that the criteria for flaw tolerance, fatigue usage, and leak-before-break recalculations should account for appropriate environmental factors on fatigue crack initiation or growth. This should be done consistent with the criteria yet to be established by the NRC for license renewal. The final criteria may affect the above judgement concerning the acceptability of recalculated fatigue usage for some components. This requires utilities to follow current industry activities to completion and the final NRC position.

Although industry work on environmental effects in fatigue is ongoing (see Subsection 3.1.2), possible potential impact of the issue on Westinghouse Class 1 piping and associated components may be assessed using published results of the status of PVRC activities [Ref. 43]. For carbon and low alloy steels, Reference 43 defines a tentative set of criteria where environmental effects on the S-N fatigue life would be expected to be moderate or acceptable, implying that the ASME Code, Section III fatigue design curves are sufficiently conservative to accommodate moderate environmental effects. These bounding limits are reproduced in Table 4-8. They are presented as independent parameters, so that satisfying any single criterion of the set would indicate that environmental effects are acceptable or moderate.

TABLE 4-8
PVRC VALUES OF INDEPENDENT PARAMETERS FOR ACCEPTABLE OR MODERATE ENVIRONMENTAL EFFECTS ON THE S-N FATIGUE LIFE OF CARBON AND LOW ALLOY STEELS [Ref. 43]

Strain Amplitude	≤ 0.1%
Strain Rate	≥ 0.1%/sec
Dissolved Oxygen	≤ 0.1 ppm
Temperature	≤ 150°C (302°F)
Sulfur in Steel	≤ 0.003%
Water Flow Velocity	> 3 m/sec (3.3 ft/sec)

The surfaces of Class 1 piping and associated components that are in contact with the primary coolant are stainless steel. Reference 43 also discusses environmental effects for austenitic steels and nickel alloys, but criteria have not been developed, primarily due to lack of data. It does show data that generally fall above the ASME design curve for these materials subjected to high oxygen contents (0.2 - 8 ppm) and slow strain rates (0.4 - 0.004 percent/sec). In general, concerns for environmental effects in these materials are apparently not as great as for carbon and low alloy steels. Therefore, if any of the Table 4-8 criteria can be satisfied for Class 1 piping and associated components materials, it is probable that the environmental effects issue will not be significant.

Some parameters represent conditions that must be addressed plant-specifically. Strain amplitude and strain rate are loading-dependent and are not easily addressed quantitatively in the context of this report. However, most actual plant transients would typically result in strain rates and amplitudes within the stated limits. The sulfur content of stainless steel materials may be within the criterion, but is dependent on the material specifications. Water flow velocities in Class 1 piping lines vary and may be within the criterion but are dependent on the fluid systems.

Other parameters represent conditions that may be addressed generally. Dissolved oxygen criteria in Westinghouse PWR primary coolant follow the EPRI Primary Water Chemistry Guidelines [Ref. 44]. These guidelines recommend actions to maintain dissolved oxygen to 5 ppb within a 7-day period and to 100 ppb within a 24-hour period during power operation. They also include limits of less than 100 ppb prior to exceeding 250EF, or prior to criticality. Therefore, it is reasonable to conclude that the PVRC dissolved oxygen criterion would be satisfied for Westinghouse Class 1 piping and associated components.

Based on the general status published by PVRC and the expected PWR water chemistry, it does not appear that environmental effects in fatigue will be a significant issue for Class 1 piping and associated components. As discussed in Subsection 3.1.2, utilities must continue to follow industry and regulatory activities in this area to address the specific parameters for Class 1 piping and associated components.

If step 3 is unsuccessful, then go to steps 1, 2, or 4.

Step 4. Repair/replace component

If step 4 is unsuccessful, then go to steps 1, 2, or 3.

4.2.1.1 Estimating Fatigue-Significant Loads

Fatigue management of Class 1 piping and associated components transients and related analyses for the license renewal period should include establishment of an efficient data collection system for transient cycle counting and component fatigue management. Plant transient data can be used to account for actual versus design transient cycles and severity for use in transient comparisons or reanalysis. In general, acceptable fatigue management programs provide the information necessary to control plant operating and maintenance practices so that critical fatigue degradation is minimized. Examples of this information are:

- Determination of actual loads experienced by the component
- Comparison of actual loads to design assumptions
- Assessment of current structural integrity
- Estimation of future loading
- Assessment of future structural integrity
- Determination to replace, repair, or continue using the component

All records generated by corrective actions and inspections shall be maintained in accordance with plant-specific administrative procedures.

Methods of accounting for transient cycles may vary based on the relative contributions of postulated transients to the actual fatigue accumulation. The extent to which records should be kept for a given transient may be determined by the relative contributions of design transients to fatigue predicted in the fatigue-sensitive subcomponents, and the expected contributions due to transient severity and cyclic activity during actual operation. Development of an effective transient cycle tracking program requires knowledge of the design basis for fatigue-sensitive subcomponents and related plant operating practices.

As part of the design basis for ASME, Section III, transients are defined in terms of their relative severity and number of occurrences. Analytical models of the components are formulated, postulated transient loads are applied, stress time histories are calculated, load combinations are performed, and fatigue accumulation over the plant design life is established. The calculated fatigue values are compared against ASME Code allowable fatigue, which limits the number of occurrences of the design basis transients. In general, plant thermal and pressure design basis transients are the major contributors to fatigue damage in fatigue-sensitive subcomponents.

Most plants have some form of transient cycle counting requirements. Typically, a small subset of the original design transients are tracked. In addition, ANSI/ASME NQA-1, "Quality Assurance Record Keeping for Nuclear Power Plants," advises plants to keep records of cyclic loading for those components designed to undergo a limited number of cycles.

Manual cycle counting practices manifest themselves in the form of periodic manual review of operating history. These reviews identify the transients defined by the Technical Specification tables and add each recognized occurrence to a cumulative log. In most cases, only those transients delineated in the Technical Specification tables are tracked by this method. Consequently, other transients that may be significant with respect to component fatigue and life cycle management are not monitored.

It is important to track all loads that are significant enough to result in material damage. When interpreted as a function of the design transients, all transients that result in some estimated fatigue degradation would have to be included in the set of transients to track. This set would include transients defined by the Technical Specifications, FSAR, and equipment specifications. Tracking of all loading conditions is not necessarily tied to the act of counting cycles but is an accounting of all loads that result in material damage and that may reduce the component life. Hence, programs that ignore loads because they were considered outside the design basis fatigue analysis (emergency or faulted condition transients) would not meet the intent of a program designed to assess actual, accumulated material damage.

Therefore, an assessment of the actual loads should be made to determine the structural adequacy of a component. In some cases, the most cost-effective approach will be to adopt some sort of automatic monitoring. In other cases, only cycles of system operations may be required to show adequate margin in component life. The amount of information required to justify actual loads is dependent largely on the nature of the loads, initiating events, and frequency, as well as the available margin in the design basis. Good estimates of actual loading can be used to show adequate margin compared to the design basis. These estimates can also serve as a sound basis for establishing the future transient set to be used in assessing the extended component life. Some locations do not require anything more that a simple record of operating cycles. This would apply to components where: the design transient is always conservative relative to the actual, there are only one or two types of transient states to consider, and there are detailed records or adequate studies to show that the actual cycles are of the same frequency as the design.

In general, operating practices have more effects on the transients associated with plant heatup and cooldown than on transients associated with power operations. The most severe normal condition thermal transient loads almost always result from plant heatup and cooldown operations. Most normal (mode 1) operating condition loads are relatively predictable and generally less severe than plant heatup and cooldown loads. Therefore, effort should be concentrated on the periods that include plant heatup, hot standby operations, and cooldown operations.

Recognizing limitations in current monitoring methods leads to the development of improved methods for transient and fatigue monitoring. The key objective is to develop a cost-effective method for collecting and maintaining records of transients and fatigue-significant loads experienced by the fatigue-sensitive components and to use that information in a way that will result in maintaining plant availability and maximizing the investment on equipment. Automated data collection methods should allow for quick retrieval of sufficient data required to prepare technical justifications in support of design operational conformance or license renewal issues. Further, some systems provide estimates of fatigue damage accumulation that can be correlated with operating modes or unusual operating events, and hence provide valuable feedback to operators to help identify ways to minimize fatigue-significant loads.

4.2.1.2 Process to Qualify Fatigue for B31.1 Piping Components

The license renewal rule permits the B31.1 code to be used to qualify fatigue if B31.1 is part of the CLB. Transient stresses and cycles for B31.1 piping are handled in a different manner than those for ASME piping. The B31.1 piping rules use a stress-range reduction factor for cyclic conditions varying from $f = 1$ for $N \leq 7{,}000$ cycles, to $f = 0.5$ for $N \geq 250{,}000$ cycles. The equation is approximately:

$$f = \frac{6}{N^{0.2}} \; 1$$

This stress range reduction factor is applied to the allowable range of thermal expansion stress for each component in the piping system. The reduction in allowable thermal expansion stress for B31.1 piping maintains at least a safety factor of 1.25 in terms of stress and 3 in terms of life [Ref. 45].

To account for additional cycles of operation during the license renewal period, several successive steps are performed based on the results of the review of the piping system stresses. These steps will effectively create screening criteria for B31.1 piping that will allow for consistent increases in cycles for most components during the license renewal period. Some components in the B31.1 piping system may need more refined evaluations to show acceptance during the license renewal period. This is because of the varying degree of thermal restraint, piping system, material properties, and the types of pipe fittings or components used. The following steps will be the same regardless of the piping system, material, or components existing in the piping system.

1. The first step for the B31.1 license renewal involves a coarse screening that applies to all piping systems, materials, and components. (This step may be bypassed by going directly to step 2 or 3, if desired.) If sufficient transient cyclic information exists, perform a transient cycle comparison to justify that the cycles expected during the license renewal period are less than those that are a part of the existing license design basis. The existing license design basis for B31.1 piping may be established using transients defined for adjacent ASME Section III Class 1 components (e.g., Reactor Vessel, Steam Generator, and Reactor Coolant Pump transients can be used for Reactor Coolant Piping.) If this comparison shows that the cycles are less than the design basis for each line, then those components are screened out and do not need to be considered further.

2. The second step applies to those lines and components that are not screened out in step 1, or those that were not included in step 1. The piping stress report, or results of the stress analysis, are reviewed to identify the minimum thermal expansion stress allowable for all remaining piping systems, materials, and components. This enveloping minimum allowable stress for all lines will be further reduced by a license renewal factor of 0.9 to account for a 50-percent increase in the number of cycles that are expected to occur during the license renewal period. This license renewal reduction factor assumes that for the calculation of f, the number of cycles is at the cyclic limit for the stress level. Assuming that the license renewal period would increase the design cycles by 50 percent, $f_{LR} = 6\,[1.5\,N]^{-.2} = 0.92\,[6N^{-.2}] = 0.92\,f$. The 0.9 license renewal factor will account for the necessary reduction regardless of the number of cycles previously used or the corresponding stress-range reduction factor (f), which is used in current licensing basis calculations.

The reduced allowable stress, or license renewal allowable, will be compared to all components in each line to determine if the existing design thermal expansion stress exceeds the license renewal allowable. Those components with thermal expansion stress ranges less than the license renewal allowable are screened out and do not need to be considered further, since the CLB essentially accounts for the license renewal period.

3. The third step applies to those components with thermal expansion stresses that exceed the minimum allowable used in step 2, or those lines that were not included in steps 1 or 2. Those components are screened by considering piping system temperature and material using the existing line-by-line, system-by-system material allowables that are already calculated in the stress report or in the qualification calculations.

The piping stress report, or results of the stress analysis, are reviewed to identify the minimum thermal expansion stress allowable for each piping system, material, and component. The allowable stress for each component will be further reduced by the license renewal factor of 0.9 to account for a 50-percent increase in the number of cycles that are assumed to occur during the license renewal period.

If the actual thermal expansion stress range is less than the reduced allowable stress, then the component considered needs no further evaluation, and the component is acceptable for increased cycles throughout the license renewal period.

For components that were not eliminated, there are several options available to address the license renewal acceptability. The most obvious options are as follows:

A. Increase the allowable stress by decreasing the number of design cycles. This can involve determining the original design basis cycles and conservatively increasing these by a factor of 1.5 to account for the increased license renewal period. Use the equation for f to recalculate a more exact stress range reduction factor. If the actual thermal expansion stress range is less than the recalculated allowable stress, then the component considered needs no further evaluation, and the component is acceptable for increased cycles throughout the license renewal period.

B. Increase the allowable stress by considering actual cycles experienced in the plant by means of cycle counting methods. A less conservative number of design cycles can be achieved by determining the actual cycles experienced in the plant during a known period of time that is representative of operations during the license renewal period and conservatively extrapolating these known cycles throughout the license renewal period. This will most likely reduce the cycles to a level that will affect the selection of the f factor and increase the allowable, as long as f is less than 1.0. Use the equation for f to recalculate a more exact stress range reduction factor. If the actual thermal expansion stress range is less than the recalculated allowable stress, then the component considered needs no further evaluation, and the component is acceptable for increased cycles throughout the license renewal period.

C. Show similarity between the B31.1 component being evaluated and an ASME, Section III Class 1 component that is shown to be qualified for the license renewal period. Typically, similarity must be shown for materials, geometry, thermal transients, and stress indices from the applicable ASME code version used for the Section III component evaluation.

   Similarity for materials should include a comparison of plant-specific subcomponent material to material used in the generic ASME evaluation in Table 4-4, which is austenitic stainless steel type 304 for all piping except at primary equipment nozzles that have a carbon steel/stainless steel interface. However, with added work the primary nozzles can be shown to be acceptable for the license renewal period regardless of the type of material used at the nozzle-to-pipe interface.

   Similarity for geometry should include a comparison of the pipe size and schedule, component type, and geometric discontinuity as applicable.

Similarity for thermal transients includes a comparison of each transient delta temperature, rate of change, and number of cycles. The thermal transients used for the ASME evaluation of components in Table 4-4 are Westinghouse standard transients and cycles.

Similarity for stress indices is important for the calculation of stress ranges, which should be maintained below the ASME NB-3653, stress allowables.

If similarity is shown using the comparisons listed above, then the results from Table 4-4 for ASME piping can be used for B31.1 piping, and those components that are acceptable for the license renewal period do not need to be considered further.

D. For components that were not shown acceptable for option C above, another option would be to perform an ASME, Section III Class 1 qualification for the component calculating the design fatigue usage factors with reduced conservatisms, as needed, for many of the components of Table 4-4 that are identified as being an issue needing evaluation for fatigue during the license renewal period. The appropriate ASME NB-3653 equation 12 and 13 stresses, thermal stress ratchet equations, and usage factor would need to be calculated to show that allowables are met. Additional cycles would need to be considered to account for the license renewal period and would need to be included in the calculation of usage factor. If the usage factor and stress equation allowables are met, then the component considered needs no further evaluation, and the component is acceptable for increased cycles throughout the license renewal period.

For components that are not acceptable for increased cycles throughout the license renewal period, then alternative approaches that are performed for piping components designed to ASME, Section III Class 1 rules should be followed to show acceptability. These are outlined in Subsection 4.2.1, steps 2, 3, and 4.

4.2.1.3 Evaluation of Actual/Postulated Flaws

The following methodology for the flaw tolerance evaluation is based on current practice for operating plants. For license renewal, the final criteria for the flaw tolerance approach should account for appropriate environmental factors on fatigue crack initiation and growth. Risk-based evaluations may also be applicable to determine the frequency and extent of the coverage. The application to license renewal requires utilities to follow current industry activities to completion and final NRC position.

Evaluation of actual (detected) or postulated (reference) flaws is an accepted technique for justifying continued operation of current operating nuclear power plant Class 1 piping and pressure boundary components. Such methodology may also be useful for evaluation of Class 1 piping and pressure boundary components where current inservice examination or design analysis procedures cannot be shown to manage the effects of potentially significant fatigue damage adequately. Potentially significant degradation mechanism fatigue, which was identified in Section 3.0 as possibly affecting a number of the Class 1 piping and associated components covered by this report, can be shown to be managed using flaw evaluation techniques on a plant-specific basis.

For example, no specific guidance is provided in Subsection IWB or the Section XI appendices for justifying continued operation of Class 1 piping and pressure boundary components with a flaw by analytical evaluation, in accordance with IWB-3142.4. However, Appendices A (Analysis of Flaws), C (Evaluation of Flaws in Austenitic Piping), and H (Evaluation of Flaws in Ferritic Piping) do provide fracture mechanics methods that can be adapted for Class 1 piping and pressure boundary component assessments. These appendices include systematic procedures for flaw model selection, calculation of stress intensity factors, material property determination, and design transients to be considered in component analysis models. The elastic-plastic fracture mechanics techniques described in Appendix C, in particular, provide a suitable framework for the evaluation of flaws and the justification for continued operation with flaws for Class 1 piping and pressure boundary components.

Fracture mechanics techniques can be employed for justification of continued operation in two ways. First, inspection results may identify flaws that can be shown by analysis either not to grow as the consequence of continued service or to grow at such a low rate that current inservice inspection schedules (or more frequent inservice inspections) will be able to ensure the integrity of Class 1 piping and pressure boundary components. Second, flaws can be postulated in Class 1 piping and pressure boundary components and shown either not to grow or to grow slowly. Through such a "flaw tolerance" approach, appropriate inservice inspection schedules can be formulated and compared to current requirements. This comparison may show that the current schedule is adequate, may need to be accelerated, or can be less frequent. Also, an appropriate schedule can be determined for Class 1 piping and pressure boundary components not included in the current inspection program.

In particular, an alternative approach for fatigue-sensitive Class 1 piping and pressure boundary components, irrespective of the existence of an explicit fatigue design basis, could be the justification of continued operation for license renewal of Class 1 piping and associated components in the presence of either a real (detected) or an assumed (reference) flaw. For the case of a relevant condition involving a crack-like indication (see IWB-3519.1(c)) where supplementary examination in accordance with IWB-3142.2 has characterized an unacceptable flaw, the justification for continued operation involves a crack growth assessment. For the case where the need for, or the adequacy of, current or augmented inservice inspection coverage is sought, a reference flaw may be postulated and the growth of such flaws determined under plant operating conditions. In either case, an analysis of fatigue crack propagation is an essential feature of the evaluation process.

While estimates of the influence of PWR water environmental effects on stainless steel base and weld metal have been made, ranging from a factor of 2 to 10 higher in crack growth rate than rates in air, the values used in license renewal applications should be justified on a plant-specific basis. For most situations, fatigue crack growth rates in PWR water environments are expected to be minimal. Similarly, it has been demonstrated [Ref. 46] that fast neutron irradiation has no significant effect on the fatigue crack growth of austenitic stainless steels for fluence levels of interest here. Further, the work of References 47 and 48 have demonstrated that C_o is not substantially impacted by the PWR water environment.

4.2.2 Aging Management Program for Thermal Aging (AMP-3.6 and AMP-3.7)

The effects of thermal aging embrittlement were found to be potentially significant for all cast austenitic stainless steel (CASS) Class 1 reactor coolant system (RCS) components in Section 3.0 as the result of the potential loss of fracture toughness caused by long-term exposure of the materials to RCS operating temperatures. The magnitude of this loss of fracture toughness depends on several factors, such as material chemistry (e.g., molybdenum, silicon, and chromium content, and measured or calculated delta ferrite), casting method (statically or centrifugally cast), and duration of exposure to operating temperature. The effects of CASS thermal aging embrittlement will be managed by a defined sequence of analytical procedures and inservice evaluations, beginning with the extension of the plant's leak-before-break (LBB) analysis and continuing, as needed, through ASME Code, Section XI inservice examinations and flaw evaluations.

Tables 4-9 and 4-10 show the program attributes for the management of the loss of fracture toughness due to thermal aging embrittlement of Class 1 piping and associated components fabricated from cast austenitic stainless steel.

Two areas of interest exist in determining the significance of the effect of thermal aging in plant license renewal, leak-before-break (LBB) analysis, and RCL pump casing analysis in-lieu-of inspections.

As discussed in Section 3.0, the LBB and fracture mechanics evaluations of the RCP casings are TLAAs. These TLAAs consider the degradation of material toughness properties due to thermal aging.

Most plants were licensed for LBB for the CLB. For the license renewal term, LBB is considered to be a licensing issue.

LBB evaluations are based on the application of elastic-plastic fracture mechanics and leak rate calculation methodologies. These methodologies incorporate the effects of thermal aging by using material fracture toughness values, J_{Ic}, J_{max}, and T_{mat} corresponding to experimental data of a cast austenitic stainless steel sensitive to effects of thermal aging. This evaluation is based on a 40-year plant design life.

Based on experimental results, it has been proven that the end-of-life material properties are a function of material chemistry composure, particularly on the amount of silicon, chromium and molybdenum.

TABLE 4-9
AGING MANAGEMENT ACTIVITIES AND PROGRAM ATTRIBUTES - AMP-3.6 FOR THERMAL AGING (LEAK-BEFORE-BREAK ANALYSIS)

Attribute	Description
Scope	Austenitic stainless steel static castings for Class 1 reactor coolant piping, RCP casings, and Class 1 valve bodies.
Surveillance Techniques	Validate evaluation of thermal aging using LBB methodology for extended plant life
Frequency	One time prior to end of design life
Acceptance Criteria	Margin > 2 between leakage flaw size and critical size and Margin > 10 between calculated leak rate and leak rate detection capability
Corrective Actions	• Repair per ASME Code, Section XI IWB-4000 OR • Replace per ASME Code, Section XI IWB-7000 OR • Inservice Examination/Flaw Evaluation
Confirmation	Preservice examinations consisting of: • IWA-4700, pressure test following repair by welding, is performed prior to return of the system to service • IWB-2420, Successive Inspections • IWB-2430, Additional Examinations

TABLE 4-10
AGING MANAGEMENT ACTIVITIES AND PROGRAM ATTRIBUTES – AMP-3.7 FOR THERMAL AGING (FRACTURE MECHANICS EVALUATION OF RCP CASINGS IN-LIEU-OF VOLUMETRIC INSPECTIONS)

Attribute	Description
Scope	RCP casings Section XI volumetric inservice inspection
Surveillance Techniques	Demonstrate compliance with Code Case N-481 for the license renewal terms, which allows the replacement of volumetric examinations of RCP casings with a fracture mechanics based integrity evaluation supplemented by visual inspections.
Frequency	One-time fracture mechanics evaluation and visual inspections per the plant's inservice inspection program
Acceptance Criteria	Per Code Case N-481
Required Actions	• Perform Section XI volumetric ISI OR • Repair per ASME Code, Section XI IWB-4000 OR • Replace per ASME Code, Section XI IWB-7000
Confirmation	Preservice examinations consisting of: • IWA-4700, pressure test following repair by welding, is performed prior to return of the system to service • IWB-2420, Successive Inspections • IWB-2430, Additional Examinations

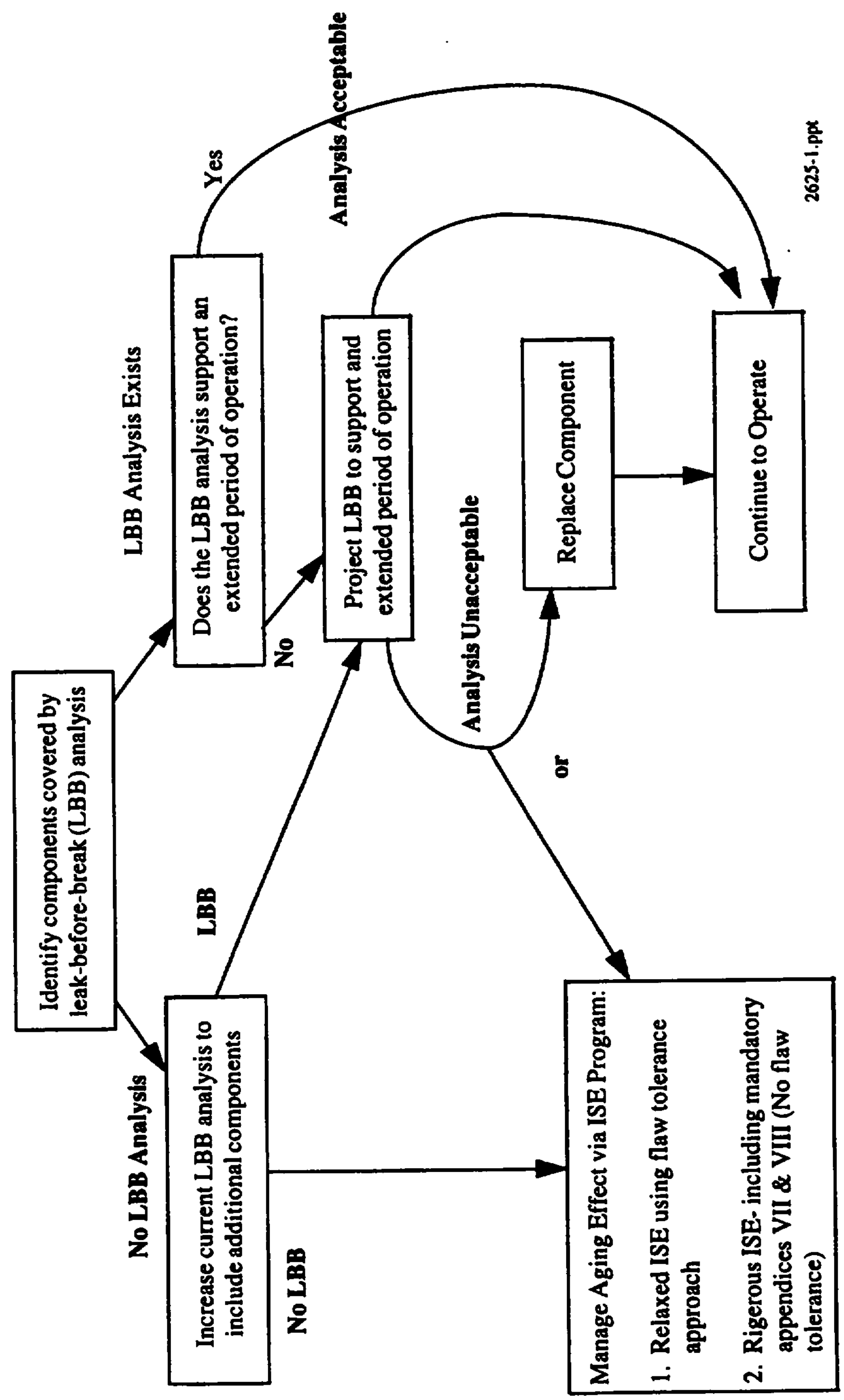

Figure 4-3 Flow Chart for Management of Thermal Aging

To validate LBB evaluations performed on plants for license renewal, fully aged fracture toughness needs to be used and the NRC's approval needs to be obtained. The LBB calculation would need to be revised for an extended period of operation.

A similar revalidation is required for the RCL pump casings. ASME Code Case N-481 allows for the replacement of volumetric inspections of the casings based on a fracture-mechanics integrity evaluation supplemented by visual examinations of the casings. To validate the Code Case N-481 analyses previously performed, revised fracture toughness data would have to be recreated based on NUREG/CR-4513, Rev 1. Techniques contained in this report consider greater aging periods.

For CASS materials, the degradation of material toughness properties due to thermal aging are considered in both the LBB and fracture mechanics evaluations of the RCP casings. Structural integrity for a component can be demonstrated by evaluating it to LBB or ASME Code Case N-481 criteria. Therefore, revalidation of the LBB evaluations and compliance with Code Case N-481 for the license renewal term will manage the thermal aging effects by demonstrating structural integrity for the component.

4.2.2.1 Aging Management Program for Thermal Aging - LBB Analyses (AMP-3.6)

The aging management program attributes shown in Table 4-9 address the LBB evaluation, which includes loss of fracture toughness of CASS due to thermal aging. The structural design basis for the primary loop piping and components required the postulation of nonmechanistic circumferential pipe breaks. As a result, plant hardware, such as pipe whip restraints and jet shields, was added to mitigate the potential dynamic consequences of these postulated pipe breaks. An LBB analysis provides an accepted, mechanistic pipe-break evaluation methodology that can be used to establish that circumferential pipe breaks will not occur within the primary loop piping.

LBB evaluations are based on the application of elastic-plastic fracture mechanics and leak rate calculation methodologies. These methodologies incorporate the effects of thermal aging embrittlement by using material fracture toughness values, J_{IC}, J_{max}, and T_{mat}, corresponding to experimental data for CASS material that is the most sensitive to the effects of thermal aging embrittlement. The evaluation is based on a 40-year plant design life. For the purposes of license renewal, the LBB calculation may need to be revised for an extended period of operation, using fully-aged fracture toughness data for the limiting materials.

LBB evaluations have been performed for the primary loop piping in all Westinghouse PWR plants. These evaluations follow the recommendations and criteria proposed in NUREG 1061, Volume 3 [Ref. 49] and the methodology described in Standard Review Plan 3.6.3 [Ref. 50]. The procedures include: (1) the postulation of an assumed surface flaw at the governing location, which will be the location with the combination of highest stress and limiting fracture toughness, with a demonstration of flaw stability under applied loads and any subsequent flaw growth; (2) postulation of an assumed through-wall flaw at the governing location with a demonstration that any leakage is assured of detection, with appropriate margin, using the installed leak detection system at the plant, when the piping is subjected to normal operating loads; and (3) demonstration of adequate margin between the leakage flaw size and the critical flaw size under faulted condition loading. Limiting locations must be based on appropriate material properties for base and weld metals, including any long-term material degradation effects such as thermal aging embrittlement.

If the LBB evaluation is not revalidated, the thermal aging effect can be managed by an ISE program (see Figure 4-3).

4.2.2.2 Aging Management Program for Thermal Aging - Inservice Examination/Flaw Evaluation (AMP-3.7)

The aging management program attributes shown in Table 4-10 address the fracture mechanics evaluation of RCP casings, which includes loss of fracture toughness of CASS due to thermal aging in-lieu of volumetric inspections. Class 1 piping and associated reactor coolant system (RCS) components also are subject to the inservice inspection requirements of the ASME Code Section XI, Subsection IWB, which include Examination Category B-J for the volumetric and surface examination of pressure retaining welds in piping; Examination Category B-L-1 for the volumetric examination of pressure-retaining welds in pump casings; Examination Category B-M-1 for the volumetric and surface examination of pressure-retaining welds in valve bodies; and Examination Categories B-L-2 and B-M-2 for the visual examination of the internal surfaces of pump casings and valve bodies, respectively, when disassembled for maintenance, repair, or volumetric examination.

Acceptance criteria for flaws detected in austenitic stainless steel piping pressure-retaining welds and adjacent base metal during the B-J volumetric and surface examinations are provided in IWB-3514.3 for materials with a specified minimum yield strength of 35 ksi or less at 100EF. Therefore, these criteria apply to all welds and adjacent base metal in piping fabricated from ASME/ASTM SA-451 Class 1 cast austenitic stainless steel piping. Acceptance criteria for flaws detected in austenitic stainless steel pump casing and valve body pressure-retaining welds and adjacent base metal during the B-L-1 and B-M-1 volumetric and surface examinations are provided in IWB-3518.1 for materials with a specified minimum yield strength of 35 ksi or less at 100EF. Therefore, these criteria apply to all welds and adjacent base metal in pump casings and valve bodies fabricated from ASME/ASTM SA-351 Class 1 cast austenitic stainless steel pump casings and valve bodies. No flaw acceptance criteria are provided for the castings themselves, other than the material adjacent to weldments, except those permitted by the workmanship requirements of the material specifications and the construction code of record.

However, the draft Safety Evaluation Report (SER) by the U.S. NRC on the B&W Owners Group Report BAW-2243 on RCS piping stated that "the staff finds the B&WOG's program for managing the loss of fracture toughness of cast austenitic stainless steel during the period of extended operation acceptable." The staff noted that "the B&WOG evaluated the loss of fracture toughness of cast stainless steel due to thermal aging and concludes (sic) that the toughness of aged cast stainless steel is similar to that of submerged arc welds (SAWs). The staff reviewed the recently developed lower-bound toughness property for aged cast stainless steel.....and agrees that......aged cast stainless steel and SAWs could be treated similarly regarding their toughness behavior." As a result of this finding by the U.S. NRC, the flaw acceptance criteria of Tables IWB-3641-5 and IWB-3641-6 for SAW and shielded-metal-arc (SMA) welds are applicable to flaws detected and sized in aged stainless steel castings.

Standards for the B-L-2 and B-M-2 visual (VT-3) examinations of the internal surfaces of pump casings and valve bodies are provided in IWB-3519.1, with relevant conditions that include "crack-like surface flaws developed in service or grown in size beyond that recorded during preservice visual examination."

An alternative to the B-L-1 and B-L-2 volumetric and visual inservice examination requirements for pump casings are the visual examinations of the internal and external surfaces of pump casings, combined with a flaw tolerance evaluation, of ASME Nuclear Code Case N-481. The U.S. NRC have now endorsed Nuclear Code Case N-481 in Regulatory Guide 1.147, with no enhancement of its provisions for the visual (VT-1) inspections of the external surface or the visual (VT-3) inspections of the internal surface. A number of plants have exercised this alternative, postulating the required reference flaw and establishing the stability of this reference flaw throughout the component service life, including thermal aging embrittlement that might degrade the properties of the pump casing during service. These Code Case N-481 fracture mechanics evaluations can be extended through the license renewal term by updating any analyses previously performed by establishing the stability of the postulated reference flaw for the extended period, using appropriate aged cast stainless steel fracture toughness properties (see NUREG/CR-4513, Revision 1).

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5.0 SUMMARY AND CONCLUSIONS

Class 1 piping and associated components have been reviewed for aging management as part of the Westinghouse Owners Group (WOG) Life Cycle Management/License Renewal (LCM/LR) program. Class 1 piping and associated components are subject to an aging management review because they perform an intended function, perform this intended function in a passive manner, and are long-lived. This aging management review has identified aging effects and evaluated these effects to determine which require management during an extended period of operation. For those effects that require management, options have been provided.

5.1 SUMMARY

Class 1 piping and associated components perform the intended function of ensuring the integrity of the reactor coolant pressure boundary. Class 1 piping and associated components also support system-level intended functions. This is discussed in detail in Section 2.0.

The mechanisms identified from review of design limits, time-limited aging analyses (TLAAs), and aging are:

- Fatigue and operational issues related to fatigue
- Corrosion/stress corrosion cracking (SCC)
- Irradiation embrittlement
- Thermal aging
- Erosion and erosion/corrosion
- Wear
- Creep and stress relaxation

Aging effects are identified in Section 2.0 and evaluated in Section 3.0 to determine potential degradation of the Class 1 piping and associated components intended function. The following aging effects require management during an extended period of operation:

- Fatigue-related cracking for fatigue-sensitive items
- Thermal aging-related cracking of austenitic stainless steel static castings
- Material loss caused by wear of reactor coolant pump (RCP) and Class 1 valve closure elements
- Loss of bolt preload due to stress relaxation of bolted RCP and Class 1 valve closures

Options to manage these aging effects have been provided. All options are described in Section 4.0.

5.2 CONCLUSIONS

Implementation of aging management options will manage the identified aging effects. Therefore, it is concluded that the Class 1 piping and associated components intended function will be maintained during an extended period of operation. System-level intended functions supported by Class 1 piping and associated components will also be maintained.

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Domestic Utilities
American Electric Power
Carolina Power & Light
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Consolidated Edison
Duquesne Light
Duke Power
Georgia Power
Florida Power & Light
Houston Lighting & Power
New York Power Authority
Northeast Utilities
Northern States Power
Pacific Gas & Electric
Public Service Electric & Gas
Rochester Gas & Electric
South Carolina Electric & Gas
Southern Nuclear
Tennessee Valley Authority
TU Electric
Union Electric
Virginia Power
Wisconsin Electric Power
Wisconsin Public Service
Wolf Creek Nuclear

International Utilities
Electrabel
Kansai Electric Power
Korea Electric Power
Nuclear Electric plc
Nukleama Elektrana
Spanish Utilities
Taiwan Power
Vattenfall

OG-97-060

NRC Project Number 686
WCAP-14575

June 13, 1997

To: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Westinghouse Owners Group
Response to NRC Request for Additional Information on WOG Generic Technical Report WCAP 14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components"

Reference: NRC letter dated April 18, 1997 from P.T. Kuo to R.A. Newton, Westinghouse Owners Group

Attached are the Westinghouse Owners Group responses to the NRC's Request for Additional Information on WCAP-14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components".

Please distribute these responses to the appropriate people in your organization for their review. These responses will provide the basis for our discussion at an NRC / WOG meeting scheduled with the License Renewal Project Directorate office for July 10, 1997 from 1-3 PM.

If you have questions on specific technical issues from these RAIs for this meeting, please contact Frank Klanica, Westinghouse, at (412) 374-6392, Charlie Meyer, Westinghouse, at (412) 374-5027, or myself at Wisconsin Electric Power Company, (414) 221-2002.

Very truly yours,

Roger A. Newton

Roger A. Newton, Chairman
LCM/LR Working Group
Westinghouse Owners Group

cc: R.J. Prato, USNRC, (1L, 1A)
Pao-Tsin, USNRC, (1L, 1A)
LCM/LR Working Group (1L, 1A)
Steering Committee (1L, 1A)
C.E. Meyer, ECE 4-22 (1L, 1A)
F. Klanica, ECE 573D(1L, 1A)
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June 13, 1997

All receive 1L, 1A

bcc:

T.A. Meyer	ECE 4-22
M.P. Osborne	ECE 4-22
S.R. Bemis	ECE 5-16
S.A. Binger	ECE 5-16
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S.M. DiTomasso	ECE 5-16

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EDRE-EMT-126

From: EQUIPMENT DESIGN & REGULATORY ENGINEERING
WIN: 284-6392
Date: June 12, 1997
Subject: License Renewal Evaluation: Aging Management for Class 1 Piping and Associated Components

To: C. E. Meyer / EC-East 4-22

cc: C. H. Boyd / EC-East 573D
M. A. Gray / EC-East 3-04
D. C. Bhowmick / EC-East 3-04
R. L. Brice-Nash / EC-East 3-04
C. C. Kim / EC-East 509D

References:
1) US NRC letter dated April 18, 1997, Project No. 686, "Request For Additional Information Regarding the Westinghouse Owners Group Topical Report WCAP-14575 (TAC No. M96439 and M92414)"

Attached are the response to the NRC RAI's on Class 1 Piping (ref.). Note that PVP-Vol. 306 (ref. 43 in WCAP-14575) is attached to the responses. The Word file, which is the electronic copy of the responses without PVP-Vol. 306, will also be sent to you.

Frank Klanica
F. Klanica
Engineering and Materials Technology

RAI and Report Section Cross Reference Table

RAI NUMBER	DESCRIPTION	REPORT SECTION
1	(a) Describe how the AMP for fatigue addresses thermowell high cycle fatigue. (b) Explain what is intended by last sentence in 3.3.2 and how the conclusion was used to develop an AMP for fatigue.	Tables 3-2 through 3-17 3.3.2
2	(a) Discuss how step 2 (ISI), in the AMP for fatigue, assures that the licensing basis criteria has been met. (b) Discuss how step 3 (flaw tolerance & LBB), in the AMP for fatigue, demonstrates that the licensing basis criteria has been met. (c) Discuss how environmental effects are addressed in the AMP for fatigue. (d) Describe test data used to establish PVRC criteria for water flow velocity in the AMP for fatigue.	4.2.1 4.2.1 4.2.1 Table 4-8
3	Identify other aging management programs to be included.	No Change
4	Describe how aging will be managed for components in inaccessible areas.	No Change
5	Describe how the owner's group reviewed applicable generic communications and associated licensee commitments.	3.1
6	Are current activities to manage boric acid corrosion consistent with the programs developed and implemented in response to Generic Letter 88-05?	No Change
7	Discuss why the program with the set of attributes identified would be an effective aging management program.	No Change
8	Will continuing commitments be addressed in plant specific applications for license renewal (rather than generically)?	No Change
9	Provide a stress corrosion cracking aging management program for components listed in NUREG-1557 (pages B-66 and B-67).	No Change
10	Provide an assessment for the cracking of thermal sleeves.	No Change

REQUEST FOR ADDITIONAL INFORMATION
WESTINGHOUSE OWNERS GROUP TOPICAL REPORT WCAP-14575
"License Renewal Evaluation: Aging Management for Class 1 Piping and Associated Pressure Boundary Components"

Request for Additional Information

1. In Section 3 of the report, aging effects that require management during the extended period of operation are identified.

1a. Section 3.3.1 of the report indicates that Class 1 thermowells are the only pressure components that are subjected to a dynamic load associated with flow-induced vibration and potentially susceptible to high-cycle fatigue damage. Describe how the aging management program for fatigue addresses this issue.

Response:

Based on the analyses of the Class 1 thermowells, the degradation sustained from the effects of low and high cycle fatigue was determined to be insignificant to all Class 1 thermowells covered by the evaluation. The hot and cold leg fast response RTD thermowells were considered to be representative for all Class 1 thermowells based on the thin walled tapered tip of the fast response thermowell and the RCS flow loads. Since the thermowells are out of the frequency range of seismic excitations, the only significant fatigue loads are induced by the turbulent lift loads which could potentially cause high cycle fatigue. Significant margin was calculated for high cycle fatigue based on an allowable for 10^{11} cycles. Extrapolating the design life from 40 years to 60 years has an insignificant effect on the allowable. Thus, the high cycle fatigue evaluations that were performed for the current licensing basis are valid for the license renewal term. The low cycle fatigue investigation showed that the exempt rules specified in NB-3222.4(d) were satisfied. Since the peak stress intensities are not a function of cycles, the fatigue waiver evaluations that were performed for the current licensing basis are valid for the license renewal term. Therefore, the Class 1 thermowells are not considered to be fatigue-sensitive items for license renewal. The report will be revised to include the Class 1 thermowells in Tables 3-2 through 3-17, as applicable. A rating of N, which are components that were considered to not be an issue, will be given for all of the potential aging effects for the Class 1 thermowells.

Request for Additional Information

1b. Section 3.3.2 contains a description of a fatigue assessment of B31.1 piping design. The final sentence concludes, "Based on the successful operating history of fossil plants (using the B31.1 approach) and the high cost of evaluating these stresses with a detailed fatigue analysis, this was considered to be an acceptable approach for nuclear plants." Explain what is intended by this sentence. Also, explain how this conclusion was used to develop the aging management program for fatigue.

Response:

This section discusses the fatigue methodology for B31.1 piping design and the results of the EPRI report (TR-102901) that compares the B31.1 criteria to the ASME code, Section III criteria for Class 1 piping. The last sentence is intended to explain why some operating nuclear power plants, that have piping designed to the B31.1 code criteria, did not need to have the piping reanalyzed when the new ASME code, Section III criteria was introduced. "The successful operating history of fossil plants designed using the B31.1 approach, indicates that the structural adequacy of a piping design can be maintained using the B31.1 code criteria. In addition, since it is very costly to reanalyze the B31.1 piping design to ASME code, Report No. TR-102901 is judged adequate to show that no further evaluations are needed to ensure safe operation."

The two sentences, shown above, will replace the sentence in question.

The aging management program for fatigue is based on the proposed industry position on fatigue evaluations for license renewal and the B31.1 code requirements. Section 4.2.1.2, "Process to Qualify Fatigue for B31.1 Piping Components" applies to plants that have the B31.1 code included in the current licensing basis.

Request for Additional Information

2. In Section 4.2.1 of the report, the aging management program for fatigue is described.

2a. Step 2 of the program appears to allow the use of ASME Code, Section XI, inspection techniques to demonstrate the acceptability of a component as an alternative to meeting the licensing basis criteria in Step 1. The staff has not endorsed this position. Discuss how the use of this alternative provides assurance that the licensing basis criteria has been met at a facility.

Response

The report will be modified to incorporate the revised proposed industry position on fatigue.

The last paragraph of step 2 of the "Position", in Section 4.2 (Revision 0, page 152), will be clarified to explain how ISI requirements manage cracking caused by fatigue and why a program based on these ISI requirements will continue to be effective during an extended period of operation. This paragraph will be modified as follows:

"Since the examination methods and related evaluations described above will allow the detection, evaluation, and/or repair of minor cracks, caused by fatigue, this management option will maintain the intended function of the Class 1 piping and pressure boundary components. Specifically, the flaw acceptance standards in subsection IWB-3500, which are the current industry accepted criteria, are stringent enough that indications identified by the evaluations do not represent a loss of the reactor coolant pressure boundary intended function of the Class 1 piping and pressure boundary components under design-basis loads. It is noted that other plant programmatic requirements (Technical Specifications - RCS Operational Leakage Limits) require a plant shutdown to repair the degradation before an intended function would be lost. The criteria of IWB-3500 would allow further evaluation and/or repair of indications prior to the loss of the intended function of the Class 1 piping and pressure boundary components. These inspections are required periodically and are not tied to a specific design life. Because the transient loading frequencies are not anticipated to significantly increase during the license renewal period, these inspection periods will remain effective throughout the license renewal period, as long as CLB cyclic commitments are met (as confirmed in step 1)."

It should be noted that the proposed industry position defines the CLB as a combination of the fatigue design basis, the fatigue operating basis, and the regulatory oversight process. This definition includes any requirements for assuring that the plant operates within commitments on cyclic duty, and items such as resolution of generic fatigue issues or regulatory information notices and bulletins.

Request for Additional Information

2b. Step 3 of the program appears to allow the use of flaw tolerance or leak-before-break analysis to demonstrate the acceptability of a component as an alternative to meeting the licensing basis criteria in Step 1. The staff has not endorsed these positions. Discuss how the use of these alternatives will demonstrate that the licensing basis criteria has been met at a facility.

Response:

The report will be modified to incorporate the revised proposed industry position on fatigue.

The second and fourth paragraphs of step 3 of the "Position", in Section 4.2 (Revision 0, page 153), indicate that the structural integrity of the Class 1 piping and pressure boundary components is maintained by using either the flaw tolerance or leak-before-break analyses.

In section 4.2.1.3, "Evaluation of Actual/Postulated Flaw", the second paragraph on page 160, (shown below), explains how the structural integrity of the Class 1 piping is maintained by using the flaw tolerance approach.

> "Fracture mechanics techniques can be employed for justification of continued operation in two ways. First, inspection results may identify flaws that can be shown by analysis either not to grow as the consequence of continued service or to grow at such a low rate that current inservice inspection schedules (or more frequent inservice inspections) will be able to ensure the integrity of Class 1 piping and pressure boundary components. Second, flaws can be postulated in Class 1 piping and pressure boundary components

and shown either not to grow or to grow slowly. Through such a "flaw tolerance" approach, appropriate inservice inspection schedules can be formulated and compared to current requirements. This comparison may show that the current schedule is adequate, may need to be accelerated, or can be less frequent. Also, an appropriate schedule can be determined for Class 1 piping and pressure boundary components not included in the current inspection program."

The Leak-Before-Break (LBB) criteria is more stringent than the flaw tolerance criteria. In section 4.2.2.1, "Aging management program for Thermal Aging - LBB Analyses (AMP-3.6)", the second and third paragraphs on page 160, (excerpts shown below), explain how the structural integrity of the Class 1 piping is maintained by using the LBB criteria.

"The LBB criteria includes elastic-plastic fracture mechanics analyses and leak rate calculation methodologies. These evaluations follow the recommendations and criteria proposed in NUREG 1061, Volume 3 and the methodology described in the Standard Review Plan 3.6.3. The procedures include: (1) the postulation of an assumed flaw at the governing location, which will be the location with the combination of highest stress and limiting fracture toughness, with a demonstration of flaw stability under applied loads and any subsequent flaw growth; (2) postulation of an assumed through-wall flaw at the governing location with a demonstration that any leakage is assured of detection, with appropriate margin, using the installed leak detection system at the plant, when the piping is subjected to normal operating loads; and (3) demonstration of adequate margin between the leakage flaw size and the critical flaw size under faulted condition loading."

Satisfying this criteria will demonstrate that the structural integrity of the Class 1 piping and pressure boundary components will be maintained for a through wall crack in the piping.

Request for Additional Information

2c. The discussion following Step 3 of the program describes issues regarding environmental effects on fatigue. The location of this discussion is confusing because Step 3 appears to be an alternative to Step 1. SECY 95-245 provided the staff recommendation regarding the use of environmental fatigue data for license renewal evaluations. Clarify the method in which the staff recommendation in SECY 95-245 is addressed by the program.

Response:

The report will be modified to incorporate the revised proposed industry position on fatigue. This revised position considers environmental effects for an extended period of operation.

The revised industry position has expanded the first step to clarify that environmental effects will be considered, as appropriate. Specifically, the first step in the revised process (identifying fatigue sensitive sub-components) includes consideration of reactor water environmental effects. If sub-components are identified as fatigue sensitive based on this preliminary screening, the second step quantitatively addresses the significance of environmental effects. For sub-components that are not identified as fatigue sensitive in step 1, the CLB cyclic duty is checked to confirm it envelopes the number of cycles expected through the license renewal term.

The appropriate changes to sections 3.3, 3.4, 4.2.1, Tables 3-2 thru 3-16, 4-4, 4-5, 4-6, and 4-7, and Figures 4-1and 4-2 will be made to incorporate the revised industry position that considers environmental effects.

The report requires utilities to follow current industry activities to completion until the final NRC position is given in this area.

Request for Additional Information

2d. Table 4-8 lists parameters developed by the PVRC to identify components where the environmental effect on fatigue life are not considered significant. Describe the test data used to establish the criteria for water flow velocity.

Response:

The criteria for water flow velocity in Table 4-8 will be corrected to show >3 m/sec (9.843 ft/sec) instead of >3 m/sec (3.3 ft/sec). The test data used to establish the criteria for water flow velocity should be described in the references listed in reference 43 (attached).

Request for Additional Information

3. Are there aging management programs (other than the ASME Code examinations that you cite) that you want the staff to generically credit to participating WOG Plants? If so, identify each program and provide more detail about actions taken, results, and validity for the period of extended operation. For example, the report does not describe programs related to generic communications and technical specifications other than to list the documents in a table. Existing augmented examinations should also be described and justified to demonstrate that the effects of aging will be adequately managed so that the intended function will be maintained for the period of extended operation.

Response:

Section 4 of the report contains all of the generic aging management programs for which the WOG wants the staff's review and approval. The WOG uses six attributes to provide the details that are applicable to all domestic WOG plants. Plant-specific License Renewal applications will provide additional details consistent with their CLB and as deemed necessary by the utility.

Current commitments (those that are part of the CLB), which are credited for aging management, will be addressed in plant-specific applications. As stated in the Rule, 10 CFR 54.33, CLB requirements will continue during the extended period of operation unless otherwise justified by the utility and approved by the NRC.

Request for Additional Information

4. Are there any relevant components in areas inaccessible for maintenance and inspection? If so, how will their aging be managed?

Response:

There may be relevant components, for the aging management programs described in Sections 4.1 and 4.2, that are located in areas inaccessible for maintenance and inspection. It is the individual plant's responsibility to identify their own inaccessible areas relative to RCS piping and associated components during preparation of their LR application. The plant specific inaccessible areas cannot be addressed in a generically applicable report. The possibility of components located in areas that are considered to be inaccessible for maintenance and inspection is discussed below for the AMPs in Sections 4.1 and 4.2.

The evaluation has determined that the aging effects that require management are identified in sections 4.1 and 4.2. Section 4.1 provides current industry practices and section 4.2 provides additional activities and attributes required to manage aging effects.

Section 4.1 of the report describes the attributes for current aging management programs for wear of closures and stress relaxation of bolts. Current industry practices for these AMPs should already account for the possibility of RCP and Class 1 valve closure flanges and bolting located in areas that are considered to be inaccessible for maintenance and inspection.

Section 4.2 of the report describes additional activities and program attributes for fatigue and thermal aging. Both of these programs include analyses, in addition to inspections, as options to manage the aging effects from fatigue and thermal aging for license renewal. If relevant components are located in areas inaccessible for maintenance and inspection, the analyses options could be considered as acceptable alternatives to the inspection options.

Request for Additional Information

5. Describe how the owner's group reviewed applicable generic communications and associated licensee commitments. The staff found generic communications of the aging effects of the RCS not discussed in the report, for example Bulletin 82092 on bolting, and Generic Letter 85-20 on thermal sleeves.

Response

Section 3.1 will be revised to describe the process used by the WOG to review Generic Communications. An updated review will be performed to capture any additional items that occurred, or were missed, since the original review was done three years ago.

The following information was provided to the authors in the WOG GTR template:

"Identify plant-specific operating experience which identifies aging effects. Review operating and maintenance history. This should include, but is not limited to: plant maintenance data, inservice inspection data, industry experience, NPRDS data, vendor data, EPRI reports, NUREGs, Licensee Event Reports (LERs), DOE Aging Management Guidelines (AMGs), NUMARC License Renewal Industry Reports, NRC generic letters/bulletins/notices, the Westinghouse Information Delivery System (IDS), and the internet. Many of these sources are readily available in the technical library. When using the internet, as any other reference, ensure that the information is timely. Identify any unresolved issues, see the Westinghouse technical lead for the latest EPRI memorandum."

Request for Additional Information

6. Your report states that current activities are sufficient to manage boric acid corrosion. Are current activities, as referenced in your report, consistent with the programs developed and implemented in response to Generic Letter 88-05? If no consistent with GL 88-05, describe current activities and provide a basis for how your current programs provide reasonable assurance that the aging effect will be managed during the period of extended operation.

Response:

The WOG feels that the current activities referenced (leakage monitoring and walkdowns) are consistent with responses to Generic Letter 88-05. Additional details will not be provided in this generically applicable report. Plant-specific License Renewal applications will provide these details as consistent with their CLB and as deemed necessary by the utility.

The WOG supports the position that boric acid corrosion of external surfaces is not related to aging. This type of corrosion is caused by an event, in this case, degradation of the reactor coolant pressure boundary. The degradation causes an abnormally harsh environment that can cause degradation. Since current activities monitor for the event (leakage) and the degradation it causes (loss of material) and repair degradation as necessary, the effects of events are managed by current activities and do not require separate aging management programs.

Request for Additional Information

7. Discuss why the program with the set of attributes identified would be an effective aging management program (i.e., provide reasonable assurance that a program with the attributes described would be able to detect and correct the effects of aging before the component would reach a condition in which it could not perform its intended function under all CLB design conditions.) Explain why all six attributes identified in your report may not be necessary for a program.

Response:

The purpose of the six WOG attributes is to describe the generic aging management programs in sufficient detail for use by the utility and review/approval by the NRC. These descriptions, as contained in section 4, explain how the program manages an aging effect(s) to maintain the appropriate intended function(s) for an extended period of operation. Section 4 also contains text explaining why these programs will remain effective during an extended period of operation.

All six attributes may not be necessary based on the type of the activity performed by the program. For example, a program that uses analytical techniques to ensure intended functions are maintained do not have a surveillance technique. An analysis does not inspect anything. The analysis (and specifically the results) would be part of the acceptance criteria used to determine further actions: acceptance, further analysis with less conservative assumptions, or replacement.

Request for Additional Information

8. Will continuing commitments be addressed in plant specific applications for license renewal (rather than generically)?

Response:

Current commitments (those that are part of the CLB), which are credited for aging management, will be addressed in plant-specific applications. As stated in the Rule, 10 CFR 54.33, CLB requirements will continue during the extended period of operation unless otherwise justified by the utility and approved by the NRC.

Request for Additional Information

9. NUREG-1557 (pages B-66 and B-67) lists stress corrosion cracking as an aging effect for a number of components in the reactor coolant system requiring aging management. For some of the identified components the issue was unresolved. Provide an aging management program for these components.

Response:

It appears that the two unresolved issues in NUREG-1557, on pages B-66 and B-67, are identified as follows.

#1) IGSCC can occur under the operating conditions (water chemistry) during shutdown because oxygen is introduced to primary coolant during cool down to control CRUD-bursts, and coolant is exposed to air during many shutdowns (S-V-38).

#2) The potential of cracking in cladding remote from welds should be addressed. SS cladding may have regions of low delta ferrite that have been sensitized during PWHT and thus susceptible to IGSCC; ASME Sect. XI requires inspection of weld and weld regions (SI S-1).

Open issue #1, (S-V-38), is judged to be resolved for the class 1 piping and associated components. For IGSCC to occur in austenitic stainless steel, three things must be present: a susceptible material, stress approaching or exceeding yield strength, and an aggressive environment such as an oxidizing environment. In the absence of one of the three above conditions, IGSCC will not occur. The stress in the class 1 piping and associated components will not approach or exceed yield strength during shutdown. The report cites steps taken by Westinghouse (pages 92 and 93) to eliminate or reduce the susceptibility of class 1 piping and associated component materials to sensitization and from coming in contact with an aggressive environment. The efficiency of this practice in the prevention of IGSCC has been demonstrated by years of operating experience without exhibiting IGSCC in the class 1 piping and associated components. Therefore, an additional program to manage the aging effects from IGSCC is not necessary for the class 1 piping and associated components because IGSCC should not be an issue.

Open issue #2, (SI S-1), does not apply to the WCAP-14575 report. The scope of WCAP-14575 does not include any class 1 piping and associated components that have cladding material.

Request for Additional Information

10. The industry has experienced cracking of thermal sleeves. Provide an assessment for the cracking of thermal sleeves.

Response:

There are five designs for the thermal sleeves which were installed in some of the branch connection nozzles in the Reactor Coolant Loop. These five designs are numbered 0 thru 4 where 0 is referred to as the original design, and 1 thru 4 are referred to as design generations 1 thru 4, respectively. Figures 2-11 and 2-12 show the thermal sleeve design configurations for large and small bore nozzles. Table 2-1 identifies the thermal sleeve design generations used for each plant and also identifies the plants that did not use thermal sleeves.

In 1977, the attachment welds anchoring the thermal sleeve to the 3" charging nozzle failed on the Farley Unit 1 plant during hot functional testing. The thermal sleeve attachment weld failure was through the attachment or anchor welds on the thermal sleeve which was a design generation 3. This industry issue is identified in Table 3-1 by document IN 82-30 on page 78 of the report.

Westinghouse investigated this problem and concluded that no safety concerns relative to loose or missing thermal sleeves were identified. Based on the evaluations performed, the probable cause, of the operating plant thermal sleeve attachment weld cracks, was high cycle fatigue resulting from flow induced vibration. The generic analyses indicated that the nozzle integrity was not expected to be compromised by the loss or removal of the sleeves. And, for the original design and design generations 1 and 2, there was no obvious cause for concern.

While no safety concern had been identified, the potential financial and plant availability exposure which could result from the existence of migrating thermal sleeves in the primary reactor coolant system were recognized. Therefore, Westinghouse suggested that those utilities with thermal sleeve design generations 3 and 4, should remove them at the next convenient opportunity. For plants under construction, Westinghouse issued field change notices to remove the thermal sleeves.

Thermal transient stresses are considerably higher in the nozzles with thermal sleeves removed, and in each case, the design basis for the plant was revised to show acceptability. Typically, Westinghouse has been able to show that the nozzles are still acceptable with thermal sleeves removed.

Since this issue has been resolved in the past, and included in the design basis, it is not judged necessary to re-address the issue for the aging management report.

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STATUS OF PVRC EVALUATION OF LWR COOLANT ENVIRONMENTAL EFFECTS ON THE S-N FATIGUE PROPERTIES OF PRESSURE BOUNDARY MATERIALS

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ABSTRACT

The Pressure Vessel Research Council (PVRC) has made a concerted effort in the past several years to compile and evaluate test data related to the effects of light-water reactor (LWR) coolant environments on the fatigue behavior of structural materials used in LWR pressure boundary applications. This paper presents the status and findings-to-date of the part of this PVRC effort concerned with effect of the LWR environment on S-N fatigue behavior. The overall purpose of this activity is to formulate recommendations to the ASME Code for methods and procedures to include any needed considerations of the coolant environment in LWR design.

A large amount of test data has been collected and analysis of this database shows that some combinations of environmental and mechanical test conditions can result in reduced S-N fatigue life of ferritic and austenitic steels compared to an air environment. The extent of reduction depends on the values of the influencing variables which include the dissolved oxygen content, the temperature and possibly the flow velocity of the coolant water, and the amplitude and the rate of cyclic straining. In addition, for ferritic steels, the sulfur content of the material may be another factor. Independent "screening" values of these variables for which environmental effects are deemed acceptable have been defined and are discussed.

An important need is the modeling and characterization of the environmental effects when conditions exceed the independent screening values and some examples of this effort are presented. In spite of the large amount of collected data, there are several areas of incomplete definition and the need for test data are noted. In addition, ASME Code implementation includes need for other analyses and test data to formulate design methods for conditions such as variations in strain rate and temperature during a cyclic transient.

INTRODUCTION

Beginning in early 1992, the Pressure Vessel Research Council (PVRC) has been engaged in the compilation, analysis, and evaluation of S-N fatigue data for tests conducted in water that is similar to or simulates light-water reactor (LWR) coolant water chemistries. This activity was prompted by results from a number of tests from laboratories in several countries which were reported in the few years preceding 1991. Although there had been papers in the preceding 25 years or so discussing the effect of LWR-type coolant water on fatigue behavior, the more recent results tend to show fairly large reductions in S-N cyclic life for some combinations of mechanical and water chemistry conditions. It may be noted that the earlier investigations focussed on the effects of the water environment on fatigue crack growth properties with less emphasis on S-N fatigue life behavior.

Because of the potential impact that the more recent S-N fatigue results could have on the fatigue design basis in Section III of the ASME Boiler and Pressure Vessel Code (ASME Code) and consequently, on pressure boundary integrity of LWRs, the ASME Code's Board on Nuclear Codes and Standards (BNCS) requested PVRC for an evaluation of the available information. Specifically, the BNCS request to PVRC in June 1991 was in summary: *"BNCS looks to PVRC to obtain, characterize and report in sufficient detail to ASME such data as may be useful to ASME in its evaluation of the fatigue curves of Sections III and XI."*

PVRC's response to the BNCS request initially included several actions. Administratively, a steering committee and three working groups (WGs) were formed to coordinate the activities. The three WGs and their scopes are:

WG on S-N Data Analysis — To collect, compile, and analyze S-N data and make recommendations for changes in fatigue design curves to the Section III of the Code.
WG on da/dN Data Analysis — To collect, compile, and analyze da/dN data and prepare fatigue crack growth curves for Section XI and other sections of the Code.
WG on Evaluation Methods — To conduct in-depth review of fatigue design criteria and methods in Section III of the Code and make recommendations for changes and improvements.

This paper focuses on the status and preliminary findings of the WG on S-N Data Analysis. The WG on da/dN Data Analysis was formerly a Materials Properties Council (MPC) activity and the results of their effort have been regularly presented to ASME Code's Section XI and in ASME PVP Conference papers. The activities of the WG on Evaluation Methods are somewhat more longer range and it is currently formulating preliminary findings and position statements on various items related to fatigue design procedures.

SUMMARY AND FEATURES OF THE S-N DATABASE

In early 1992, PVRC organized a workshop consisting of a number of contributions by experts and investigators in various aspects of fatigue and related environmental effects. The information presented has been published in Reference 1 (1992). One of the purposes of the workshop was to determine the potential worldwide sources of relevant S-N test results and information.

As a result of inquiries and solicitation, a large amount of data totaling nearly 2800 S-N test results in air and water environments has been received and compiled. A listing of the sources of the data and some remarks about the data are presented in Table 1. Summaries of this database categorized by material, test environment, and test parameters for carbon and low alloy steels are presented in Table 2 and in Table 3 for austenitic steels and nickel alloys. The water environments used in these investigations varied substantially. For the purposes of this evaluation, all environments which contained boric acid, lithium hydroxide, and less than 10 ppb oxygen were considered to represent a pressurized-water reactor (PWR) environment. The environments which contained high-purity water were considered to represent boiling-water reactor (BWR) environments. There is not a single water composition which could be called typical of a PWR or a BWR environment. For this reason, environmental variables that have been shown to be important in describing the environmental effects on fatigue are being considered in the development of models.

Although a large amount of data has been collected as indicated by Tables 2 and 3, detailed information about test specimens, test condition, and test materials are missing in several instances. Additionally, it will be evident in later discussions that test data for some vital ranges of variables are not covered in spite of the large database.

EVALUATION OF AIR ENVIRONMENT CARBON AND LOW ALLOY STEEL TEST DATA

The original Section III fatigue design curves were based on a relatively small amount of test data. In the case of carbon and low alloy steels, the data were limited to room temperature tests utilizing either cantilever bending or axial hourglass specimens. Plots of

TABLE 1
SOURCES OF TEST DATA AND REMARKS

COUNTRY	FACILITY	REMARKS
Japan	17 plus laboratories	Nearly all axial specimens tests, some hour-glass specimens; test results compiled in "JNUFAD" database [Reference(2)]
U.S.A.	Argonne Lab.	Axial specimens
	Mat. Eng. Assoc.	Axial specimens
	General Electric	Cantilever bend specimens tested in autoclave connected to Dresden 1 BWR plant
	Babcock & Wilcox	Tubular specimens; as-milled processed ID test surface, fossil plant water chemistry
Germany	Siemens KWU	Test material and water chemistry information incomplete
	MPA Stuttgart	Test material and water chemistry information incomplete
Russia	Promotey Institute	Test material and water chemistry information incomplete

TABLE 2
SUMMARY OF PVRC S-N DATABASE FOR CARBON AND LOW ALLOY STEELS

Material	Test Environment	Number of Data Points	Temperature, °C	Strain Amp., %	Strain Rate, %/sec	Oxygen Content, ppm
Carbon Steel	Air	191	25 to 288	0.11 to 1.78	9.8 to 0.004	
	PWR	45	300	0.11 to 1.27	5.1 to 0.4	
	BWR	221	240 to 300	0.13 to 1.8	20 to 0.0004	[illegible] to 8
Low Alloy Steel	Air	425	25 to 350	0.08 to 8.6		
	PWR	25	288	0.17 to 0.4	0.4 to 0.004	0.003 to [illegible]
	BWR	388	50 to 290	0.15 to 1.22	2.4 to 0.0004	0.005 to 8
Weld Metal	Air	20	25	0.14 to 1.5	1 Hz	
	BWR	41	290	0.16 to 0.8	6.4 to 0.0004	0.2 to 8

TABLE 3
PVRC S-N DATABASE FOR AUSTENITIC STEELS
AND NICKEL ALLOYS

[illegible]	[illegible]	[illegible]	[illegible], °C	[illegible], %	[illegible]	[illegible]
[illegible]	Air	100	25 to 350			
	PWR	90	288	0.25 to 0.6	[illegible]	
[illegible]	Air	13	25 to 288	0.175 to 0.8	0.4	
	PWR	16	288	0.25 to 0.9	[illegible]	0.2
304 SS	Air	234	25 to 350	0.11 to 2.02		
	PWR	102	288 to 350	0.12 to 1.3	[illegible]	0.2 to 0.3
316 SS	Air	205	21 to 650	[illegible]	[illegible]	
	PWR	60	288	0.13 to 1.5	[illegible]	0.2 to 0.3
Alloy 800	Air	34	20 to 427	[illegible]		
310 SS	Air	8	20	0.3 to 4.0		
348 SS	Air	13	427	0.7 to 1.67		
347SS	Air	110	20 to 300	0.2 to 1.3		
[illegible]	Air	205	20 to 427	0.15 to 4.2		

the data and the derivation of the mean life or "best fit" curves can be found in the Code Criteria Document (Reference 3; 1969). Since only room temperature test data were available, adjustment for temperature effects were made through the temperature variation of the elastic modulus in the calculation of the so-called "fictitious stress". The net result of this procedure is that when these mean curves are adjusted to derive total strain amplitude (or range) versus cyclic life at higher temperatures, the curves are shifted upwards with increasing temperature. More recently, the Code curves for austenitic steels and nickel base alloys have been revised based on a expanded database. However, the fictitious stress procedure is still used in Code design analysis.

The PVRC database contains a considerable amount of air environment baseline tests on test materials utilized for water environment tests; these air data have been compared to the Code mean "best fit" curves as well as to other fitted curves. Figure 1 shows a S-N plot of the air test data for carbon steels in the PVRC compilation. (NOTE: For this and other plots in this paper, cyclic life is generally based on life at 25% load drop from a stable hysteresis loop.) It can be noted that in the life range below 100,000 cycles, most of the values are below the ASME mean curve. Part of the reason is that a considerable number of the tests in the PVRC compilation are at higher temperatures up to 288 C (550 F), and the ASME mean curve when adjusted to higher temperatures results in an upward shift as discussed above.

Recently, Argonne National Laboratory (ANL) reported a statistical analysis of much of the database in the PVRC compilation (Keisler, et al., 1994). The analysis provides mean fit results for carbon and low alloy steels for air and water environments and includes the effect of temperature, sulfur content of the test material, and water chemistry variables. The ANL mean for carbon steels in air at 288 C is shown in Figure 1 and it can be seen that it is slightly lower than the ASME mean and provides a better fit to all of the data. However, it may be noted that the low side scatter from the ANL mean can range up to about a factor of 3 on cyclic life. The scatter factor is about 4 to 5 in relation to the ASME mean. These factors will be referred to later, in connection with water environment tests.

Figure 2 presents a similar comparison of air test data in the PVRC compilation for low alloy nuclear pressure vessel steels with ASME and ANL mean curves. In this case, the ASME and the ANL mean curves are very similar in the life range below 10,000 cycles. At higher cycles, the ANL mean is lower than the ASME mean. Similar to the carbon steel data, the low side scatter ranges up to a factor of 4 on life. One purpose of general plots such as Figures 1 and 2 is to identify potential outliers; this has been an active task in the PVRC activity.

Although the data shown in Figures 1 and 2 are quite extensive, they are limited to tests conducted to provide a baseline for materials tested in water environments. Air environment S-N fatigue tests have been conducted by a number of other investigators (Conway and Sjodahl, 1991; Yoshida, et al., 1978; General Electric, 1966 and 1968) in the case of carbon steels. These additional carbon steel data have been examined in the PVRC work and appear to fall into approximately the same scatterband as the data shown in Figure 1. Examination of air test data for low alloy steels, not in the present PVRC compilation, remains to be done.

In summary, a good database of air environment S-N fatigue test results for carbon and low alloys steels at temperatures of interest to LWR applications has been compiled. A remaining task is to determine the best representation of these data for ASME Code purposes.

EVALUATION OF WATER TEST DATA FOR CARBON AND LOW ALLOY STEELS

The results of the examination and analysis of the compiled data for S-N fatigue tests on carbon and low alloy steels conducted in LWR-type water environments show that the severest detrimental effects on cyclic life occurs when the test conditions involved a combination of certain factors:

- High test temperatures but in the range of normal LWR coolant temperatures
- High dissolved oxygen content in the water, higher than normal LWR operating conditions
- Slow cyclic strain rate, i.e., low frequency
- Strain amplitudes (range) involving plastic strains
- Relatively high sulfur content in the test material

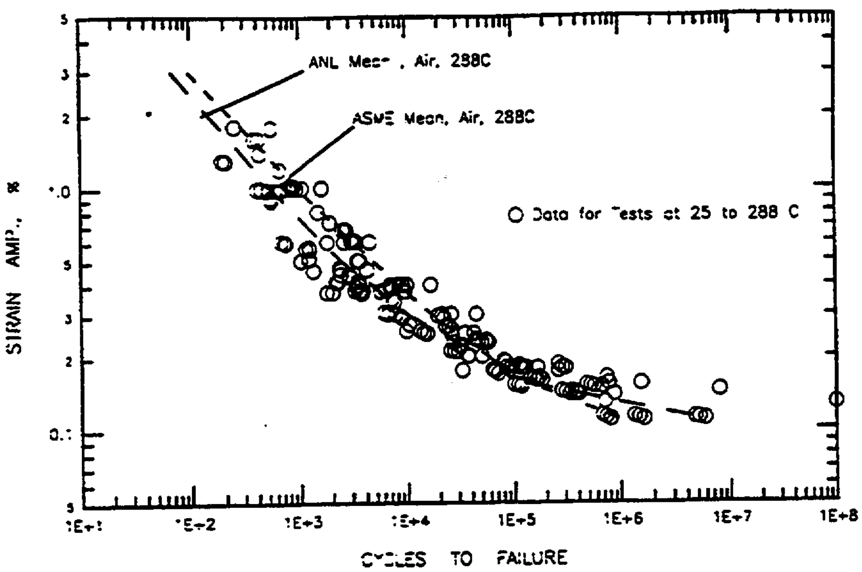

FIGURE 1. COMPILATION OF AIR TEST DATA FOR CARBON STEELS AND COMPARISON TO ASME AND ANL MEAN FIT CURVES

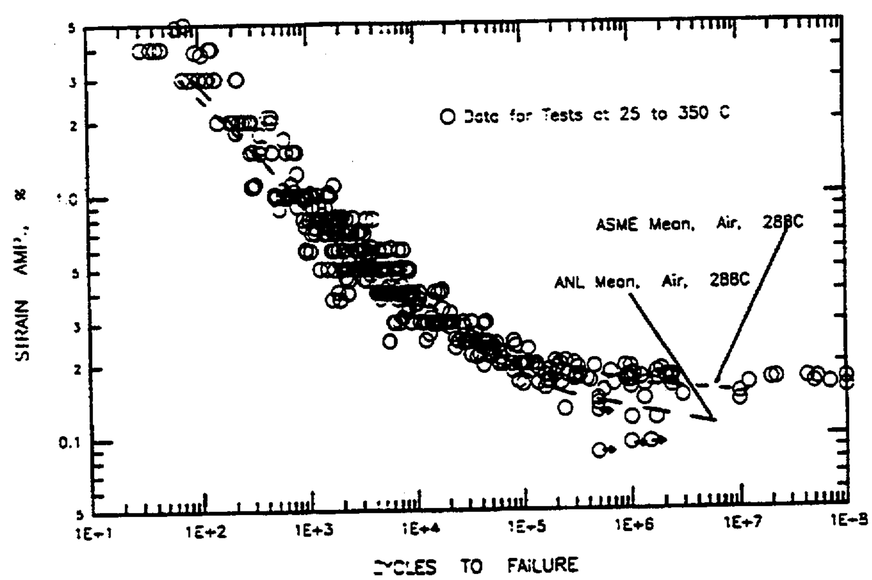

FIGURE 2. COMPILATION OF AIR TEST DATA FOR LOW ALLOY STEELS AND COMPARISON TO ASME AND ANL MEAN FIT CURVES

Although the detrimental effect on S-N life can be large for the worst combinations of these test conditions, these worst-case combinations are generally not typical of LWR operating conditions. The combination of very low strain rates and the relatively large strain ranges that result in large environmental effects do not seem to be typical of events in operating plants. In addition, the high oxygen levels at which much of the data have been obtained are above the levels typical of BWR plants. Therefore, one of the tasks in the PVRC activity consisted of defining a tentative set of criterion values for test and material parameters where the environmental effects would be expected to be moderate or acceptable. This required quantification of "moderate" or "acceptable" environmental effects with respect to the air environment data used in the development of the ASME Code fatigue design curves. Recalling that the analysis of the collected air environment test data indicated a factor of about 4 for temperature and data scatter effects, a factor of 4 on the ASME mean life was chosen as a working definition of "moderate" or "acceptable" water environment effect.

Based on examination of the database, it was determined that values of independent parameters as listed in Table 4 should result in only a moderate detrimental effect on cyclic life. It should be noted that independent means that only one criterion needs to be satisfied, regardless of the values of the other parameters. It has been observed that in order to have a large effect of the environment on the S-N fatigue life, a critical combination of conditions is necessary. If any one of the conditions is missing, the effect of the environment on the fatigue life will be moderate. For example, if the strain rate is greater than 0.1% per second, only a moderate environmental effect is expected even if the dissolved oxygen is high, the temperature is 288 C (550 F), and the material has a high sulfur content. Additional discussion of the selection of the values for the independent criterion has been presented by Van Der Sluys (1993). A major task of the PVRC WG on S-N data analysis is the validation of each of the criterion listed in this table.

TABLE 4
VALUES OF INDEPENDENT PARAMETERS FOR ACCEPTABLE OR MODERATE ENVIRONMENTAL EFFECTS ON THE S-N FATIGUE LIFE OF CARBON AND LOW ALLOY STEELS

Strain Amplitude	≤0.1%
Strain Rate	≥0.1%/sec
Dissolved Oxygen	≤0.1 ppm
Temperature	≤150°C
Sulfur in Steel	≤0.003%
Water Flow Velocity	>3 m/sec

A plot which examines the validity of the values derived for each of the independent criterion for moderate environmental effects for carbon and low alloy steels is shown in Figure 3. It can be seen that a factor of 4 on the ASME mean curve encompasses a large portion of the data for tests which meet any one of the independent criterion value. Although not shown in the figure, a factor of 5 would encompass virtually all of the data. It may be noted that although Figure 3 shows only the ASME mean curve for carbon steel reduced by a factor of 4. Reference 3 shows that the ASME mean curves for carbon steels and for low alloy steels are very similar. Thus, the carbon steel curve with the factor of 4 in Figure 3 would also apply to low alloys steels.

Another consistency check of the criterion values for moderate environmental effects can be made using the ANL statistical analysis model (Keisler, et al.; 1994) mentioned earlier. This model when calculated for parameter values of 0.1 ppm dissolved oxygen, 288 C, 0.015% sulfur, and 0.001%/sec strain rate results in a mean life curve which is approximately a factor of 4 on life reduced below the ANL mean air curve at 288 C. In this case, the 0.1 ppm dissolved oxygen is the governing independent criterion.

With two exceptions, an adequate validation of the values of independent criterion for moderate environmental effects has been found. The exceptions involve the sulfur content and the flow velocity criteria. Definitive experimental data for the effects of these two variables on S-N behavior are lacking. However, results of fatigue crack growth tests involving these variables indicate that low sulfur content in the test material or high flow velocities significantly diminish crack growth rate. The values adopted for the independent criterion for S-N behavior are based on fatigue crack growth results and do require confirmation.

EVALUATION OF WATER ENVIRONMENT TEST DATA FOR AUSTENITIC STEELS AND NICKEL ALLOYS

Table 3 shows that the PVRC database contains quite a large number of tests in LWR-type water environment on austenitic steels and nickel-base alloys. Evaluation of the results for these materials has lagged in the PVRC effort because of the greater apparent concerns about the behavior of carbon and low alloy steels.

Figure 4 provides an overview of the water environment test results for base metal and welds of nickel-base Alloy 600 and base metals of Types 304 and 316NG (a low carbon, nitrogen-added 316) austenitic stainless steels. Except for a few results, the data are above the ASME design curve despite the fact that the test conditions include high oxygen contents and slow strain rates. Also, the Alloy 600 weld metal appears to behave similar to the base metal. The evaluation of the data has not proceeded to the point of establishing independent criterion values where only moderate environmental effects are observed as for ferritic steels.

In austenitic stainless steels, a metallurgical phenomenon of sensitization can occur when the material is held at intermediate temperatures or in weld heat affected zones (HAZs). Sensitization is known to aggravate stress corrosion cracking of austenitic stainless steels in high oxygen LWR water. The PVRC compilation includes water environment S-N test results for sensitized austenitic stainless steels and the data are shown in Figure 5. The results for sensitized 304 SS in this figure clearly show that sensitization can aggravate the environmental effects of high temperature water. In contrast, the results for 316NG shows very little effect and are within the scatterband of results in Figure 4. There are at least two metallurgical reasons for the difference. These are that 316NG has a lower carbon content and the sensitizing heat treatment applied was a 2-hour holding time compared to 10 hours for the 304 material. Both factors would result in greater sensitization in the 304 material and presumably greater sensitivity to a water environment. As mentioned before, weld HAZs can also become sensitized but there are no data available to determine the behavior of typical weld HAZ in LWR-type water environments.

STATISTICAL ANALYSIS OF ENVIRONMENTAL EFFECTS

The Argonne National Laboratory (ANL) statistical analysis (Keisler, et al.; 1994) discussed earlier predicts mean S-N fatigue life curves for carbon and low alloy steels for air and water environments and includes the effect of temperature, sulfur content of the test material, and water chemistry variables. The mean curve predicted for a BWR environment (300 C, 200 ppb oxygen) is compared with data from the database at four strain rates on carbon steel in Figures 6 through 9. The PVRC database includes Babcock & Wilcox (B&W) test results that are not in the database used by ANL in the statistical analysis. These data are included in the data plotted in these figures. There are 28 data points from B&W in these four figures. They do not stand out or represent outliers. The ANL predicted mean curve in Figure 6 fits the low cycle fatigue data well. It underpredicts the high cycle fatigue limit however similar to the effect observed in the air data. For data at a lower loading frequency presented in Figure 7, the predicted curve is a good representation of the data. For the data at an even lower loading frequency of 0.001%/sec, Figure 8 shows that the predicted curve appears to be closer to a lower bound than a prediction of the mean. In this figure, most of the data are the B&W data in the PVRC database. The prediction at the lowest strain rate for which data are available is shown in Figure 9. The predicted curve fits the data well for data at the low strain rate of 0.0004%/sec. In this figure, one of the three data points is a B&W test result.

In general, the Argonne model appears to predict the mean S-N curves of carbon steel in the BWR water environment quite well. It is a ongoing project to compare the model with other data sets in the PVRC data base.

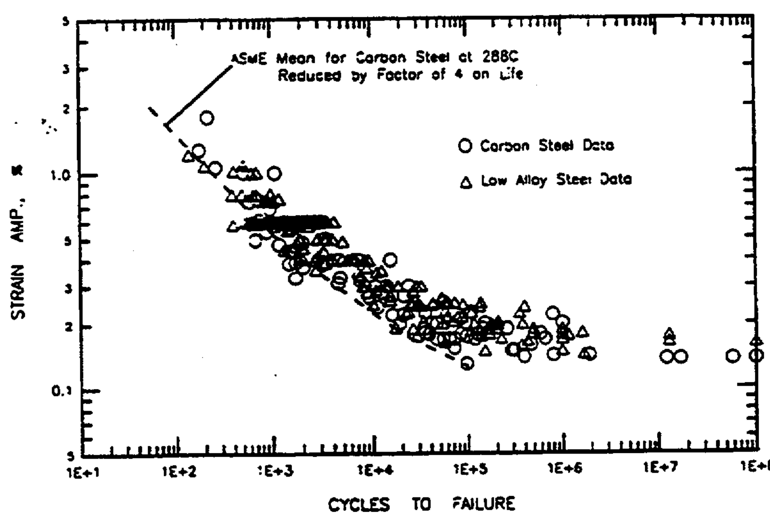

FIGURE 3. COMPILATION OF LWR-TYPE WATER ENVIRONMENT TEST DATA SATISFYING ANY OF THE INDEPENDENT CRITERION FOR MODERATE ENVIRONMENTAL EFFECT AND COMPARISON TO ASME MEAN CURVE REDUCED BY A FACTOR OF 4 ON LIFE

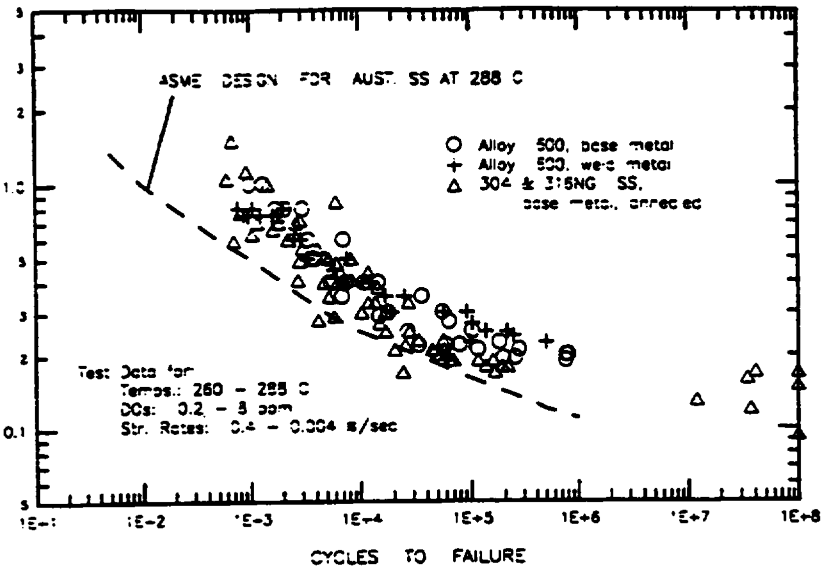

FIGURE 4. LWR-TYPE WATER ENVIRONMENT TEST DATA FOR ANNEALED AUSTENITIC STEELS AND NICKEL-BASE ALLOY 600

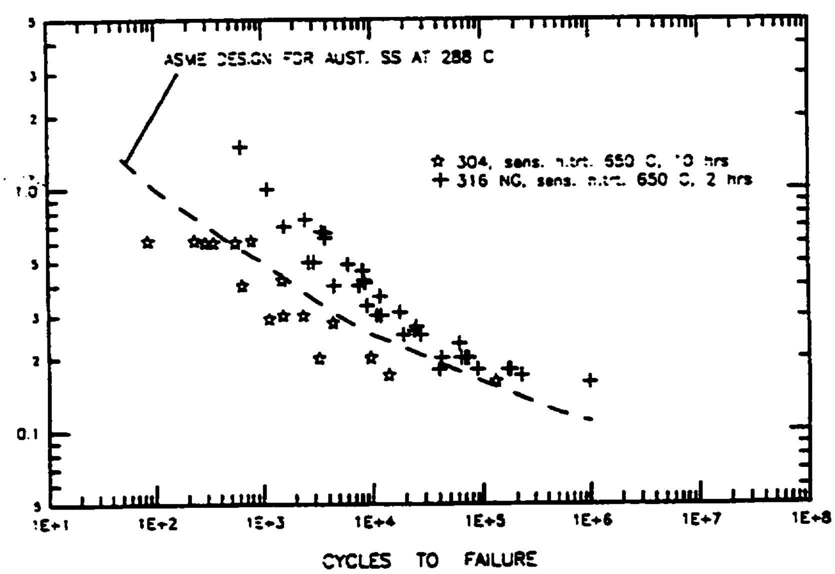

FIGURE 5. LWR-TYPE WATER ENVIRONMENT TEST DATA FOR ANNEALED AUSTENITIC STEELS GIVEN SENSITIZING HEAT TREATMENTS

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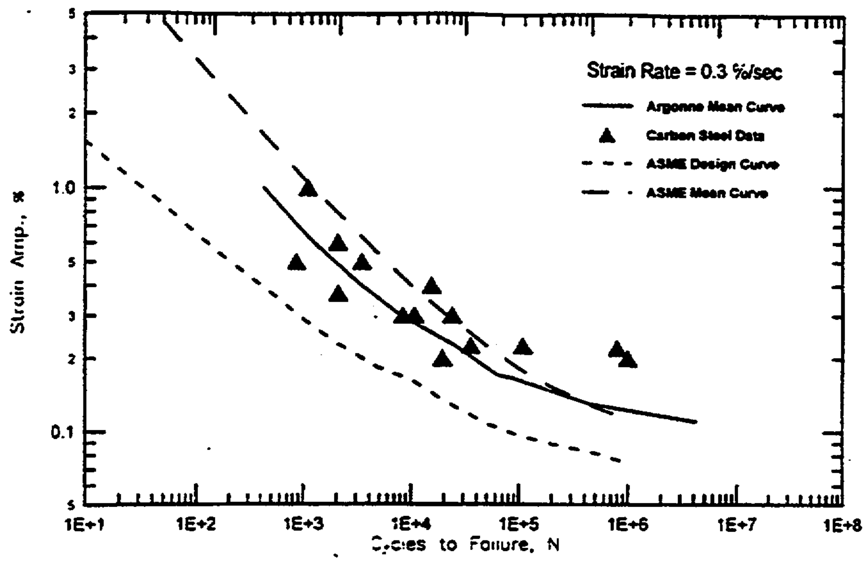

FIGURE 6. DATA FOR CARBON STEEL IN HIGH-TEMPERATURE WATER WITH 200 PPB OXYGEN CONTENT COMPARED WITH THE ASME MEAN AND DESIGN CURVES AND THE ANL MEAN FIT CURVE

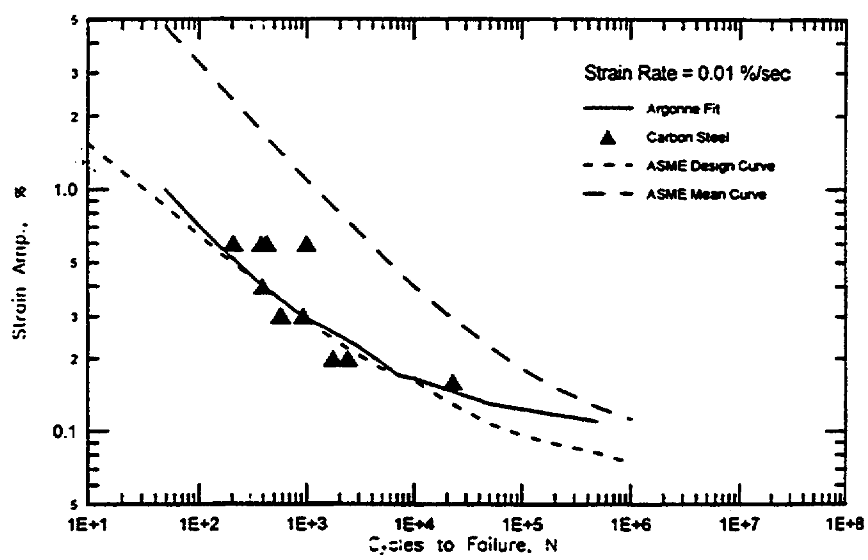

FIGURE 7. DATA FOR CARBON STEEL IN HIGH-TEMPERATURE WATER WITH 200 PPB OXYGEN CONTENT COMPARED WITH THE ASME MEAN AND DESIGN CURVES AND THE ANL MEAN FIT CURVE

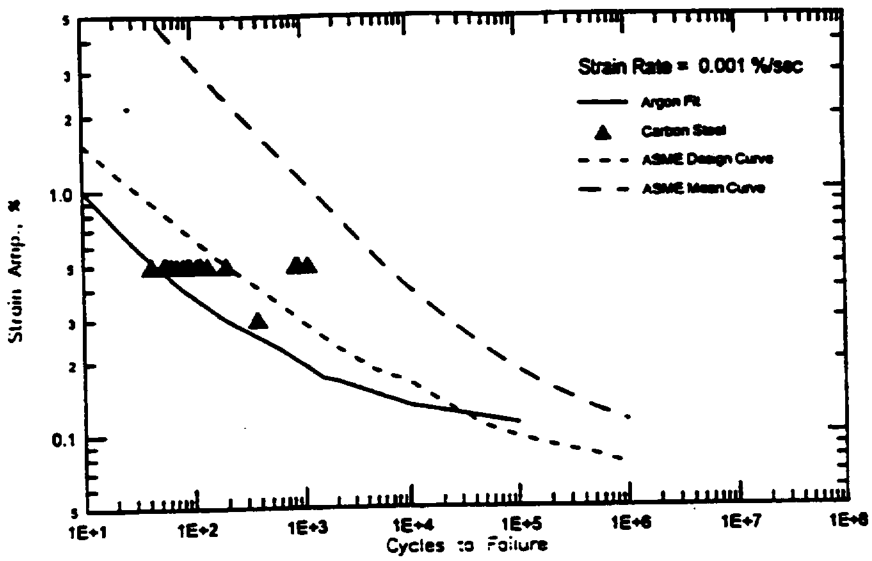

FIGURE 8. DATA FOR CARBON STEEL IN HIGH-TEMPERATURE WATER WITH 200 PPB OXYGEN CONTENT COMPARED WITH THE ASME MEAN AND DESIGN CURVES AND THE ANL MEAN FIT CURVE

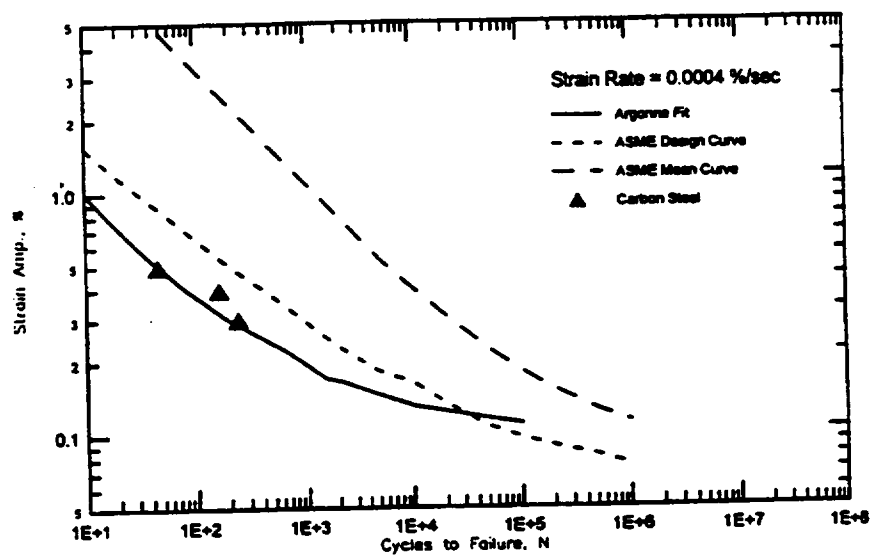

FIGURE 9. DATA FOR CARBON STEEL IN HIGH-TEMPERATURE WATER WITH 200 PPB OXYGEN CONTENT COMPARED WITH THE ASME MEAN AND DESIGN CURVES AND THE ANL MEAN FIT CURVE

55

ASME CODE IMPLEMENTATION CONSIDERATIONS

As mentioned earlier, the BNCS request to PVRC was to evaluate the results and make recommendations for ASME Code implementation if warranted. The following discusses some of the considerations that will need to be included in the recommendations.

"Crack Initiation" and Code Design

PVRC has had extended discussions of "crack initiation" in S-N fatigue tests especially in the low cycle regime and its relevance to the ASME Code design curve. Although the design curve includes a factor of 20 on mean life in the low cycle regime, some hold the view that a crack is "initiated" when the imposed number of cycles exceeds the design curve cycles. Other opinions contend that the mean cyclic life is an indicator of crack initiation. This difference is compounded in a water environment test where the question is whether the decrease in cyclic life is attributable to earlier crack initiation or to faster crack growth, or both.

Some information about the influence of the water environment on this point can be inferred from cyclic stress-strain data. This refers to the peak tension and compression loads in the cycle in a strain controlled fatigue test. The usual practice is to use the loads at one-half of the cyclic life to construct a plot of the stress at half-life as a function of the strain amplitude. Figure 10 shows such data for three carbon steels tested at 288 C in the PVRC compilation. A couple of features can be noted. For the results shown in the figure, the A106B material shows higher stresses than the A333, Grade 6 or the A508, Cl 1 materials. This is an indication of greater cyclic strain hardening in the A106B test material relative to the other two test materials. More important however, there seems to be no systematic indication that the stresses in a water environment test are different from those in an air environment. This suggests that the "crack initiation" event is not markedly different in cycles between the two environments. If it were, a difference in the stresses could be expected.

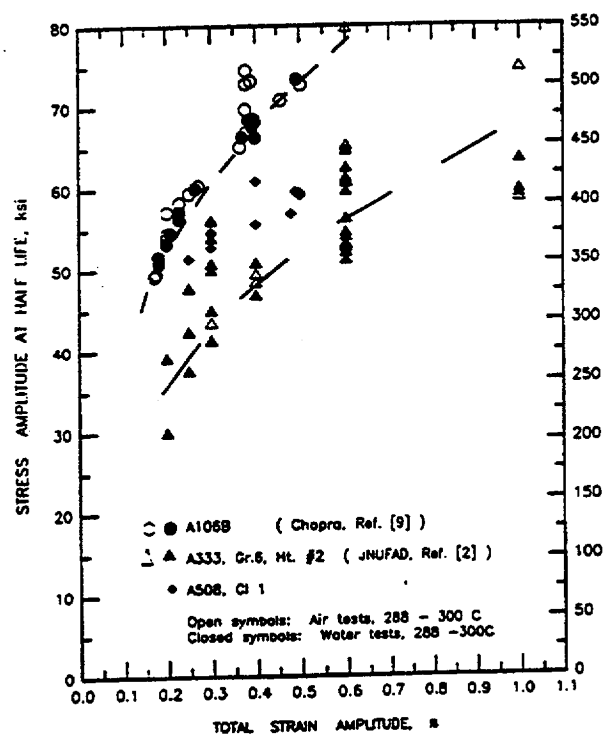

FIGURE 10. CYCLIC STRESS-STRAIN DATA AND TREND CURVES FOR SEVERAL CARBON STEELS IN AIR AND LWR-TYPE WATER ENVIRONMENTS

Design Margins and Statistical Considerations

The fatigue design curves for ferritic steels in Section III of the ASME Code contain the well-known factors of 2 on stress or 20 on cyclic life relative to the mean life curves. However, for the results of statistical analyses such as the ANL formulation, the relation between the traditional factors and the statistical results requires additional consideration. The assessment needs to include the fact that the factor of 20 includes a factor of 4 intended to account for "surface finish, atmosphere, etc." according to Cooper in Reference 1 as well as the fact that environmental effects are a multivariable situation where the significance of global variance and confidence limits is not clear. The PVRC effort in the Working Group on Evaluation Methods is developing guidance on this issue.

Test Data for Design and Operational Transients

All of the test results discussed so far were obtained under relatively constant test conditions such as constant strain rate during the cycle and constant water temperature. However, transient events in operating plants often involve time-varying conditions resulting in varying strain rates and temperatures. For Code implementation, it will be necessary to develop rational procedures to define the applicable parameter values for time-varying actual and design conditions. Several investigators have recently initiated environmental fatigue tests to study these questions and PVRC will assist in the evaluation of the results.

Another design consideration is the evaluation of mean stress effects. The ASME Code provides for this effect in the current design curves. However, S-N behavior in a water environment for conditions of relatively small cyclic stresses but in the presence of high mean stresses is unknown. Currently, it is assumed that the effect is small because the cyclic strain amplitudes are small and in the range where environmental effects are small to moderate without the mean stress, but this assumption requires verification tests when mean stresses are present.

DATA NEEDS

Although a large amount of test data has been collected and evaluated, the environmental effects cannot be definitively characterized due to the lack of some critical data. This situation is understandable in view of the number and range of variables that are involved. Several instances of need for additional tests and data have already been noted. For convenience, these and other areas where additional information would result in more definitive conclusions are listed below:

- Effect of flow velocity on S-N behavior: if verified, this effect would be very significant in defining the relevance of laboratory test results to operating plants
- S-N properties of austenitic stainless steels in PWR primary coolant chemistry water
- S-N properties of carbon steels in BWR coolant water containing 100 to 200 ppb oxygen and at 150 to 250 C
- S-N properties of austenitic, carbon, and low alloy steel weld metals in representative LWR coolant water; also, the properties of weld HAZ for austenitic steels
- More information on the relationship between sulfur content and environmental effects for low alloy and carbon steels
- Environmental effects for high mean stress (R ratio) at low strain amplitudes (or range).

Studies on some of these needs are underway at several laboratories but a long time will be required for results considering long times required for many of the tests and the complexity of the data needs.

SUMMARY AND CONCLUSIONS

The PVRC effort over the past several years on evaluating the effect of LWR-type coolant water on the S-N fatigue properties of pressure boundary materials has resulted in the following accomplishments and tentative findings:

- A large number of S-N fatigue results for tests conducted in baseline air environment and in water environments of various chemistries have been collected and compiled.
- Moderate to large reductions in S-N life relative to life in air environment tests can occur for some combinations of water chemistry, mechanical test parameters, and material characteristics; however, the range of combinations resulting in large effects are generally not typical of operating LWR plants.
- A set of independent "screening" values which define conditions where the environmental effects would be moderate or acceptable have been formulated and partially validated by the available test data.
- The statistical model recently developed by Argonne National Laboratory (ANL) appears to have reasonably good capability of correlating the results of laboratory tests conducted on carbon and low alloy steels in various combinations of water chemistries, mechanical test parameters, and material sulfur content, and for predicting the mean S-N life for these test conditions.
- Design evaluation of the environmental effects of LWR coolants will require additional studies and testing to relate statistical analysis results to design margins, to develop design procedures for design and plant operating events that have varying strain rates and temperature conditions during cyclic transients, to define the effect of high mean stresses, and to obtain additional S-N

data in certain critical ranges of water chemistries and temperatures.

ACKNOWLEDGMENTS

The authors wish to acknowledge the fruitful and informative discussions with many PVRC members and participants on the topic of this paper. We would also like to thank the Japanese Thermal and Nuclear Power Engineering Society for the donation of their test results. However, it should be noted that the interpretations and discussions presented in the paper are those of the authors and do not necessarily represent PVRC statements.

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NRC Project Number 686
WCAP-14575

OG-97-101

September 30, 1997

To: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Westinghouse Owners Group
Modified Response to NRC Request for Additional Information on WOG Generic Technical Report WCAP 14575, "License Renewal Evaluation: Aging Management for Class 1 Piping and Associated Pressure Boundary Components" Number 2(d)

References: 1) NRC letter dated April 18, 1997 from P.T. Kuo to R.A. Newton, Westinghouse Owners Group
2) Westinghouse Owners Group letter, OG-97-060, June 13, 1997 (Response to RAIs)

Attached is a modified Westinghouse Owners Group (WOG) response to Request for Additional Information Number 2(d) on WCAP-14575, "License Renewal Evaluation: Aging Management for Class 1 Piping and Associated Pressure Boundary Components." The modification is based on a vote taken at a meeting of the PVRC Working Group on S-N Curve Data, which was discovered after the original WOG response was provided in OG-97-060, June 13, 1997. The modification was previously presented to the staff during a meeting between the WOG and NRC on July 10, 1997.

Please distribute this response to the appropriate people in your organization for their review.

If you have questions on this modified response, please contact Frank Klanica, Westinghouse, at (412) 374-6392, Charlie Meyer, Westinghouse, at (412) 374-5027, or myself at Wisconsin Electric Power Company, (414) 221-2002.

Very truly yours,

Roger A. Newton

Roger A. Newton, Chairman
LCM/LR Working Group
Westinghouse Owners Group

cc: R. Anand, USNRC, (1L, 1A)
LCM/LR Working Group (1L, 1A)
Steering Committee (1L, 1A)

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REQUEST FOR ADDITIONAL INFORMATION
WESTINGHOUSE OWNERS GROUP TOPICAL REPORT WCAP-14575
"License Renewal Evaluation: Aging Management for Class 1 Piping and Associated Pressure Boundary Components"

Request for Additional Information # 2(d)

Table 4-8 lists parameters developed by the PVRC to identify components where the environmental effect on fatigue life are not considered significant. Describe the test data used to establish the criteria for water flow velocity.

Initial response (in OG-97-060, June 13, 1997):

The criteria for water flow velocity in Table 4-8 will be corrected to show >3 m/sec (9.843 ft/sec) instead of >3 m/sec (3.3 ft/sec). The test data used to establish the criteria for water flow velocity should be described in the references listed in reference 43 (attached)

Modified response:

At a meeting of the PVRC Working Group on S-N Curve Data on June 2, 1997, it was decided that there was insufficient data to support a flow velocity threshold and the metal sulfur content threshold was inadequate as a useful discriminator for actual component materials in current applications. Table 4-8 and associated descriptive text will be updated to remove these parameters. The PVRC work on environmental effects will continue to define thresholds based on the remaining four parameters.

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bcc: All get 1L, 1A

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Vattenfall

OG-99-070

NRC Project Number 686

July 19, 1999

To: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: R.K. Anand, Project Manager
License Renewal Project Directorate

Subject: Westinghouse Owners Group
Response to NRC Request for Additional Information on WOG Generic Technical Reports: WCAP-14574, "Aging Management Evaluation For Pressurizer" and WCAP-14575, "Aging Management for Class 1 Piping and Associated Pressure Boundary Components"

Reference: Request For Additional Information (Received from NRC, NRR - Raj Anand via fax 6/4/99)

Attached are the Westinghouse Owners Group responses to the NRC's Request for Additional Information on WOG Generic Technical Reports: WCAP-14574, "Aging Management Evaluation For Pressurizer" and WCAP-14575, "Aging Management for Class 1 Piping and Associated Pressure Boundary Components." Please distribute these responses to the appropriate people in your organization for their review.

If you have any questions regarding these responses, please contact Charlie Meyer, Westinghouse, at (412) 374-5027, or myself at Wisconsin Electric Power Company, (414) 221-2002.

Very truly yours,

Roger A. Newton

Roger A. Newton, Chairman
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cc: R.K. Anand, Project Manager, USNRC License Renewal Project Directorate, (1L, 1A)
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NRC Request for Additional Information on WOG Generic Technical Report WCAP-14574, "Aging Management Evaluation For Pressurizer"

(1) Does any of the applicable plants rely on RCS pressure control function of the pressurizer to prevent or mitigate the consequences of design-basis events? If it does, please do the following:

(a) The report should include this factual information, indicating that RCS pressure control function is an intended function of the pressurizer, per 10 CFR 54.4(a)(1)(iii).
(b) Explain, why the components, such as spray head, which are relied upon to spray subcoooled water inside the pressurizer to control RCS pressure, is not included within aging management review (AMR). The staff believes that such components are passive, and are not subject to replacement based on a qualified life or specified time period.

Response to RAI #1:

There is no safety analysis which utilizes the RCS pressure control functions of the pressurizer (heaters and sprays) to prevent or mitigate the consequences of a design basis event.

Request for Additional Information on WCAP-14575, "Aging Management for Class 1 Piping and Associated Pressure Boundary Components"

(1) In page 27, Section 2.3.2.2, "Branch Line Restrictors," please clarify the following:

(a) Whether the Class 2 pipes, and the flow restrictors in Class 2 pipes are within the AMR.
(b) Explain, why the flow restrictors in early plants may not be applicable.
(c) The report has listed only one intended function for flow restrictors, which is the pressure boundary function, per 10 CFR 54.4(a)(1)(i). However, the report also indicates that the 3/8-inch flow restrictors are relied upon to limit mass flow rate during postulated breaks. Explain, why the intended function of flow restrictors to prevent or mitigate the consequences of design-basis events, per 10 CFR 54.4(a)(1)(iii), was not identified as an intended function relevant to AMR. Recall that the rule requires one to demonstrate that the effects of aging must be adequately managed so that all the intended functions of a component will be maintained consistent with the CLB for the period of extended operation. Therefore, all the passive intended functions per 10 CFR 54.4(a) should be specifically listed in the report.

Response to RAI #1

(a) WCAP-14575 covers only the Class 1 piping and those flow restrictors installed in Class 1 piping. Class 2 piping and flow restrictors installed in Class 2 piping are not included.
(b) The wording used in the report which suggests that "flow restrictors in early plants may not be applicable" is included in a sentence which qualifies that statement to early plants that were not covered by safety classifications. In the context of preceding sentences, the ability to downgrade safety classification from Class 1 to Class 2 downstream of an installed flow restrictor is not applicable to these early plants, since they are not covered by this safety classification protocol.

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(c) The report states that "restrictors limit the maximum flow through a broken line to a value below the makeup capability of the CVCS." Therefore, any line break downstream of a flow restrictor is not a design basis event, because of this design feature. The absence of a design basis event eliminated Part 54.4(a)(1)(iii) as a reason for including this flow restrictor function as an intended function, i.e., a design basis event could not be prevented or mitigated because there is no design basis event.

This interpretation of Part 54 has been modified since the report was written. Section 2.3.2.2 and the "summary" sections will be modified to identify "limit flow due to a downstream break to a value less than the normal RCS makeup capability" as an intended function of the flow restrictors. Because the flow restrictor forms an integral part of the piping where it is installed, subsequent discussion on aging effects and aging management for the piping are applicable also to the flow restrictors.

Request for Additional Information #2

(2) In page 43, Section 2.3.2.4, "Thermal Barrier and RCP Seals," the report states that, "....the RCP seals are a replaceable component and, as such, are exempt from license renewal." The staff disagrees with this conclusion because in accordance with 10CFR54.21(a)(1)(ii), just being a replaceable component does not qualify it to be exempt from AMR. The rule states that in order to be exempt from AMR, a component must be replaceable based on a qualified life or specified time period. Therefore, the report should also provide a specified time period of replacement for the RCP seals, as required by the license renewal rule.

Response to RAI#2

The intent of the wording in Section 2.3.2.4 was to explain that the RCP seals do not require an AMR for the purposes of license renewal.

Section 3.1.6 discusses the operating and maintenance experience relating to RCP seals, and states "The pump seals are considered part of the overall active function of the pump. This issue is not a licensing renewal concern because pump seals are part of a preventive maintenance (replacement) program."

Although the rule requirement for exemption from an AMR is quite explicit, the Statements of Consideration to Part 54 does allow for an applicant to provide site-specific justification in an application for excluding components that are replace based on performance or condition monitoring from an AMR.

RCP seals are a highly visible and closely monitored element of the reactor coolant system. Unlike other parts of the system, they do not maintain a pressure boundary, but rather allow controlled leakage which is acknowledged in plant Technical Specifications. This leakoff is closely monitored in the control room, and a high leakoff flow is alarmed as an abnormal condition, requiring corrective action. Certain parts of the RCP seal "package" (e.g., backup seals) are subject to wear, and these parts are routinely replaced, as are installed o-rings. The main RCP seal is routinely inspected during plant outages based on manufacturer's recommendations, and is replaced based on either the results of that inspection, or on leakoff performance during operation. The RCP seal was never intended to be a long-lived (life of the plant) component, although the specific time period for replacement of the seals will vary between plants, depending on individual operating practices and experience. The usual period ranges between 3 and 6 fuel cycles of operation.

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Based upon consideration of the RCP seals as an active component of the pump, or upon consideration of their periodic replacement, the conclusion that the seals do not require an explicit aging management review remains valid.

The wording "a replaceable component" in Section 2.3.2.4 will be changed to "an active component which is subject to replacement based on performance."

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