Westinghouse Non-Proprietary Class 3

WCAP-14422
Rev. 2-A

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Licensing Renewal Evaluation: Aging Management for Reactor Coolant System Supports

Westinghouse Energy Systems

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-14422
Rev. 2-A

LICENSE RENEWAL EVALUATION: AGING MANAGEMENT FOR REACTOR COOLANT SYSTEM SUPPORTS

Revision 2 - A
December 2000

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Funded by:

Westinghouse Owners Group
Life Cycle Management/License Renewal Program
and Electric Power Research Institute

Approved: ______________________________
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Prepared by Westinghouse Electric Company for use by Members of the Westinghouse Owners Group. Work performed in Shop Order MUHP-6114 under direction of the WOG LCM/LR Program Core Group and Shop Order EPGP-3075E for the Electric Power Research Institute.

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

November 17, 2000

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

SUBJECT: ACCEPTANCE FOR REFERENCING OF GENERIC LICENSE RENEWAL PROGRAM TOPICAL REPORT ENTITLE, "LICENSE RENEWAL EVALUATION: AGING MANAGEMENT FOR REACTOR COOLANT SYSTEM SUPPORTS", WCAP-14422, REVISION 2, FEBRUARY 1997

Dear Mr. Newton:

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 Reactor Coolant System Supports", WCAP-14422, which the Westinghouse Owners Group (WOG) submitted in February 1997, 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 members' 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 reactor coolant system supports 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-14422 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-14422-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

LICENSE RENEWAL EVALUATION: AGING MANAGEMENT FOR

REACTOR COOLANT SYSTEM SUPPORTS

WESTINGHOUSE OWNERS GROUP GENERIC TECHNICAL REPORT WCAP-14422, Rev. 2

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LIST OF ACRONYMS AND ABBREVIATIONS

ACI	American Concrete Institute
AISC	American Institute of Steel Construction
AMP	aging management program
ARDM	age-related degradation mechanism
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
CASS	cast austenitic stainless steel
CFR	Code of Federal Regulations
CLB	current licensing basis
CUF	cumulative usage factor
EPRI	Electric Power Research Institute
FSE	final safety evaluation
GL	NRC generic letter
GSI	generic safety issue
GTR	generic technical report
IEB	NRC inspection and enforcement Bulletin
IPA	integrated plant assessment
IR	industry report
ISI	inservice inspection
LCM/LR	Life Cycle Management/License Renewal Program
NDE	nondestructive examination
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
PZR	pressurizer
PWR	pressurized-water reactor
RAI	request for additional information
RCP	reactor coolant pump
RCS	reactor coolant system
RG	NRC regulatory guide
RPV	reactor pressure vessel
RVSS	reactor vessel support structures
SCC	stress corrosion cracking
SG	steam generator
SRP-LR	standard review plan for license renewal
SSE	safe-shutdown earthquake
TLAA	time-limited aging analyses
TNP	Trojan Nuclear Plant
USI	unresolved safety issue
WOG	Westinghouse Owners Group

1 INTRODUCTION

Pursuant to Section 50.51 of Title 10 of the Code of Federal Regulations (10 CFR 50.51), the U.S. Nuclear Regulatory Commission (NRC) issues licenses to operate nuclear power plants for a fixed period of time not to exceed 40 years. The NRC may renew these licenses for a fixed period of time not to exceed 20 years beyond expiration of the current operating license. The revised license renewal rule, 10 CFR Part 54 (60 FR 22,461, May 8, 1995), sets forth the requirements for the renewal of operating licenses for commercial nuclear power plants (Ref. 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, as set forth in 10 CFR 54.21(a)(1), requires the applicant to identify and list structures and components that are subject to an aging management review, and 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). In addition, 10 CFR 54.21(a)(3) requires that, for each structure and component identified in 10 CFR 54.21(a)(1), the applicant demonstrate 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, as required by 10 CFR 54.21(c), the application must provide an evaluation of time-limited aging analyses (TLAAs), as defined in 10 CFR 54.3, including a list of TLAAs.

1.1 Westinghouse Owners Group Generic Technical Report

By letter dated July 13, 1995, the Westinghouse Owners Group (WOG) Life Cycle Management/License Renewal (LCM/LR) Program and the Electric Power Research Institute (EPRI) submitted Generic Technical Report (GTR) WCAP-14422, "License Renewal Evaluation: Aging Management for Reactor Coolant System (RCS) Supports," Revision 0 (Ref. 2), for staff review and approval. Subsequently, WOG and EPRI submitted Revision 1 and Revision 2 to WCAP-14422 on March 22, 1996, and March 4, 1997, respectively. The purpose of the report is to provide a technical evaluation of the effects of aging on the RCS supports and demonstrate that the aging effects on the RCS supports within the scope of the report can be adequately managed for the period of extended operation. The report is intended to provide individual WOG-member utility owners with sufficient technical details to support an application for license renewal.

1.2 Conduct of Staff Review

The staff reviewed WCAP-14422 to determine whether it met the requirements set forth in 10 CFR 54.21(a)(3) and (c)(1) for the RCS supports. The staff issued requests for additional information (RAIs) after completing the initial review. WOG responded to the staff's RAIs and subsequently submitted two revisions to the report. A meeting was held on October 3, 1996, between WOG representatives and the staff to discuss various aspects of the response to the RAIs. A telephone conversation between the staff and WOG was held following the meeting and clarified several technical positions in the report, including the effect of aging and aging management of inaccessible areas. This safety evaluation is based upon the staff review of Revision 2 of WOG GTR WCAP-14422. Argonne National Laboratory provided technical assistance in reviewing the WOG report.

2 SUMMARY OF THE GENERIC TECHNICAL REPORT

The WOG report contains a generic evaluation for managing the effects of aging on the RCS supports in facilities owned by WOG members so that the intended functions will be maintained under all design load conditions for the period of extended operation. The evaluation applies to the following WOG member operating plants:

Beaver Valley Units 1 & 2	Braidwood Units 1 & 2
Byron Units 1 & 2	Callaway Unit 1
Catawba Units 1 & 2	Comanche Peak Units 1 & 2
Diablo Canyon Units 1 & 2	Donald C. Cook Units 1 & 2
Farley Units 1 & 2	Ginna Unit 1
Indian Point Unit 2	Indian Point Unit 3
Kewaunee Unit 1	Millstone Unit 3
McGuire Units 1 & 2	North Anna Units 1 & 2

Point Beach Units 1 & 2	Prairie Island Units 1 & 2
Robinson Unit 2	Salem Units 1 & 2
Seabrook Unit 1	Sequoyah Units 1 & 2
Shearon Harris Unit 1	South Texas Units 1 & 2
V. C. Summer Unit 1	Surry Units 1 & 2
Turkey Point Units 3 & 4	Vogtle Units 1 & 2
Watts Bar Unit 1	Wolf Creek Unit 1

The WOG report describes the RCS supports, including support type, configuration, design basis, materials of construction, and environmental loading conditions. The WOG report identifies and evaluates the aging effects which are applicable to these supports and which can ultimately degrade their intended function. The WOG report also identifies and evaluates the TLAAs involving the RCS supports.

2.1 Components and Intended Functions

The supports for the following RCS components are within the scope of the WOG report:

(1) Primary component supports for:

- the reactor pressure vessel (RPV) (note that the neutron shield tank is included in the scope and is described in RPV configuration 4; the support ring is included as described in configuration 3)
- the steam generator (SG)
- the reactor coolant pump (RCP)
- the pressurizer (PZR)

(2) PZR surge line supports, including springs

The boundary between an RCS component support and its supporting structures is defined as follows:

(1) The component support includes the entire support up to, but not including, integral attachments on the component. The integral attachments are described in generic reports on specific components.

(2) Lugs, nozzles, and welds on component shells are not included. They are discussed in generic reports on specific components.

(3) Concrete "local to" an embedment is included, but concrete adjacent to an embedment is covered in the generic report associated with seismic Class 1 structures. Base plates, embedded plates, and anchor bolts are considered part of local embedments and within the scope of this report.

The WOG report excludes the following components and structures:

(1) Pipe whip restraints
(2) Masonry walls
(3) Portions of snubber supports that perform intended functions in an active manner.

The intended function of the RCS supports, as stated in Section 2.1 of the WOG report, is to maintain the RCS components in equilibrium and to maintain structural integrity of the RCS piping and primary components under all plant operation and design conditions.

These supports are designed in accordance with the standards of the American Institute of Steel Construction (AISC) manual and specifications (Ref. 3) or the American Society of Mechanical Engineers (ASME) Code, Section III, Division 1, Subsection NF (Ref. 4). The supports are fabricated from structural plates, shapes, bars, forgings, pipes, and tubes and have welded and bolted constructions. Table 2-4 of the WOG report provides materials used for the primary component supports.

The configurations for the RCS supports are as follows:

- RPV supports—There are four different configurations for the RPV supports. Each RPV has three to six supports, depending on the number of coolant loops employed. These supports and their supporting steel components (steel ring, steel columns, or neutron shield tank) are within the scope of this report. They are designed to provide both vertical and lateral restraint while allowing the RPV to expand and contract during service. Detailed descriptions of these support configurations are provided in Section 2.3.1 of the WOG report.

- SG supports—The SG supports provide vertical and lateral restraint and allow for free thermal expansion and contraction of the SG and the RCS piping. Five different configurations are used for the SG supports. Detailed descriptions of these support configurations are provided in Section 2.3.2 of the WOG report.

- RCP supports—Six different support configurations are identified for the RCP, but only five of the configurations are employed by plants addressed in the WOG report. The sixth configuration was not included for use in Table 2-2 of the WOG report. These supports provide vertical and lateral restraint and allow free thermal expansion and contraction of the RCP and the RCS piping. Detailed descriptions of these support configurations are provided in Section 2.3.3 of the WOG report.

- PZR supports—Three support configurations are used for the PZR. These supports restrain the PZR and also allow free thermal expansion and contraction of the PZR during service. Detailed descriptions of these support configurations are provided in Section 2.3.4 of the WOG report.

- PZR surge line supports—The 12- and 14-inch diameter surge lines that connect the PZR and the RCS hot leg can be supported vertically by structural members or component standard supports. Where required, lube plates are used to assure free movement resulting from thermal expansion and contraction of the surge line piping. The WOG report does not provide a detailed description or configuration of these supports.

The WOG report contains a table (Table 2-2) listing the plant names and the specific support configurations used in each plant for the RPV, SG, RCP, and PZR. The WOG report also states that the support sketches provided represent only some of the actual configurations used.

Furthermore, in response to RAI #10 (WCAP-14422, Rev. 0), WOG stated that there are no supports on the primary coolant loop piping; therefore, the RCS component supports and the PZR surge line supports are the only supports within the scope of the WOG report.

2.2 Effects of Aging

The WOG report contains an evaluation of the applicability of the following aging mechanisms and their associated aging effects on the RCS supports within the scope of the GTR:

Aging Mechanism	Aging Effects
Steel components:	
Stress corrosion cracking (SCC)	Crack initiation, crack growth
Corrosion and aggressive chemical attack	Decrease of strength, loss of materials
Neutron embrittlement	Decrease fracture toughness and ductility
Thermal aging embrittlement	Decrease material toughness
Mechanical wear	Loss of material
Fatigue	Accumulated fatigue damage
Creep and stress relaxation	Deformation
Low fracture toughness and lamellar tearing	Decrease structural integrity

Concrete components and embedments:

Cracking and rebar corrosion	Loss of material, cracking, spalling
Leaching and aggressive chemicals	Loss of materials, cracking, increasing of porosity and permeability
Elevated temperature	Loss of strength and modulus, cracking, scaling
Neutron irradiation	Cracking, loss of strength and modulus

The WOG report briefly discusses the thermal environment and relative humidity inside the containment in which these supports are located. The WOG report also states, "No documentation related to industry operating experience associated with aging has been found for the RCS supports within the scope of this report." The WOG report uses the summary of findings in EPRI report TR-104305 (Ref. 5) to identify aging management issues germane to the RCS supports.

The WOG report concludes that the aging effects associated with neutron embrittlement, thermal aging embrittlement, mechanical wear, fatigue, creep and stress relaxation, and low fracture toughness and lamellar tearing are insignificant for the RCS supports. It also concludes that none of the aging effects caused by concrete degradation mechanisms are significant except the aging effects from neutron irradiation, which will be included in plant-specific evaluations. The WOG report concludes that aging effects requiring an aging management program (AMP) for the RCS supports are caused by aggressive chemical attack, stress corrosion cracking (SCC), and corrosion.

2.3 Aging Management Programs

Section 4 of the WOG report describes the AMP attributes and their effectiveness during the period of extended operation. Since the WOG report is generic to the plants listed, plant-specific aging management activities are not addressed. The plant-specific details of the aging management attributes described in the WOG report will be developed on a plant-specific basis. The WOG report concludes that no new maintenance management programs or inspection activities need to be implemented for the period of extended operation. The WOG report also indicates that any prior commitments by utilities to address recommendations from

Generic Letter (GL) 88-05 (Ref. 6) and Generic Safety Issue (GSI) 29 (Ref. 7) constitute part of the aging management program for the RCS supports. These commitments are part of the CLB and will be extended into the extended period of operation unless modifications are made.

The WOG report contains attributes for three AMPs. These AMPs, when fully developed, will manage the detrimental effects of SCC, corrosion, and aggressive chemical attack on the RCS supports. The WOG report considers that aggressive chemical attack and corrosion have similar degradation effects for the RCS supports, and therefore, addresses them together. The AMPs are as follows:

- AMP-1.1, "Aggressive Chemical Attack and Corrosion for Steel"
- AMP-1.2, "Aggressive Chemical Attack and Corrosion for Concrete Embedments"
- AMP-1.3, "Stress Corrosion Cracking for Bolting"

There are six attributes in each AMP and they are (1) Scope, (2) Surveillance Technique, (3) Frequency, (4) Acceptance Criteria, (5) Corrective Actions, and (6) Confirmation.

The scope describes the components and applicable aging effects; the surveillance technique describes the monitoring, inspection, or testing techniques used to detect aging effects; the frequency describes the time period between program performance or when a one-time inspection must be completed; the acceptance criteria contains the qualitative or quantitative criteria that determine when corrective actions are needed; the corrective actions are the actions to further analyze, prevent, or correct the consequences of the effect; and the confirmation provides the post-maintenance test or other techniques to confirm that the actions have been completed and are effective.

For AMP-1.1, the scope contains the steel supports including embedments and addresses the aging effect of corrosion due to borated or demineralized water. The aging effect reduces load-carrying capacity caused by loss of material and loss of movement caused by roughened surface or corrosion product build-up. The means and methods proposed for the surveillance techniques include (1) examination (inspection) in accordance with the standards of ASME Code Section XI, Subsection IWF-2500 and Table IWF-2500-1 or Subsection IWA 2240

(Ref. 8), (2) leakage identification walkdowns, and (3) leakage monitoring. The examination frequency is in accordance with the standards of Subsection IWF-2410 (Inspection Programs), Table IWB-2412-1, and Subsection IWB-2412. The frequency of leakage walkdown is at each refueling outage, and the frequency of leakage monitoring is "as needed." Acceptance criteria for inspection are specified by Subsection IWF-3410 (Acceptance Standards - Component Support Structural Integrity). The acceptance criterion for a leakage walkdown is the identification of fluids, and the acceptance criteria for leakage monitoring are in accordance with plant-specific leakage monitoring criteria. Corrective actions for consequences of an effect identified by inspection are specified by Subsection IWF-3112 (Acceptance for Preservice Examinations) with Subsections IWF-3200 (Supplemental Examinations) or Subsection IWF-3122 (Acceptance for Inservice Examinations) with Subsection IWF-3200. Corrective actions for consequences of an effect identified by leakage walkdown or leakage monitoring are cleaning and restoration of the affected surfaces, removal of standing fluid, evaluation of boric acid buildup, and identification and repair of leak sources. Confirmation for inspection is provided by Subsections IWF-2200 (Preservice Inspection) following adjustment, repair, or replacement prior to returning the system to service, IWF-2420 (Successive Inspection), and IWF-2430 (Additional Inspection). Confirmation for leakage walkdowns is re-examination of affected surfaces after cleaning or restoration and reexamination at the next outage. Confirmation for leakage monitoring is continuous monitoring.

For AMP-1.2, the scope contains the concrete embedments and addresses the aging effects of acidic solution which reduces strength caused by concrete degradation and rebar corrosion and leaching which reduces strength caused by increased concrete porosity. The surveillance techniques include (1) examination (inspection) recommended by the American Concrete Institute (ACI) standards (Refs. 9 -12), (2) leakage identification walkdowns, and (3) leakage monitoring. The frequency of the surveillance is the same as described in AMP-1.1 except that leakage monitoring is continuous. Acceptance criteria for inspection are specified by ACI recommendations (Refs. 11, 13, and 14). The acceptance criterion for a leakage walkdown is the identification of fluids, and the acceptance criteria for leakage monitoring are in accordance with plant-specific leakage monitoring criteria. Corrective actions for consequences of an effect identified by inspection are in accordance with the recommendations of ACI standards (Refs. 10, 11, 13, 14, and 15) and the standards of ASME Section XI, Subsection IWF-3112 (Acceptance for Preservice Examinations) or Subsection IWF-3122 (Acceptance for Inservice

Examinations). The corrective action for consequences of an effect identified by leakage walkdown or leakage monitoring are removing standing fluid, cleaning and restoration of the affected surfaces, and identification and repair of leak sources. Confirmation for inspection is provided by Subsections IWF-2200 (Preservice Inspection) following adjustment , repair, or replacement prior to return of the system to service, IWF-2420 (Successive Inspection), and IWF-2430 (Additional Inspection). Confirmation for leakage walkdowns is re-examination of affected surfaces after cleaning or restoration and reexamination at the next outage. Confirmation for leakage monitoring is continuous monitoring.

AMP-1.3 contains attributes to manage aging effects for bolts, studs, and anchors. The scope includes the RCS support bolting with nominal diameter greater than one inch and the aging effects of crack initiation and localized cracking failure caused by SCC. The means and methods proposed to detect these aging effects include examination in accordance with the standards of ASME Code Section XI (Ref. 8), Subsection IWF-2500 (Examination Requirements) and Table IWF-2500-1 with Subsection IWF-2520 (Method of Examination) or Subsection IWA-2240 (Alternative Examinations). The frequency of surveillance is set by the standards of Subsection IWF-2410 and Table IWB-2412-1 with Subsection IWB-2412. Acceptance criteria are specified by the standards of Subsections IWF-3410 (Acceptance Standards - Component Support Structural Integrity), IWF-3200 (Supplemental Examinations), and IWA-2000 (Examination and Inspection). Corrective actions include the evaluation and modification of the existing materials and design, or replacement of defective bolts. The confirmation is to re-examine the replaced bolts at the next inspection interval if they are still susceptible to SCC.

2.4 Time-Limited Aging Analysis (TLAA)

Section 3.3 of the WOG report provides an evaluation of the TLAAs involving RCS supports in accordance with the requirements of 10 CFR 54.21(c) (Ref. 1). The report concludes that fatigue is the only aging mechanism associated with a TLAA for the RCS supports. The report also states that "no fatigue calculations have been performed for the RCS supports as part of their design since the number of cycles was much less than 20,000." This conclusion is based on the representative number of loading cycles at stress levels representing normal and upset conditions and comparing this number to the expected number of cycles to failure. The results of this comparison show that the cumulative usage factor (CUF) is 0.088 for a 40-year design life. Extrapolation to 60 years gives a CUF of less than 0.15. Since this is less than the CUF of 1.0 allowed by the ASME Code, the WOG report concludes, in Section 3.2.6, that "fatigue is not an effect that is a concern for the RCS support structures." Similarly, the WOG report asserts in Section 3.2.6, "the concrete embedments that are part of the RCS supports are not subject to high stress and load cycle combinations. . . [therefore], degradation due to fatigue is unlikely."

2.5 Plant-Specific Programs

Section 5 of the WOG report states that the following items are to be addressed by the license renewal applicant as plant-specific programs in their applications:

- Identification and evaluation of any plant-specific TLAAs applicable to their RCS supports.

- Identification and evaluation of current-term programs implemented within the current licensing term to address technical issues from industry practices and United States Nuclear Regulatory Commission (NRC) directives [that] should be continued into the license renewal term. Modifications to or elimination of these programs have to be justified.

- Identification and justification of plant-specific programs that deviate from the recommended AMPs.

- Technical justification for programs that deviate from the 1989 Edition of ASME Section XI and Appendices VII and VIII should be provided in a plant's license renewal application.

- Identification of any specific program necessary to ensure that proper preload is retained for the component supports within the scope of this report.

- Identification of any evidence of aging degradation in inaccessible areas during the current licensing term that is considered to potentially affect system intended functions. A plan of action to address any identified potential degradation should be provided.

- Verification that the plant is bounded by this GTR. The actions applicants must take to verify that their plant is bounded will be described in an implementation procedure.

- Plant-specific evaluation of potential degradation due to irradiation of the components within the scope of this report.

3 STAFF EVALUATION

The staff reviewed the WOG report, WCAP-14422 and additional information submitted by WOG to determine if the WOG report satisfies 10 CFR 54.21(a)(3) for the RCS supports. In doing so, the staff determined whether the AMPs, as described in the report, can adequately manage the effects of aging relating to the RCS supports so that the intended functions will be maintained consistent with the CLB during the period of extended operation. The staff also reviewed the WOG report to determine if the WOG report has adequately addressed TLAAs involving the RCS supports in accordance with the requirements of 10 CFR 54.21(c)(1).

3.1 Scope of Components

The RCS supports addressed and listed in the WOG report are supports for the RPV, SGs, RCPs, PZR, and PZR surge line. As described in Section 2.1 of the WOG report the boundaries of the RCS support are defined so that the supports include the integral attachments, including bolting, base plates, and concrete "local to" an embedment but not concrete adjacent to an embedment. Section 2.1 of the WOG report also lists the components and structures that are excluded from the WOG report, namely, pipe whip restraints (addressed by other generic report) , masonry walls (none related to the RCS supports), and the active portion of snubbers. The WOG did not clearly define the term "local" in its report. However, the aging management programs should be comparable and consistent for all concrete structures and structural components. Since the WOG report does not define the interface between the local and adjacent concrete, the license renewal applicants must describe the aging management program for adjacent concrete structures and any differences from the aging management program for the local concrete structures. **This is Renewal Applicant Action Item 1.**

Section 2.1 of the WOG report states that "the RCS supports for the plants included in this study share commonality of function, yet differ in the details of their design." It further states that "the support configurations and materials of the plants included in this study vary because of the variety of organizations that design supports." Consequently, utilities referencing the WOG report in a license renewal application do not necessarily have the same RCS supports as those described in the report. Therefore, when referencing this report, utilities will have to confirm that the RCS supports in their plants are the same as one of the designs within the scope of this report or provide justifications for any deviations from the referenced design. **This is Renewal Applicant Action Item 2.**

The staff also notes that the WOG report contains the following discrepancies and omissions:

1. Wear plates and bearing pads are included as support components and within the scope of this WOG report but are not identified in Table 2-1 as parts and sub-components requiring an aging management review.

2. Sketches of RCP support configuration 4 and PZR support configuration 2 are not provided in the WOG report.

3. Section 3.2.9 of the WOG report indicates that ASTM A36 steel is used in SG and RCP supports, however, ASTM A36 steel is not included in the list of material for the primary component supports (Table 2-4).

4. The 1963 AISC manual (Ref. 3) states that ASTM A7, A36, A242, A373, A440, and A441 structural steel and ASTM A325 bolts are commonly used for steel construction but they are not listed in Table 2-4 of the WOG report.

5. There are no specific descriptions and sketches for the PZR surge line supports.

A license renewal applicant will have to resolve these discrepancies and omissions in its application. **This is Renewal Applicant Action Item 3.**

3.2 Intended Functions

The intended functions of the RCS supports, as stated in Section 2.1 of the WOG report, are to maintain the positions of the RCS components prescribed by design, and ensure the structural integrity and safe operation of the RCS piping and primary components under all plant design and operating conditions. The staff agrees with the WOG statement of the intended functions of the RC support system.

3.3 Effects of Aging

The effects of aging evaluated in the WOG report are those associated with the aging mechanisms of SCC, corrosion, aggressive chemical attack, neutron embrittlement, thermal aging embrittlement, mechanical wear, fatigue, creep and stress relaxation, concrete degradation, and low fracture toughness and lamellar tearing. The aging effects include loss of material, loss of strength and stiffness, cracking (crack initiation and growth), decreased fracture toughness and ductility, accumulated fatigue damage, and deformation. These aging mechanisms and aging effects are consistent with those listed in Table 3.1-1 of the draft standard review plan for license renewal (SRP-LR) (Ref. 16). The aging effects that are not included in the WOG report are those associated with the aging mechanisms of settlement, abrasion and cavitation, freeze and thaw, reaction with aggregates, corrosion of steel piles, and cathodic protection current. These aging mechanisms are not applicable to the RCS supports. Because the RCS supports are located near the center of the containment mat, settlement will be fairly even and will not cause significant distortion to the RCS supports. Because the RCS supports are inside the containment, freeze-thaw of concrete components will not be a concern; the RCS supports are not exposed to flowing water, so abrasion and cavitation will not occur. The RCS is not supported on piles, so corrosion of steel piles and cathodic protection current are not applicable. Reaction with aggregates is not a problem because none of the concrete components of the RCS supports are exposed to alkalies.

The WOG report addressed those aging mechanisms for their potential applicability to the RCS supports. On the basis of the information published in NRC staff and contractor reports relating to RCS supports (Refs. 17-21), the staff agrees that WOG has properly identified the potential aging effects to be evaluated for the RCS supports. Specific aging mechanisms and their associated aging effects on various components of the RCS supports are discussed below.

3.3.1 Steel Components

The WOG report states that the potential aging effects on the steel components of the RCS supports are loss of material, decrease of strength, decrease of fracture toughness and ductility, cumulative fatigue damage, deformation, and cracking (crack initiation and growth). The WOG report states further that these effects result from the aging mechanisms of stress corrosion cracking, corrosion and aggressive chemical attack, neutron embrittlement, thermal embrittlement, mechanical wear, fatigue, creep and stress relaxation, and low fracture toughness and lamellar tearing. Each of these aging effects is addressed below for steel components.

3.3.1.1 Aging Effects from Stress Corrosion Cracking of Bolting

The key factors for SCC to occur are the use of high-strength materials, a moist environment, and a high level of sustained tensile stress. In the absence of any one of these factors, SCC is unlikely to occur. The only steel components of the RCS supports that are potentially subject to SCC are bolts and anchors made of high-strength material. Most bolts used for the RCS supports within the scope of the WOG report are made of high-strength, low-alloy steel, as indicated by Table 2-4 of the WOG report, and therefore, are subject to SCC. RCS bolts are known to have failed because of SCC and excessive applied loads. The staff agrees with the WOG assessment of aging effects from SCC on bolting as aging effects potentially requiring management. Inspection and Enforcement Bulletin (IEB) 82-02 (Ref. 23) and NUREG-1339 (Ref. 21) specifically addressed this concern for bolting.

3.3.1.2 Aging Effects from Corrosion and Aggressive Chemical Attack

The WOG approach combines corrosion and aggressive chemical attack as the age-related degradation mechanisms that cause loss of materials and decrease of strength of the steel components of the RCS supports. The cause of the degradation is leakage of primary coolant. The WOG assessment is consistent with Table 3.1-1 of the draft SRP-LR. Because both of these aging mechanisms cause similar degradations to the steel components of the RCS supports, the staff agrees with this approach.

3.3.1.3 Aging Effects from Neutron Embrittlement

Chapter 9 of the WOG report repeats statements from NUREG-1509 (Ref. 17), which was issued in May 1996 by the NRC to provide technical resolution of GSI-15, "Radiation Effects on Reactor Pressure Vessel Supports" (Ref. 22). Section 4.2.4 of NUREG-1509 states that "by satisfying the following criteria, the supports should be free from radiation embrittlement, the integrity may be reasonably assured, and no further investigation should be required." These criteria are:

- The initial nil-ductility transition of the RPV supports is well below the minimum operating temperature.
- The radiation exposure at the support is low.
- The peak tensile stresses are 6 ksi or less.

In addition, the executive summary of NUREG-1509 states that "the RPV supports at the Trojan Nuclear Plant (TNP) were identified as the most vulnerable to neutron embrittlement degradation and the consensual agreement was that the result of the TNP study would envelop the industry. Different engineering approaches and various degrees of sophistication were employed by the analysts. Although the analyses provided some confidence that the issue did not appear to pose a serious safety threat, the results showed that there was no single method, applicable to all reactors, by which GSI-15 could be resolved."

Furthermore, in resolving GSI-15 concerns, Revision 3 to NUREG-0933 (Ref. 24) concludes that:

> The preliminary conclusion indicated that the potential problem did not pose an immediate threat to public safety. . . . The tentative results indicated that plant safety could be maintained despite reactor vessel support structures (RVSS) radiation damage. . . In order to encompass the uncertainties in the various analyses and provide an overall conservative assessment, several structural analyses conducted demonstrated the following:
>
> (1) Postulating that one of the four RPV supports was broken in a typical PWR, the remaining supports would carry the reactor vessel and the load even under safe-shutdown earthquake (SSE) seismic loads;
>
> (2) If all supports were assumed to be totally removed (i.e., broken), the short span of piping between the vessel and the shield wall would support the load of the vessel.
>
> The results of the analyses virtually eliminated the concern for both radiation embrittlement and significant structural damage from a postulated RPV failure . . . . Based on the staff's regulatory analysis, the issue was resolved with no new requirements. Consideration of a license renewal period of 20 years did not change this conclusion.

Because of the foregoing, the staff considers that neutron embrittlement is not a concern for the RCS supports, and does not warrant an aging management program.

3.3.1.4 Aging Effects from Thermal Aging Embrittlement

The WOG report states that "temper embrittlement and strain aging embrittlement are forms of thermal aging that are seen in ferritic material. Aging of cast austenitic stainless steels (CASS) at elevated temperatures (above 600°F), temper embrittlement, and strain aging embrittlement are the most common forms." Various forms of embrittlement due to thermal aging have been observed for CASS and low-alloy steel. The WOG report concludes that "there is no CASS used in the supports that are within the scope of the WOG report. Furthermore, in general, RCS supports are operated at temperatures below 450°F. Therefore, temper embrittlement is not a concern for the ferritic materials of RCS supports. Hence, thermal aging embrittlement is not applicable." The staff agrees with the WOG assessment that temper embrittlement is not a concern based on the conclusion of NUREG-1557 "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal" (Ref. 19) provided that the temperature of the RCS supports is maintained below 450°F during operation. Table B9 of NUREG-1557 concludes that "elevated temperature is not a significant age-related degradation mechanism (ARDM) for Class 1 structural steel components, metal sidings, or liners maintained at temperature <371°C (700°F)". However, WOG report did not address the applicability of the aging effects caused by strain aging embrittlement to the RCS supports. The license renewal applicants will address the applicability of the aging effects due to strain aging embrittlement to their plants in their renewal applications. **The is Renewal Applicant Action Item No. 4.**

3.3.1.5 Aging Effects from Mechanical Wear

Aging effect associated with mechanical wear is the loss of surface material caused by surface contact. Slow movements can occur between sliding surfaces of the RCS supports, such as the sliding foot assemblies associated with the RPV and SG supports. The WOG report states that "the RCS supports are not susceptible to mechanical wear that would cause loss of the RCS intended function. This is because of the wear-resistant material used, the low frequency of movement, and the slow movement between sliding surfaces. Lubricants are employed in some of the primary component supports. . . . there is no need of aging management options." This assessment is consistent with Tables B5 and B6 of NUREG-1557 (Ref. 19) which indicate that mechanical wear is not significant for integral supports. Therefore, the staff agrees with WOG that the aging effects of mechanical wear are not a significant concern for the steel components of the RCS supports and, therefore, aging management is not needed for this effect.

3.3.1.6 Aging Effects from Low Fracture Toughness and Lamellar Tear

The WOG report assesses that "low fracture toughness and lamellar tearing do not cause detrimental aging effects that must be addressed by maintenance programs." The abstract of NUREG-0577 "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports" (Ref. 20), which documents the resolution of Unresolved Safety Issue (USI) A-12, addresses lamellar tearing by stating that "lamellar tearing is generally detected and corrected during construction and that a reasonable safety factor on strength can bound the experimental results on torn joints. The staff has concluded that the lamellar tearing aspect of the [USI] A-12 issue is resolved." Therefore, the staff agrees with the WOG that lamellar tearing does not cause detrimental aging effects that must be addressed by aging management programs.

NUREG-0577 also addresses low fracture toughness and provides guidelines and acceptance criteria for utilities with RCS supports potentially low in fracture toughness to demonstrate that adequate fracture toughness exists in the steel components of the RCS supports of their plants. In Appendix C of NUREG-0577, many WOG member facilities are identified as Group 1 "plants requiring further evaluation." The WOG report recognizes this fact and, without justification, states in Section 3.2.9 on Page 3-17 that "low fracture toughness does not cause detrimental aging effects that must be addressed by maintenance programs." The staff is unable to find sufficient information in the WOG report to support this generic conclusion. A license renewal applicant will address, if its plant is listed as Group 1 in Appendix C of NUREG-0577, that its plant had performed an anlysis and the steel components of its RCS supports have adequate fracture toughness that no maintenance program is necessary. **This is Renewal Applicant Action Item 5.**

3.3.1.7 Aging Effects from Fatigue

The WOG report asserts that fatigue is not an applicable aging mechanism because of the low number of cycles or fluctuating loads and the low cumulative usage factor (CUF) as discussed in Section 2.4 of this safety evaluation report. In general, the staff would agree with this assessment if the materials used for the supports had the yield strength as represented in Table 2-4 of the WOG report.

However, many WOG plants used the 1963 AISC (Ref. 3) manual for the design and construction of the steel components of the RCS supports. This manual specifies an upper limit of 10,000 fatigue cycles where no reduction in applied stress or increase in load carrying area is required, rather than the 20,000 cycles presumed in the WOG report. According to the 1963 AISC manual, the primary structural steels used for design and construction are A7 and A36 steels, and it appears that some of the RCS supports were fabricated from these steels. These steels do not have as great a yield strength or fatigue resistance as the more modern structural steels listed in Table 2-4 of the WOG report. Consequently, the CUF values given in Table 3-2 of the WOG report may not be representative. Therefore, a license renewal applicant should address this concern in its application. **This is Renewal Applicant Action Item 6.**

3.3.1.8 Aging Effects from Creep and Stress Relaxation

WOG assesses the aging effects from creep and stress relaxation and concludes that, "the temperature (T) in the PWR RCS supports is generally below 650°F (1110°R), well below half of the melting point (T_m) of steels (T_m=2410°F=2870°R, and T/T_m=0.39), creep and stress relaxation are not issues for the RCS supports for extended operation." Generally, creep becomes of engineering significance only at a homogeneous temperature ratio (T/T_m) greater than 0.5 (Ref. 31). However, one of the aging mechanisms for loss of preload is stress relaxation. Section 4.1 of the WOG report states that RCS supports are not generally designed to use bolted joint connections requiring preload. If used, in order to develop the full design strength of the structural member, the AISC manual (Ref. 3) requires a minimum bolt tension which equals 70 percent of the ultimate strength of the material. WOG recognizes this requirement on Page 5-2 of the WOG report and states that a license renewal applicant must identify "any specific program necessary to ensure that proper preload is retained for the component supports within the scope of this report." The staff considers this approach acceptable and makes this action as part of **Renewal Applicant Action Item 16** (see Section 3.6).

3.3.2 Concrete Components

Like most structural materials, concrete is susceptible to age-related degradation from the exposure to weathering, ground water, elevated temperature, irradiation, and other unfavorable conditions. The WOG report addresses the aging effects resulting from rebar corrosion, leaching of calcium hydroxide, aggressive environments, and elevated temperature. The staff's evaluation of each of these effects is set forth below. The WOG report does not address the aging effects from irradiation and states that they should be addressed by the renewal applicant as plant-specific evaluations. This is part of **Renewal Applicant Action Item No. 7** (see Section 3.3.2.3).

3.3.2.1 Aging Effects from Leaching of Calcium Hydroxide

Aging effects from the leaching of calcium hydroxide occur to concrete when water enters and passes through a concrete body, washing out the readily soluble calcium hydroxide and other solids. As a result, the porosity of the concrete is increased, boosting vulnerability to a hostile environment while reducing strength. Note that leaching is significant only when water flows into cracks or improperly constructed joints. The staff agrees that leaching is a concern for the concrete components of the RCS supports because the concrete might be in contact with water and concrete cracking does occur.

3.3.2.2 Aging Effects from Aggressive Environments

The WOG report states that most concrete components were designed and constructed in accordance with various editions of the ACI-318 or ACI-349 Codes (Refs. 25 and 26, respectively) resulting in dense, well-cured concrete with low permeability and proper reinforcement. Hence, aging effects from aggressive chemical attack are not concerns unless the concrete component is exposed to aggressive chemicals with a pH value less than 5.5 or chloride or sulfate solutions beyond defined limits (500 ppm chlorides and 1500 ppm sulfates) for an extended period of time. This statement is in agreement with the staff's assessment in Table B9 of NUREG-1557 that "for class 1 structures that meet the basis requirements, (pH < 5.5), or to chloride or sulfate solutions beyond defined limits (>500 ppm chloride, or >1500 ppm sulfate); or if exposed to [an] aggressive environment that exceeds the pH, chloride, or sulfate limits, the exposure is for intermittent periods only, aggressive chemical attack is [a] non-significant ARDM". The concrete components of the RCS supports may be exposed to one or more of these conditions for an extended period during the extended period of operation; therefore, the staff concurs with the WOG assessment that aging effects from aggressive chemical attack are a concern for the RCS support concrete components.

3.3.2.3 Aging Effects from Irradiation

Section 3.2.8.d of the WOG report states that concrete degradation due to radiation will be addressed by plant-specific evaluations. This action is part of **Renewal Applicant Action Item 7.**

3.3.2.4 Aging Effects from Elevated Temperature

Section 3.2.8.c of the WOG report states that "sustained exposure to high temperature (300°F or higher) or to numerous hot-cold cycles may cause concrete to deteriorate." In response to RAI #36 (WCAP-14422, Rev. 0), WOG stated that the 300°F concrete exposure temperature is the temperature at which the concrete begins to deteriorate and surface scaling and cracking become physically visible. The WOG report further states that "concrete operating temperature should not exceed 150°F, and local area temperature should be kept under 200°F. Reactor vessel supports could be subjected to high temperatures that could potentially result in a local temperature above 200°F if supplemental cooling is not provided. For those support configurations where the local temperature at concrete surfaces could exceed 200°F, special design features are incorporated based on air or water cooling to keep the local temperature below 200°F." These values are within the allowable of the ASME Code (Ref. 27), therefore, elevated temperature is not a concern for concrete.

The staff considers that the aging effects of elevated temperature are applicable to the RCS supports and are being managed by supplemental cooling features. The license renewal applicant will address this concern stating that the aging effects associated with elevated temperature are applicable and demonstrating that the existing design features in their plant are capable of preventing any unacceptable elevated temperature caused degradation during the period of extended operation. **This is Renewal Applicant Action Item 8.**

3.3.2.5 Aging Effects from Cracking and Rebar Corrosion

Concrete cracking is common because cracking can be caused by tensile stress and concrete has very low tensile strength. The WOG report states that "cracking is the path to leaking and hostile environments, which in turn become a source for further damage, such as rebar corrosion and concrete leaching." The WOG report further states that "under normal conditions, the highly alkaline environment of concrete provides a protective film to prevent corrosion of the steel rebar. The presence of cracks promotes the carbonation of concrete, resulting in the reduction of pH and breaking down of the protecting film, and leading to subsequent rebar corrosion." The staff agrees with WOG that cracking and rebar corrosion is plausible for the concrete components of the RCS supports.

3.4 Aging Management Programs

WOG reviewed and evaluated the original design bases, TLAAs that were inherent in the original designs, maintenance practices, inspection results, and aging effects on the RCS supports. Section 2.6.5 of the WOG report states, "No documentation related to industry operating experience associated with aging has been found for the supports within the scope of this report." On the basis of its review and evaluation, WOG proposed three AMPs to manage the effects of aging so that the intended functions of the RCS supports will be maintained consistent with the CLB for the period of extended operation. These AMPs are:

- AMP-1.1, "Aggressive Chemical Attack and Corrosion for Steel"
- AMP-1.2, "Aggressive Chemical Attack and Corrosion for Concrete Embedments"
- AMP-1.3, "Stress Corrosion Cracking for Bolting"

WOG proposed AMPs contain the following attributes: (1) scope, (2) surveillance techniques, (3) frequency [of surveillance], (4) acceptance criteria, (5) corrective actions, and (6) confirmation. These attributes are comparable to the attributes identified in the guidance contained in Table A1-1of Appendix A of the draft SRP-LR (Ref. 16) except that the AMPs do not specifically address the review elements of parameters monitored or inspected, trending activities, administrative controls, and operating experience. A license renewal applicant that intends to reference the WOG report must to provide plant-specific AMPs that address the missing review elements contained in Table A1-1 of Appendix A of the SRP-LR. **This is Renewal Applicant Action Item 9.**

The staff evaluates the attributes discussed in the WOG report of each AMP to determine if the intended functions of the RCS supports will be maintained consistent with the CLB during the period of extended operation.

3.4.1 Scope of Aging Management Programs

AMP-1.1 encompasses the steel components of the RCS supports specified in Section 2.1 of the WOG report and addresses the aging effects from aggressive chemical attack and corrosion. AMP-1.2 covers the concrete embedments described in Section 2.1 of the WOG report and addresses the aging effects from aggressive chemical attack and corrosion (including leaching). AMP-1.3 covers all RCS support bolts and studs specified in Section 2.1 of the WOG report and addresses the aging effects from stress corrosion cracking. The staff agrees that the AMPs should address the RCS support components that are within the scope of the WOG report and the aging effects identified by WOG.

3.4.2 Surveillance Techniques

The surveillance techniques for AMP-1.1 (aggressive chemical attack and corrosion of steel) specify inspection (examination) to the standards of Section XI, Subsection IWF-2500 (Examination Requirements) and Table IWF-2500-1 (Examination Categories), with Subsections IWF-2520 (Method of Examination) or Subsection IWA-2240 (Alternative Examinations) of the ASME Code. Table IWF-2500-1 addresses Examination Categories F-A Supports. This table sets forth the items to be examined, examination requirements, examination method, acceptance standard, extent of examination, and frequency of examination.

The surveillance techniques specified for detection of leakage are leakage identification walkdowns. For leakage monitoring, the techniques are to monitor the increase in humidity level, change in fluid volume, increase in temperature, or increase in radioactivity.

AMP-1.2 (aggressive chemical attack and corrosion of concrete embedment) specifies ACI standards 201.1R-68 (Ref. 9), 207.3R-79 (Ref. 10), 224.1R-89 (Ref. 11), and 349.3R-96 (Ref. 12) to be used as guidance to inspect concrete embedments. The staff has reviewed the above mentioned standards, especially ACI 349.3R-96, which refers to other ACI standards, and considers that ACI 349.3R-96 can be used to manage concrete aging effects for license renewal applications because it is written for such situations and it considers all potential facets for evaluating existing nuclear concrete structures. Standard ACI 349.3R-96 provides an engineering review of an existing concrete nuclear structure with the purpose of determining physical condition and functionality of the structure. It provides an evaluation procedure, degradation mechanisms, evaluation criteria (acceptance criteria), evaluation frequency, and qualifications of evaluation team. ACI 349.3R-96 also provides a repair procedure based on the requirements specified in the ACI 349 Code. ACI 349 is the code that governs the design of nuclear safety-related concrete structures.

AMP-1.2 also specifies leakage identification walkdowns and leakage monitoring program as part of the surveillance techniques for leakage walkdown and leakage monitoring. However, the AMP does not provide details of the leakage identification walkdowns and leakage monitoring program. Therefore, license renewal applicants will have to provide plant specific programs for leakage walkdowns and leakage monitoring. **This is Renewal Applicant Action Item 10.**

AMP-1.3 (stress corrosion cracking of bolting) specifies inspection (examination) to the standards of Section XI, Subsection IWF-2500 (Examination Requirements) and Table IWF-2500-1 (Examination Categories), with Subsections IWF-2520 (Method of Examination) or Subsection IWA-2240 (Alternative Examinations) of the ASME Code. Table IWF-2500-1 addresses Examination Categories F-A supports. This table sets forth the items to be examined, examination requirements, examination method, acceptance standard, extent of examination, and frequency of examination.

The staff considers the above surveillance techniques acceptable due to the fact that they have been used by the industry and have been demonstrated to be capable of identifying aging effects of the RCS support components with the following exceptions:

1. Baseline inspection - Baseline inspection is intended to document the current condition of a structure or structural component, consequently, any previous inspection which satisfies this purpose can be credited as the baseline inspection. Section 4.2.2 of the WOG report indicates that "the aging management program attributes in Section 4 of the report are intended to be implemented after completion of an initial baseline evaluation of the bolts in the RCS supports." It also states that "the initial baseline evaluation should follow the guideline in EPRI report NP-5769." The WOG report does not provide any specific information about the baseline evaluation. The staff reviewed the EPRI report entitled "Degradation and Failure of Bolting in Nuclear Power Plants (Ref. 29)." Section 11 of Volume 2 of the EPRI report, "Evaluation Procedure for Assuring Integrity of Bolting Material in Component Support Applications" provides an approach to evaluate the allowable bolt load based on the fracture properties of the materials. However, the EPRI report only addresses the evaluation of bolting degradation. The staff concludes that a baseline inspection is needed to document the condition of the structures and structural components which will serve to validate the scope, acceptance criteria, and aging effects for the applicable aging management programs. Therefore, the renewal applicants will have to have plant-specific baseline inspection results for all structures and structural components, or a planned inspection to obtain such results and validate the aging management programs prior to entering the period of extended operation. **This is Renewal Applicant Action Item 11.**

2. Inaccessible areas - Inaccessible areas are subject to age-related degradation effects from the aging mechanisms mentioned in the AMPs. Section 4.2.1 of the WOG report indicates that the maintenance program should address inaccessible areas. The WOG report also indicates that utilities must rely on visual examinations (direct and indirect) for evidence of degradation, such as binding, leaking of fluid, and discoloring or flaking of the surface coating. This evidence will alert the inspectors to potential degradation, aid in assessing degradation, and help in performing more detailed inspections. The WOG report further states, "The management program recommended acceptable technical procedures using indirect visual evidence of degradation to identify potential aging degradation within these areas." The WOG report does not address the situations where

there is no indirect visual evidence or when evidence is not representative of the inaccessible areas.

In response to the staff's RAI #11 of Revision 1, WOG states that inspections of inaccessible areas are not necessary in load bearing areas because "no significant aging effect has occurred . . . and potential degradation due to wear is not considered a significant aging mechanism." The WOG report further states:

> Further, the inspection program given in the GTR for inaccessible areas is adequate to manage the potential aging degradation identified for these supports. If [a] utility has evidence of aging degradation in inaccessible areas during the current licensing term which they may deem potentially affecting system intended function, then the utility should so identify this situation in their plant-specific application. This is [a] plant-specific item that may result in a need for a one-time direct inspection of an inaccessible area prior to the extended licensing term.

The staff agrees with WOG that the inspection of inaccessible areas is plant-specific and should be left for the license renewal applicants to address it. A license renewal applicant must provide an inspection program to inspect inaccessible areas or provide technical justification for not performing inspection. **This is Renewal Applicant Action Item 12.**

3.4.3 Frequency

For AMP-1.1, the frequency of examination (inspection) to detect aging effects is based on the ASME Code at intervals set by the Code, leakage walkdowns performed at each refueling outage, and leakage monitoring performed "as-needed." The staff considers that the inspection frequency is acceptable because they are based on NRC accepted ASME Codes. However, this AMP does not provide an explanation on how the "as-needed" frequency is determined. The frequency of leakage monitoring should be addressed by the applicants in license renewal applications as part of the plant-specific programs (see Renewal Applicant Action Item 10).

AMP-1.2 specifies inspection frequency in accordance with the standard of Subsection IWF-2410 (Inspection Program) and Table IWB-2412-1, each 10-year interval following the first interval, 10-year inspection program, with IWB-2412. The staff considers that the frequency proposed by WOG is not adequate. The inspection frequencies recommended by ACI 349.3R-96 are every 10 years for below grade structures and controlled interiors and every 5 years for all other structures. Section 4.2.4.1 of NUREG/CR-6424 has the same recommendation for inspection frequencies. The surveillance technique of AMP-1.2 specifies that ACI standards are to be used, therefore, the inspection frequency from the same ACI standards should be used. An license renewal applicant must address this concern in its application. **This is Renewal Applicant Action Item 13.** The frequencies for leakage walkdowns are at each refueling outage and continuous leakage monitoring are acceptable and they should be included as part of a plant-specific AMP (see Renewal Applicant Action Item 10).

AMP-1.3 (SCC of bolting) requires the inspection frequency for the bolting to be that of Subsection IWF-2410 (Inspection Program) and Table IWB-2412-1, each 10-year interval, following the first interval, 10-year inspection program, with Subsection IWB-2412 (Inspection Program B). Table IWB-2412-1 (Inspection Program B) specifies a 100 percent inspection every 10 years. The staff finds this approach acceptable because it is based on NRC accepted ASME Code.

3.4.4 Acceptance Criteria

The acceptance criteria for inspection specified for AMP-1.1 (aggressive chemical attack and corrosion of steel) are to the standards of Subsection IWF-3410 (Acceptance Standards-Component Support Structural Integrity). For leakage walkdowns, the acceptance criteria are identification of fluid leakage and for leakage monitoring, they are plant-specific leakage monitoring criteria. The acceptance criteria for inspection are adequate because they are based on NRC endorsed ASME recommendations. The acceptance criteria for leakage walkdowns and monitoring follow the utilities' plant-specific criteria, therefore, they are plant-specific and have to be provided by the license renewal applicants. **This is Renewal Applicant Action Item 14.**

The acceptance criteria for AMP-1.2 (aggressive chemical attack and corrosion of concrete embedments) include some ACI standards that may be used as a guide for establishing acceptance criteria for inspections. These ACI standards are ACI 201.2R-77 (Guide to Durable Concrete) (Ref. 13), ACI 224.1R-89 (Cause, Evaluation, and Repair of Cracks in Concrete Structures) (Ref. 11), and ACI 224R-89 (Control of Cracking in Concrete Structures) (Ref. 14). The staff has reviewed these ACI standards and concluded that, except for ACI 224.1R, they are mainly for design and construction rather than for aging effects management since those concrete properties (e.g., durability, crack resistance) are built-in by design and construction. However, the standards do contain attributes that can be used to develop inspection acceptance criteria for AMP-1.2. For leakage walkdowns and leakage monitoring, the acceptance criteria are the same as that listed for AMP-1.1. The staff has also reviewed ACI 349.3R-96 (Evaluation of Existing Nuclear Safety-Related Structures) (Ref. 12) and concluded that the acceptance criteria of this standard can be modified and used as the inspection acceptance criteria for AMP-1.2. These criteria include acceptance without further evaluation, acceptance after review, and conditions requiring further evaluation. The license renewal applicants will provide a description of the inspection acceptance criteria in their application for the staff to review. **This is Renewal Applicant Action Item 15.**

The table in AMP-1.3 (SCC for bolting) specifies acceptance criteria to the standards of Subsections IWF-3410, IWF-3200, and IWA-2000, based on VT-1 and VT-3 visual examinations. Subsection IWF-3410 (Acceptance Standards-Component Support Structural Integrity) is the only cited code section with acceptance standards. Subsection IWF-3200 is entitled "Supplemental Inspections" and indicates that detected conditions that require evaluation in accordance with the requirements of IWF-3100 (Evaluation of Examination Results) may be supplemented by other examination methods and techniques (IWA-2000) to determine the character of the flaw. Subsection IWA-2000, which is entitled "Examination and Inspection," specifies examination methods, qualifications of examination personnel, and inspection programs among other things. The staff finds that the WOG report provides adequate acceptance criteria for AMP-1.3 because WOG uses an NRC endorsed ASME Code which has been used effectively to detect degradations of component supports.

3.4.5 Corrective Actions

The WOG report specifies corrective actions for inspection for AMP-1.1 (aggressive chemical attack and corrosion for steel) to the standards of Subsections IWF-3112 (Acceptance Criteria for Preservice Examinations) and IWF-3200 (Supplemental Examinations) or Subsection IWF-3122 (Acceptance Criteria for Inservice Examinations) and IWF-3200. Subsection IWF-3112 and Subsection IWF-3122 are almost identical. They include acceptance by examination, acceptance by correction, and acceptance by evaluation or test. IWF-3200 specifies that examinations that detect conditions that require evaluation in accordance with the requirements of IWF-3100 may be supplemented by other examination methods and techniques to determine the character of the flaw. These procedures and methods are recognized as acceptable means to address inspection and maintenance issues by the industry and the NRC; the NRC has endorsed ASME Code Section XI through 10 CFR 50.55a(b)(2). Therefore, the staff concludes that the corrective action programs proposed by WOG are appropriate. The corrective actions for leakage walkdowns and leakage monitoring are to remove standing fluids, evaluate boric acid buildup, clean and restore affected surfaces, and identify and repair sources of leaks. The staff judges these corrective actions appropriate because they clean and restore the affected surfaces and identify and repair the source of leakage to prevent recurrence.

The corrective actions for inspection specified for AMP-1.2 (aggressive chemical attack and corrosion for concrete embedments) list five ACI standards in addition to Subsection IWF-3112 or IWF-3122. These five ACI standards are:

(1) ACI 201-2R-77, "Guide to Durable Concrete" (Ref. 13)
(2) ACI 207.3R-79, "Practices for Evaluation of Concrete in Existing Massive Structures for Service Conditions" (Ref. 10)
(3) ACI 222R-89, "Corrosion of Metal in Concrete" (Ref. 15)
(4) ACI 224.1R-89, "Causes, Evaluation, and Repair of Cracks in Concrete Structures" (Ref. 11)
(5) ACI 224R-89, "Control of Cracking in Concrete Structures" (Ref. 14)

As indicated in the previous paragraph, the staff considered that Subsections IWF-3112 and IWF-3122 are acceptable means for corrective actions. The staff also reviewed the above mentioned ACI standards and concluded that those ACI standards, together with Subsections IWF-3112 and IWF-3122, constitute an acceptable corrective action program because they provide the necessary guidance to correct and repair various flaws in concrete structures. The corrective actions for leakage walkdown and leakage monitoring are to remove standing fluids, clean and restore affected surfaces, and identify the source of leak and repair and are judged to be appropriate (see staff evaluation on AMP-1.1).

AMP-1.3 (SCC of bolting) specifies the corrective actions to be evaluating existing materials and design, modifying susceptible materials or design as necessary, or replacing the defective bolts. The staff considers the corrective action appropriate because it evaluates bolts with degradation and, if warranted, replaces the defective bolts.

3.4.6 Confirmation

For AMP-1.1, confirmation entails a preservice examination specified by Subsection IWF-2200 following adjustment, repair, or replacement prior to return of the system to service. Successful inspection at specified intervals pursuant to the standards of Subsection IWF-2420, and additional examinations in accordance with the standards of Subsection IWF-2430. The staff considers this approach acceptable because it ensures that the support is functional before it is returned to service and periodic examination to ensure that the support stays functional and that other supports immediately adjacent to those requiring corrective action be examined. The method of confirmation for leakage walkdowns is to re-examine the affected surfaces after cleaning or restoration and re-examine at the next outage. Confirmation for leakage monitoring is continuous monitoring to ensure the leakage monitoring program is effective. The staff considers these confirmations acceptable because they ensure the functionality of the supports.

The confirmation for the effectiveness of the inspection specified for AMP-1.2 following adjustment, repair, or replacement to ensure that the corrective actions have been completed and effective before returning the structure or component to service includes a preservice examination conforming to the standards of Subsection IWF-2200, successful inspections at specified intervals pursuant to the standards of Subsection IWF 2420, and additional examinations meeting the standards of Subsection IWF-2430. For leakage walkdowns, the confirmation is to re-examine the affected surfaces after cleaning and restoration and re-examine at the next outage. For leakage monitoring, the confirmation is continuous monitoring to ensure the leakage monitoring program is effective. The staff considers the confirmation approach adequate (see staff evaluation for AMP-1.1).

Confirmation for AMP-1.3 (SCC of bolting) starts with a post-maintenance test or other technique to confirm that the actions have been completed and are effective. This is followed by a re-examination of replaced bolts at the next inspection interval to determine if they are still susceptible to SCC. The staff judges this approach adequate due to the fact that it uses post-maintenance testing to ensure the functionality of the supports and re-examination to ensure that the supports stay functional.

3.5 Time-Limited Aging Analyses

The WOG report lists fatigue as the only applicable TLAA for the components of the RCS supports. Section 2.5 of the WOG report discusses and evaluates TLAAs and fatigue damage to the RCS supports and determines that "fatigue is not part of design qualification analyses for the component supports within the scope of this report since they are not subject to high fatigue usage factors and significant stress cycles in excess of 20,000. It is concluded that no additional analyses are required to be performed by the utility for demonstration that TLAAs are acceptable for the extended period of operation." Section 3.2.6 of the WOG report further states that "from Table 3-2, the estimated maximum fatigue usage in the RCS supports is less than 0.1 for 40 years of plant operation. For 60 years of operation, the estimated fatigue usage is less than 0.15. Further, the number of cycles are much less than [the] 20,000 cycles [as] discussed in Section 2.5, recognized by ASME Section III, Subsection NF, and the AISC as the potential number of cycles where fatigue may need to be considered in design. Therefore, fatigue is not an effect that is a concern for the RCS support structures." The staff evaluation is presented in Section 3.3.1.7 of this safety evaluation report.

3.6 Plant-Specific Programs

The staff reviewed the recommended plant-specific programs in Section 5 of the WOG GTR and finds that most of these recommended programs are necessary to manage aging effects of the RCS supports within the scope of this GTR. Therefore, they are included as part of the renewal applicant action items in Section 4.1 of this safety evaluation report. The recommended programs are:

- Identification and evaluation of any plant-specific TLAAs applicable to their RCS supports.

- Identification and evaluation of current-term programs implemented within the current licensing term to address technical issues from industry practices and United States Nuclear Regulatory Commission (NRC) directives [that] should be continued into the license renewal term. Modifications to or elimination of these programs have to be justified.

- Identification and justification of plant-specific programs that deviate from the recommended AMPs.

- Identification of any specific program necessary to ensure that proper preload is retained for the component supports within the scope of this report.

- Identification of any evidence of aging degradation in inaccessible areas during the current licensing term that is considered to potentially affect system intended functions. A plan of action to address any identified potential degradation should be provided.

- Verification that the plant is bounded by this GTR. The actions applicants must take to verify that their plant is bounded will be described in an implementation procedure.

- Plant-specific evaluation of potential degradation due to irradiation of the components within the scope of this report.

The staff concurs with the WOG report and makes this **Renewal Applicant Action Item 16.**

4 STAFF CONCLUSION AND RENEWAL APPLICANT ACTION ITEMS

4.1 Renewal Applicant Action Items

A utility which wants to reference this WOG report in a license renewal application has to perform the following applicant action items and submit them for staff review.

(1) **Renewal Applicant Action Item 1** Definition of "local" and "adjacent" (Section 3.1)

The WOG did not clearly define the term "local" in its report. However, the aging management programs could be the same for all concrete structures and structural components, therefore, the license renewal applicants must describe the aging management program for adjacent concrete structures and any differences from the aging management program for the local concrete structures.

(2) **Renewal Applicant Action Item 2** Detailed description of the RCS supports (Section 3.1)

A license renewal applicant will have to justify any differences between its RCS support system and the figures and descriptions of the supports systems contained in the WOG report.

(3) **Renewal Applicant Action Item 3** Discrepancies and Omissions (Section 3.1)

The WOG report contains many discrepancies and omissions:

1. Wear plates and bearing pads are included as support components and are within the scope of this WOG report but are not identified in Table 2-1 as parts and sub-components requiring an aging management review.

2. Sketches of RCP support configuration 4 and PZR support configuration 2 are not provided in the WOG report.

3. Section 3.2.9 of the WOG report indicates that ASTM A36 steel is used in SG and RCP supports, however, ASTM A36 steel is not included in the list of material for the primary component supports (Table 2-4).

4. The 1963 AISC manual (Ref. 3) states that the following steel materials are commonly used for steel construction but they are not listed in Table 2-4 of the WOG report. They are ASTM A7, A36, A242, A373, A440, and A441 structural steel and ASTM A325 bolts.

5. There are no specific descriptions and sketches for the PZR surge line supports.

A license renewal applicant needs to resolve these discrepancies and omissions in its application.

(4) **Renewal Applicant Action Item 4** Strain Aging Embrittlement (Section 3.3.1.4)

Temper embrittlement and strain aging embrittlement are the most common forms of thermal embrittlement that are seen in ferritic materials as stated in Section 3.2.4 of the WOG report. The WOG report has determined that temper embrittlement is not a concern for the ferritic materials of RCS supports. However, the WOG report does not address the aging effects from strain aging embrittlement but states that thermal embrittlement is not applicable. The license renewal applicants will address the applicability of the aging effects due to strain energy embrittlement to their plants.

(5) **Renewal Applicant action Item 5** Low Fracture Toughness (Section 3.3.1.6)

Appendix C of NUREG-0577 addresses this item and groups many WOG member plants as Group I "plants requiring further evaluation." Although Table B9 of NUREG-1557 indicated that "low fracture toughness is not significant for containment internal structures," in general, these two documents only addressed the containment internal structures as a whole and did not specifically address the RCS support components. WOG recognizes this concern and states in Section 3.2.9 of its report that "Utilities with potential problems were required to demonstrate that the suspect structures have adequate fracture toughness to comply with the criteria defined in NUREG-0577." However, it further states that "low fracture toughness does not cause detrimental aging effects that must be addressed by maintenance programs." The staff does not believe that the WOG report provides sufficient information to support this conclusion. A license renewal applicant will address, if its plant is listed as Group 1 in Appendix C of NUREG-0577, that its plant had performed an analysis and the steel components of its RCS supports have adequate fracture toughness that no maintenance program is necessary.

(6) **Renewal Applicant Action Item 6** Fatigue (Section 3.3.1.7)

A license renewal applicant will have to justify any differences between the materials used for its RCS supports and the values listed in Table 2-4 of the WOG report.

(7) **Renewal Applicant Action Item 7** Irradiation of Concrete (Section 3.3.2.3)

The WOG report states that concrete degradation from irradiation will be addressed by plant-specific evaluation. The staff agrees with this suggestion and the license renewal applicant must develop plant-specific program(s) to evaluate this concern.

(8) **Renewal Applicant Action Item 8** Elevated Temperature of Concrete (Section 3.3.2.4)

The WOG report states that concrete operating temperature should not exceed 150°F and local area temperature should be kept under 200°F. The WOG report further states that RPV supports could be subjected to high temperatures that could potentially result in a local temperature above 200°F if supplemental cooling is not provided. For those support configurations where the local temperature at concrete surfaces could exceed 200°F, special design feature are incorporated based on air or water cooling to keep local temperature below 200°F. These temperatures are specified in the ASME Code. Therefore, elevated temperature is not a concern for concrete.

Because the operating temperature of concrete components are kept below the limits specified by the code by means of supplemented cooling, the staff considers that the aging effects of elevated temperature are applicable to the RCS supports and are being managed by supplemented cooling features. The license renewal applicants will address the concern that the aging effects associated with elevated temperature are applicable and demonstrate that the existing design features in this plants are capable of preventing any unacceptable degradation during the period of extended operation.

(9) **Renewal Applicant Action Item 9** SRP-LR (Section 3.4)

The attributes of the AMPs provided in the WOG report do not address all elements as listed in Table A1-1 of Appendix A of the SRP-LR. The applicants should address the missing review elements and describe the plant-specific experience, if any, related to aging degradation of the RCS supports in their applications.

(10) **Renewal Applicant Action Item 10** Details of leakage walkdowns and leakage monitoring program (Section 3.4.2)

A license renewal applicant must provide the necessary details to perform leakage identification walkdowns and the details of the leakage monitoring program(s), especially the frequencies, for AMP- 1.1 and AMP- 1.2.

(11) **Renewal Applicant Action Item 11** Baseline Inspection (Section 3.4.2)

All structures and structural components need a baseline inspection to document the condition of the structures and structural components. Therefore, the renewal applicants must have plant-specific baseline inspection results for all structures and structural components, or a planned inspection to obtain such results and validate the aging management programs prior to entering the period of extended operation.

(12) **Renewal Applicant Action Item 12** Inspection of inaccessible areas (Section 3.4.2)

For RCS supports located in inaccessible areas, a license renewal applicant must provide an inspection program to inspect these RCS supports or provide technical justification for not performing inspection.

(13) **Renewal Applicant Action Item 13** Surveillance Frequency for AMP-1.2 (Section 3.4.3)

AMP-1.2 specifies inspection frequency in accordance with the requirements of Subsection IWF-2410 (Inspection Program) and Table IWB-2412-1, each 10-year interval following the first interval, 10-year inspection program, with IWB-2412. The staff considers the frequency proposed by WOG to be inadequate. The proposed frequency is in accordance with ASME standards, but the inspections are to the requirements of ACI Standards, therefore, the frequency of inspection should also follow the recommendations of the ACI standards. Inspection frequencies recommended by ACI 349.3R-96 are every 10 years for below grade structures and controlled interiors and every 5 years for all other structures. Section 4.2.4.1 of NUREG/CR-6424 has the same recommendation for inspection frequencies. An license renewal applicant must address this concern in its applicant.

(14) **Renewal Applicant Action Item 14** Acceptance criteria for leakage walkdowns (Section 3.4.4)

In accordance to the WOG report, leakage walkdowns and monitoring are plant-specific. Therefore, a license renewal applicant will have to provide the necessary qualitative or quantitative acceptance criteria for leakage walkdowns and monitoring.

(15) **Renewal Applicant Action Item 15** Acceptance Criteria for AMP- 1.2 (Section 3.4.4)

AMP-1.2 specifies acceptance criteria in accordance with several ACI standards. These ACI standards are ACI 201.2R-77, ACI224.1R-89, and ACI 224R-89. The staff has reviewed these ACI standards and concluded that, except for ACI 224.1R, they are mainly for design and construction rather than aging effects management because those concrete properties are built-in by design and construction. However, they do contain attributes that can be used to develop inspection acceptance criteria for AMP-1.2. For leakage walkdowns and leakage monitoring, the acceptance criteria are the same as that listed for AMP-1.1. The staff has also reviewed ACI 349.3R-96, which is referenced in the WOG report for surveillance technique, and concluded it has acceptance criteria that can be modified and used as the inspection acceptance criteria for AMP-1.2. These criteria include acceptance without further evaluation, acceptance after review, and conditions requiring further evaluation. The license renewal applicants will provide a description of the inspection acceptance criteria in their application for the staff to review.

(16) **Renewal Applicant Action Item 16** Plant-Specific Programs. Recommendations from Section 5 of the WOG report (Section 3.6)

- Identification and evaluation of any plant-specific TLAAs applicable to their RCS supports.

- Identification and evaluation of current-term programs implemented within the current licensing term to address technical issues from industry practices and United States Nuclear Regulatory Commission (NRC) directives [that] should be continued into the license renewal term. Modifications to or elimination of these programs have to be justified.

- Identification and justification of plant-specific programs that deviate from the recommended AMPs.

- Identification of any specific program necessary to ensure that proper preload is retained for the component supports within the scope of this report.

- Identification of any evidence of aging degradation in inaccessible areas during the current licensing term that is considered to potentially affect system intended functions. A plan of action to address any identified potential degradation should be provided.

- Verification that the plant is bounded by this GTR. The actions applicants must take to verify that their plant is bounded will be described in an implementation procedure.

- Plant-specific evaluation of potential degradation due to irradiation of the components within the scope of this report.

4.2 STAFF CONCLUSION

The staff has reviewed the subject WOG GTR WCAP-14422 (Ref. 2) and additional information submitted by the WOG. On the basis of its review, as set forth above, the staff concludes that, upon completion of all renewal applicant action items in Section 4.1 of this safety evaluation, the staff will be able to find that a license renewal applicant who references the WOG report adequately demonstrates that the effects of aging of the components of the RCS support within the scope of this WOG report can be managed so that there is reasonable assurance that the RCS supports components will perform their intended function(s) in accordance with the CLB during the period of extended operation. Accordingly, the staff concludes that, subject to completion of the renewal applicant action items described in Section 4.1, any operating WOG member plant may reference WCAP-14422 in a license renewal application and doing so will provide the staff with sufficient information to make the necessary findings required by 10 CFR 54.29(a)(1) for components within the scope of this WOG report.

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DISCLAIMER OF RESPONSIBILITY

This report was prepared by the WOG LCM/LR Working Group as an account of work sponsored by the WOG. Neither members of the WOG, Westinghouse Electric Company, nor any person acting on their behalf:

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EXECUTIVE SUMMARY

This report evaluates aging of the reactor coolant system (RCS) supports to ensure that their intended functions can be maintained during an extended period of operation. The RCS supports maintain the system intended functions of:

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

The intended functions of the RCS supports are to maintain the RCS components in equilibrium within spatial positions prescribed by design and ensure the structural integrity and safe operation of the RCS piping and primary components under design conditions (the system intended function, as defined in 10 CFR 54).

The RCS supports are subject to an aging management review because they maintain intended functions, are passive, and are long-lived. This aging management review identifies mechanisms that cause aging effects and presents options that manage these effects to ensure that intended functions are maintained.

This report focuses on age-related issues associated with the support structures for the primary components of the RCS. This includes the reactor pressure vessel (RPV), the steam generator (SG), the reactor coolant pump (RCP), the pressurizer (PZR), and the PZR surge line supports.

All design limits, aging effects, and industry issues have been evaluated. Options to manage aging effects that impact intended functions are provided. For RCS supports, the aging effects caused by the following mechanisms require management:

- Aggressive chemical attack
- Corrosion
- Stress corrosion cracking (SCC)

Aging management options have been given to manage aging effects. These options use methods and techniques that utilities employ in their maintenance practices.

Aging effects caused by mechanisms that are not significant and are not a consideration in ongoing inspection and maintenance programs associated with RCS supports are:

- Thermal aging embrittlement
- Mechanical wear
- Fatigue

- Creep and stress relaxation
- Concrete degradation
- Low fracture toughness and lamellar tearing

No additional analyses are required to be performed by the utility for demonstration that time-limited aging analyses (TLAAs) are acceptable for the extended period of operation. All required demonstration requirements are contained in this report.

Options to manage aging that are part of current industry practice are presented, and the effectiveness of these programs during an extended period of operation is justified.

GSI-15, "Radiation Effects on Reactor Vessel Supports," has not been generically addressed in this document. A utility will submit a plant-specific resolution at the time of the renewal application. Concrete degradation due to radiation is addressed in this report.

Implementation of the aging management options will manage the identified aging effects. In conclusion, this evaluation shows that intended functions of the RCS supports will be maintained during an extended period of operation by implementation of the identified aging management options.

Revisions 1 and 2 to this report incorporated responses to Nuclear Regulatory Commission Requests for Additional Information (RAI). This approved version (WCAP-14422, Rev. 2-A) incorporates the NRC Final Safety Evaluation.

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LIST OF ABBREVIATIONS AND ACRONYMS

°F	Degrees Fahrenheit
Δk	Range of the applied stress intensity factor
σ_{p+b}	Enveloping membrane plus bending stresses
β	Thickness requirement for a plane strain condition
σ_{ys}	Yield strength of material
A	Cross-sectional area
ACI	American Concrete Institute
AISC	American Institute of Steel Construction
AISI	American Institute of Steels and Iron
ANL	Argonne National Laboratory
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWS	American Welding Society
B&PV	Boiler & pressure vessel
BWR	Boiling water reactor
C,n	Material constants depending on stressing and environmental conditions
CLB	Current licensing basis
CLEE	Cyclic life and environmental effects
dpa	Iron atom displacement rate, displacement per atom
dA/dN	Fatigue crack growth rate
D-Specs	Design specifications
E	Energy
ENDF/B-VI	Evaluated nuclear data file B-VI
E-Specs	Equipment specifications
EFPY	Effective full-power years
EPRI	Electric Power Research Institute
Fy_{min}	Minimum material yield stress
GI	Generic issue
GPM	Gallon per minute
GSI	Generic Safety Issue
GTR	Generic technical report
HFIR	High-flux isotope reactor
IEB	Inspection and Enforcement Bulletin
IEN	Inspection and Enforcement Notice
IGSCC	Intergranular stress corrosion cracking
INPO	Institute of Nuclear Power Operations
ISI	Inservice inspection
j	Stress level index
J_{Ic}	Fracture resistance factor
K_{IC}	Plane strain fracture toughness
Ksi	Kips per square inch
LAQT	Low alloy quenched and tempered, related to steel

LIST OF ABBREVIATIONS AND ACRONYMS (Continued)

LBB	Leak before break
LCM/LR	Life Cycle Management/License Renewal
LEFM	Linear elastic fracture mechanics
LER	Licensee event report
LOCA	Loss-of-coolant accident
LST	Lowest service temperature
LWR	Light-water reactor
MeV	Million electron volts
MPa	Mega-pascals
n/cm^2	Newton per square centimeter
NDE	Nondestructive examination
NDT	Nil ductility temperature
NEI	Nuclear Energy Institute
N	Number of cycles
n_j	Number of cycles fluctuating at level j
N_j	Number of cycles to failure for stress level j
NSSS	Nuclear steam supply system
ORNL	Oak Ridge National Laboratory
p	Total number of stress levels
pH	Measure of acidity/alkalinity
ppb	Parts per billion
PVRC	Pressure Vessel Research Council
PWR	Pressurized water reactor
PZR	Pressurizer
RAI	Request for additional information
RSIC	Radiation Shielding Information Center
RCP	Reactor coolant pump
RCS	Reactor coolant system
RPV	Reactor pressure vessel
S	Applied stress
SAR	Safety Analysis Report
SC	Structure or component
SCC	Stress corrosion cracking
SER	Safety evaluation report
SG	Steam generator
Sm	Allowable stress
S-N	Fatigue life curve showing stress versus cycles to failure
SNL	Sandia National Laboratory
t	Specimen thickness
T	Test or service temperature
TGSCC	Transgranular stress corrosion cracking
TLAA	Time-limited aging analysis

LIST OF ABBREVIATIONS AND ACRONYMS (Continued)

Tm	Melting point of a metal or alloy
U_f	Fatigue usage factor
U.S. NRC	U.S. Nuclear Regulatory Commission
USI	Unresolved Safety Issue
UT	Ultrasonic testing
WOG	Westinghouse Owners Group

DEFINITIONS

Aging management review

Identification and evaluation of aging effects to determine which aging effects require management during an extended period of operation.

Current licensing basis (CLB)

The set of NRC requirements applicable to a specific plant and a licensee's written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect.

Nuclear power plant

Nuclear power facility of a type described in 10 CFR 50.21(b) or 50.22.

Time-limited aging analyses (TLAAs)

Licensee calculations and analyses that:

- Involve systems, structures, and components within the scope of license renewal, as delineated in § 54.4(a)
- Consider the effects of aging
- Involve time-limited assumptions defined by the current operating term, for example, 40 years
- Were determined to be relevant by the licensee in making a safety determination
- Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b)
- Are contained or incorporated by reference in the CLB

1.0 INTRODUCTION

The objectives of this report are to:

- Identify and evaluate aging effects that degrade component functions that support system intended functions
- Identify and evaluate time-limited aging analyses (TLAAs)
- Provide options, in terms of activities and program attributes, to manage these aging effects, and if necessary address TLAAs

System-level intended functions will be supported by maintaining structure or component (SC) functions that support system intended functions. Hereafter, those SC functions that support system intended functions will be referred to as SC intended functions.

Aging management options identified in this report, when implemented, will ensure that the reactor coolant system (RCS) supports intended function is maintained during an extended period of operation.

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 supports an intended function. SSCs within the scope of the rule are:

1. The safety-related systems, structures, and components that are relied on 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 non-safety-related systems, structures, and components whose failure could prevent any of the functions identified in paragraphs 1 a, 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 United States Nuclear Regulatory Commission's (U.S. NRC's) regulations 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 50.62), 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 then determines if the SC is subject to an aging management review. An SC is subject to an aging management review if the SC:

- Supports or performs an intended function of a system or structure within the scope of Part 54
- Performs an intended function in a passive manner
- Is long-lived

The parts of the RCS supports within the scope of the rule and subject to an aging management review are identified in Section 2.0. Section 2.0 also identifies TLAAs and mechanisms that cause aging effects. The aging management open issues in the nuclear industry have been compiled in an Electric Power Research Institute (EPRI) report [Ref. 1]. The aging management review (Section 3.0) describes age-related degradation mechanisms to identify resulting aging effects. Aging effects and TLAAs are then evaluated to determine the degradation of intended functions. Options for managing the effects of aging and TLAAs that degrade intended functions are then provided in Section 4.0.

The aging management options provided in this evaluation are to be developed into programs by utilities applying for a renewed license. Implementation of these programs demonstrates that aging effects are managed and that the intended functions will be maintained.

1.1 APPLICABILITY

This evaluation is generically applicable to domestic commercial nuclear power plants with the Westinghouse nuclear steam supply system (NSSS), as listed in Table 1-1. 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 a verification of all the bounding information against plant-specific data. This verification will identify that the report is applicable to the plant or what plant-specific data are not covered by this report and will be evaluated as part of the license renewal application.

Table 1-1 lists the plants included in this evaluation for license renewal. As noted from this table, initial commercial operation dates for these plants range from 1968 to 1996.

1.2 AGING MANAGEMENT EVALUATION SCOPE

Specifically, the evaluation of the RCS supports addresses the supports of the reactor pressure vessel (RPV), steam generators (SGs), reactor coolant pumps (RCPs), the pressurizer (PZR), and the PZR surge line.

The configuration of the Westinghouse reactor coolant loop for a four-loop plant is shown in Figure 1-1. This figure exemplifies the basic system components with the exception of the number of loops, the pressurizer, and the surge line.

TABLE 1-1
COMMERCIAL OPERATING WESTINGHOUSE NUCLEAR POWER PLANTS IN THE UNITED STATES

Plant Name	Net MWe	Commercial Operation Date
Beaver Valley 1 & 2	810 & 833	10/76 & 11/87
Braidwood 1 & 2	1120	7/88 & 10/88
Byron 1 & 2	1105	9/85 & 8/87
Callaway	1125	4/85
Catawba 1 & 2	1129	6/85 & 8/86
Comanche Peak 1 & 2	1150	8/90 & 7/93
Diablo Canyon 1 & 2	1073 & 1087	5/85 & 3/86
Donald C. Cook 1 & 2	1020 & 1060	8/75 & 7/78
Farley 1 & 2	814 & 824	12/77 & 7/81
Ginna	470	7/70
Haddam Neck	590	1/68
Indian Point 2	970	8/74
Indian Point 3	965	8/76
Kewaunee	503	6/74
McGuire 1 & 2	1129	12/81 & 3/84
Millstone 3	1146	4/86
North Anna 1 & 2	911 & 909	6/78 & 12/80
Point Beach 1 & 2	485	12/70 & 10/72
Prairie Island 1 & 2	503 & 500	12/73 & 12/74
Robinson 2	683	3/71
Salem 1 & 2	1106	6/77 & 10/81
Seabrook	1150	7/90
Sequoyah 1 & 2	1148	7/81 & 6/82
Shearon Harris	860	5/87
South Texas Project 1 & 2	1250	8/88 & 6/89
Summer	885	1/84
Surry 1 & 2	781	12/72 & 5/73
Turkey Point 3 & 4	666	12/72 & 9/73
Vogtle 1 & 2	1100 & 1097	6/87 & 5/89
Watts Bar 1 & 2	1177	1996 & Indef.
Wolf Creek	1135	9/85
Zion 1 & 2	1040	12/73 & 9/74

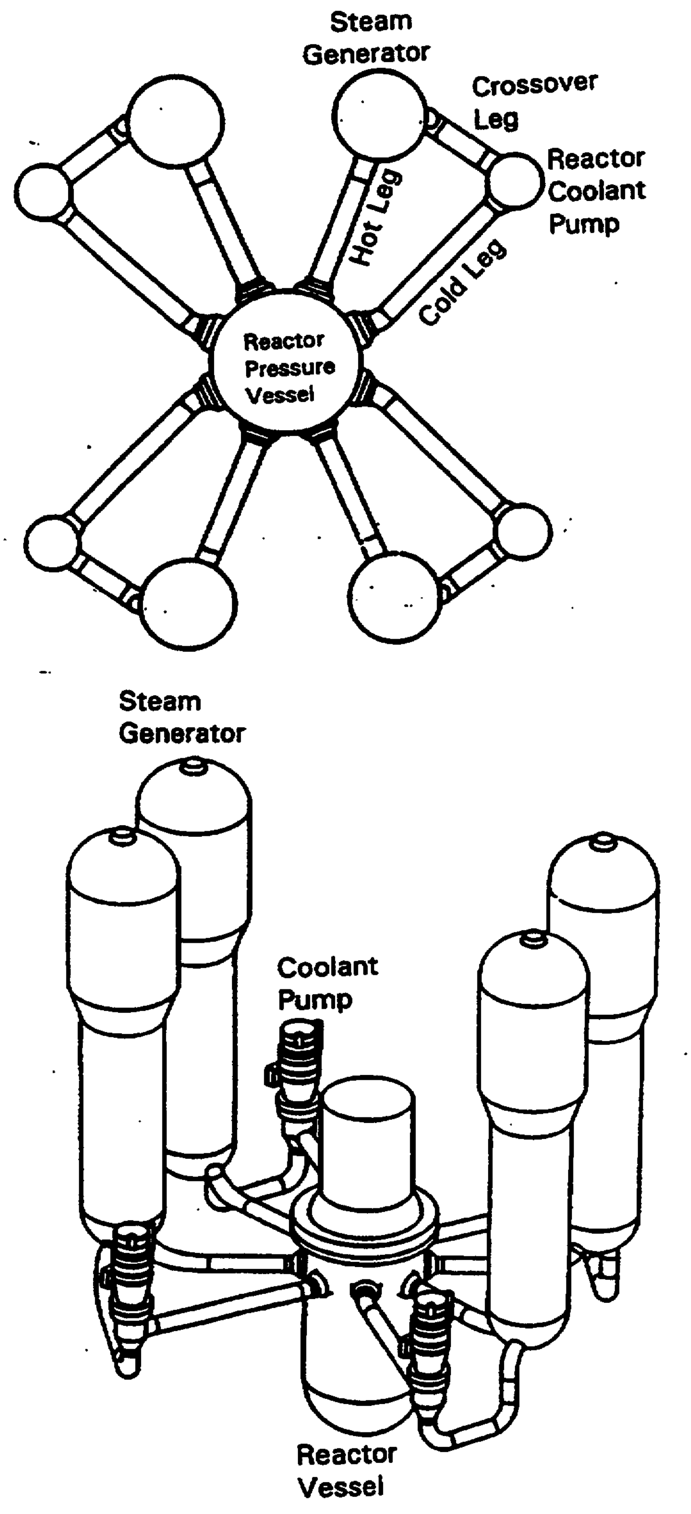


Figure 1-1 **Layout of a Four-Loop Westinghouse Commercial Nuclear Reactor Coolant Loop**

2.0 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES AND AGING EFFECTS

This section identifies the time-limited aging analyses (TLAAs) and aging effects related to the reactor coolant system (RCS) supports. First, the RCS supports are described in general terms. This description includes the boundary of the RCS supports covered in this report. Next, the reason why the RCS supports are within the scope of the license renewal rule is provided. This reason identifies the intended function maintained by the RCS supports. The parts of the RCS supports that are subject to an aging management review are then identified and described in detail. These detailed descriptions identify TLAAs and age-related degradation mechanisms. Finally, aging effects resulting from age-related degradation mechanisms are identified.

2.1 GENERAL DESCRIPTION

The RCS supports are located inside containment. The intended functions of the RCS supports are to maintain the RCS components in equilibrium within spatial positions prescribed by design and ensure the structural integrity and safe operation of the RCS piping and primary components under design conditions (the system intended function, as defined in 10 CFR 54). The supports are designed to meet the American Society of Mechanical Engineers (ASME) Code, Section III, Subsection NF, or the American Institute of Steel Construction (AISC) Specifications [Ref. 2] for plants whose contract date is prior to 1974. Code-allowable stress limits reflect a factor of safety against capacity. Different code factors of safety exist for the loading conditions (i.e., service level A, B, C, D) and modes of failure (e.g., buckling, shear, tension, bearing).

The RCS supports for the plants included in this study share commonality of function, yet differ in the details of their design. Fundamentally, the function of the RCS supports on all plants is to maintain the structural integrity of the reactor coolant loop piping and major equipment for all plant operating and design conditions. However, the support configurations and materials of the plants included in this study vary because of the variety of organizations that design supports. Utilities, architect/engineers, and Westinghouse have each developed and implemented their own unique RCS support designs.

The scope of this report includes:

- Primary component supports for:
    - Reactor coolant pump (RCP)
    - Reactor pressure vessel (RPV) (note that the neutron shield tank is included in the scope and is described in RPV configuration 4; also the support ring is included, as described in configuration 3)
    - Steam generator (SG)
    - Pressurizer (PZR)

- PZR surge line supports, including springs

References made within this report to RCS supports include the above scope.

The boundary between the components and structures is:

- Up to, but not including, integral attachments that are on components (integral attachments are discussed in specific component generic reports, e.g., PZR support skirt boundary at PZR).

- Lugs, nozzles, or welds on component shells are also not included. They are also discussed in specific component generic reports.

- Concrete local to embedment is included, but not concrete adjacent to embedment (this portion is included in the generic report associated with seismic Class 1 structures). Base plates, embedded plates, and anchor bolts are included as part of the local embedment that is within the scope of this report.

Excluded are:

- Pipe whip restraints - covered in another generic technical report
- Masonry walls - none related to RCS supports
- Portions of snubber supports that perform intended functions in an active manner

2.2 REACTOR COOLANT SYSTEM SUPPORTS SUBJECT TO AN AGING MANAGEMENT REVIEW

The RCS supports perform the intended function to maintain the RCS components in equilibrium within spatial positions prescribed by design, thereby ensuring the structural integrity and safe operation of the RCS piping, surge line piping, and primary components under design conditions.

The parts or subcomponents that specifically support this intended function are listed in Table 2-1. It is noted that all of the items listed are subject to an aging management review.

TABLE 2-1
SUMMARY OF PARTS OR SUBCOMPONENTS REQUIRING AGING MANAGEMENT REVIEW

Part or Subcomponent	Aging Management Review Required?
Structural Shapes	Yes
Structural Plates	Yes
Lateral Tie Bars	Yes
Steel Pins	Yes
Bolting	Yes
Embedments	Yes

The RCS supports are considered passive components in that they perform their intended function without moving parts and without a change in configuration or properties. The RCS supports are also long-lived because they are not intended to be replaced during the current or any extended period of operation. The exceptions to this classification are the active portions of snubbers. Since these portions active, they are not considered within the scope of this report.

Since the RCS supports (excluding portions of snubber supports that perform intended functions in an active manner) perform the intended function in a passive manner and are long-lived, they are subject to an aging management review.

2.3 DESCRIPTION

As noted previously, configurations of RCS supports vary from plant to plant. However, they all share the same primary intended function of ensuring the structural integrity and safe operation of RCS piping and primary components. The designs can be grouped into several general basic configurations, which are defined in the following paragraphs and shown in the figures at the end of Section 2.3. It is noted that the support sketch arrangements shown are representative of some configurations in use. The RCS support configuration classification for each plant of this study is summarized in Table 2-2. Specific details of each plant support are not needed. The description of each configuration is given in the following subsections.

There are various primary equipment support concepts used to accommodate the thermal expansion of the reactor coolant loop and equipment. A large number of plants use pinned-end columns to support the SGs and RCPs, which rotate a small amount as the plant heats up. For SG support configuration 2, the skirt is mated to a stationary skirt ring girder anchored to the building structure floor using a number of roller assemblies. These roller assemblies permit horizontal movement of the support skirt relative to the stationary ring girder in the hot leg direction, while preventing movement perpendicular to the hot leg and vertical direction. A set of stops engage the skirt at the end of its thermal travel to provide restraint for seismic and pipe rupture loads. For support configuration 5, SGs and RCPs slide on Lubrite bearing pads located between the components and stationary supports. A large holddown bolt at each pad, or foot, fits in a slot in the stationary support plate oriented in the direction of thermal motion of the component. The holddown bolts also have shear load capability for seismic and pipe rupture restraint.

Since pressurizers remain stationary during plant heatup, the thermal growth that must be accommodated is in the vertical and radial direction at the upper lateral support. This growth is generally permitted by the use of jaw-type supports to engage the pressurizer lugs in the tangential, and sometimes radial, directions. The supports have small gaps in the cold condition, and these gaps are shimmed to nominal zero clearance in the hot condition. These jaw-type supports permit unrestrained vertical movement of the pressurizer lugs.

In cases where a primary component is designed to slide on a support structure to accommodate thermal movement during heatup, special materials are used as wear plates at the support interface with the component. Wear plate materials are: 1) self-lubricated "Lubrite" plate, and 2) Timken Graph-Air tool steel. The type of base material used for the Lubrite plate is American Society for Testing and Materials (ASTM) A-48. It is noted that the extent of relative movement between a component and its support over the lifetime of the plant is quite small. A reactor vessel nozzle pad moves about 3/8-in. during plant heatup, which works out to less than 1 in. movement per year of plant operation.

The embedments subjected to aging management review are those that are between the interface of the structural member of the support within the scope of this report (see Section 2.1) and the concrete.

TABLE 2-2
PRIMARY COMPONENT SUPPORT CONFIGURATION CLASSIFICATION

Plant Name	RPV	SG	RCP	PZR
Beaver Valley 1 & 2	4	1	1	3
Braidwood 1 & 2	1	3	3	1
Byron 1 & 2	1	3	3	1
Callaway	1	3	2	1
Catawba 1 & 2	1	3	3	1
Comanche Peak 1 & 2	1	3	2	1
Diablo Canyon 1 & 2	3	3	1	1
Donald C. Cook 1 & 2	1	3	3	1
Farley 1 & 2	1	3	2	1
Ginna	1	3	2	2
Haddam Neck	4	2	4	3
Indian Point 2	3	1	1	1
Indian Point 3	3	1	1	1
Kewaunee	1	3	2	2
McGuire 1 & 2	1	3	3	1
Millstone 3	4	3	2	3
North Anna 1 & 2	4	1	1	3
Point Beach 1 & 2	2	3	2	2
Prairie Island 1 & 2	1	3	2	2
Robinson 2	1	1	1	2
Salem 1 & 2	1	1	1	1
Seabrook	1	3	2	1
Sequoyah 1 & 2	1	3	2	1
Shearon Harris	1	3	2	1
South Texas Project 1 & 2	1	3	2	1
Summer	1	3	2	1
Surry 1 & 2	4	4	1	3
Turkey Point 3 & 4	1	5	5	2
Vogtle 1 & 2	1	3	2	1
Watts Bar 1 & 2	1	3	2	1
Wolf Creek	1	3	2	1
Zion 1 & 2	1	3	2	1

The primary equipment support embedments are typically cast-in-place anchor bolts, through-wall anchor bolts, or cast-in-place weldments.

Anchor bolt designs include hook bolts, threaded bolts with individual washer plates and nuts, and groups of bolts sharing a common washer plate. Bolt sizes may range as large as 4 in. diameter and may have lengths up to 7 or 8 ft. Concrete expansion anchors are generally not used for the primary equipment supports (they may be used for surge line hangers in some cases).

Embedded weldments are typically fabricated using structural plate material, structural shapes, or a combination of both. The embedded weldments are constructed of the same materials as are the support structures.

2.3.1 Reactor Pressure Vessel Supports

RPV supports are shown in Figures 2-1 to 2-4. The RPV support system must restrain the RPV for all design loading conditions, while allowing the RPV to expand and contract under service temperature conditions. RPV supports provide vertical and tangential support to the RPV. The supports are located near the beltline region of the RPV under the inlet and/or outlet nozzles or under load brackets between nozzles. Load is transferred from the RPV to the supports either through the RPV nozzles or through load brackets between nozzles. In most cases, the nozzle or load bracket interface is a lug-type configuration that is restrained tangentially, after allowing for tangential thermal growth, and vertically by an RPV "shoe." This shoe interface provides tangential and vertical restraint to the RPV, while allowing unrestrained radial thermal growth of the RPV. The shoe is attached to a structural weldment that transfers all shoe loadings to the supporting concrete, which must remain within acceptable temperature and stress limits. In some cases, the RPV support is cooled by forced air circulation or, in some designs, by water.

During normal operation, the RPV supportrience compression loadings resulting from the dead weight of the RPV.

The number of supports per RPV are summarized in Table 2-3.

TABLE 2-3
NUMBER OF SUPPORTS PER REACTOR PRESSURE VESSEL

Number of Reactor Coolant Loops	Number of RPV Supports
2	6
3	3 or 6
4	4

The RPV supports have been grouped into four basic configurations:

- Configuration 1 (RPV)

  This configuration, consisting of an individual shoe and structural weldment at each reactor vessel support point, is shown in Figure 2-1. The structural support pad under the RPV nozzles or support bracket rests on the shoe, and the shoe is fastened to the structural weldment. Some plants have a pair of short (approximately 2-ft. long) stub columns under the weldment. Cooling air or water may be forced through or around the structural weldment to keep the supporting concrete within acceptable temperature limits. For configuration 1 supports, lateral loads are transferred to the concrete either by bearing loads on the vertical end faces of the weldments or by shear at the weldment baseplate, depending on the primary shield wall concrete design.

- Configuration 2 (RPV)

  This configuration is shown in Figure 2-2. The RPV support shoes are mounted on a six-sided structural steel ring girder that is vertically supported at each apex by steel columns extending to a point below the reactor vessel and restrained horizontally at the center of each segment of the ring by structural members embedded in the surrounding concrete.

  The reactor vessel has six supports, four pads, one at each nozzle, and two brackets. Each support bears on a support shoe, which is fastened to the support structure. The support shoe is a structural member that transmits the support loads to the supporting structure. The support shoe is designed to restrain vertical, lateral, and rotational movement of the reactor vessel but allows for thermal growth by permitting radial sliding at each support on bearing plates.

- Configuration 3 (RPV)

  This configuration is shown in Figure 2-3. The RPV support shoes are mounted on a continuous circular ring girder instead of individual weldments. The ring girder is mounted directly on the primary shield wall concrete and transfers loadings from the RPV through the shoe to the reinforced concrete primary shield wall. Water cooling is provided at each support shoe to maintain concrete temperatures at acceptable levels.

- Configuration 4 (RPV)

  This configuration is shown in Figure 2-4(a) and 2-4(b). The RPV support shoes are mounted on the neutron shield tank, which surrounds the RPV. Loadings from the RPV are transferred through the shoe to the shield tank and through the shield tank to the reinforced concrete structure.

2.3.2 Steam Generator Supports

SG supports are shown in Figures 2-5 to 2-8. The SG support system restrains the SG for all design loading conditions and allows free thermal expansion of the RCS piping and the SG itself. Several support systems have been developed to accomplish these functions. To accommodate thermal growth of the RCS hot leg (approximately 1.75 in.), the SGs have been hung from rod hangers, supported by pinned-end columns, attached to rigid frames that slide on the building structure, or designed to slide over a rigid frame attached to the building structure. Lateral supports on the SG allow the RCS to grow thermally through the use of gapped interfaces or snubbers. Thermal growth of the SG itself is normally accounted for by gapped interfaces.

The primary function of the SG supports is to provide vertical and lateral restraint to the SG. The vertical support is attached to the SG near the channel head and tubesheet. The variety of vertical support configurations include rod hangers, pinned-end columns, frame-type structures, and skirts. In most cases, lateral support for the SG is provided at two locations, the first near the channel head or tube sheet and the second near the center of gravity of the SG. Lateral support is accomplished through the use of rigid struts, frame-type structures, and snubbers. Five support configuration groups have been identified for the SG.

- Configuration 1 (SG)

  This configuration, shown in Figure 2-5, consists of a lower support frame structure that transfers both vertical and lateral loads from the SG to the reinforced concrete building structure. The frame structure is attached to the SG channel head just below the tube sheet. An upper support, consisting of a ring girder and typically four or five large-bore hydraulic snubbers, is usually located just below the SG transition cone. A few plants have the upper support located on the SG upper shell (steam drum). The upper support transfers lateral loads from the SG to the building structure. An alternative configuration, used at Indian Point Units 2 and 3, combines the upper and lower support into a single support frame. Support of the SG is accomplished through four support lugs located on the channel head and loosely bolted to the SG support structure. Guides are provided in the structure to allow for radial expansion of the SG shell. The entire support structure moves on Lubrite plates. In addition to the four snubbers located at the top of the support structure, two snubbers are provided at the bottom of the structure to prevent sudden large movements.

- Configuration 2 (SG)

  In this configuration, shown in Figure 2-6(a), the entire SG support consists of a skirt support structure attached to the bottom of the SG. The skirt support transfers lateral and vertical loadings from the SG to the reinforced concrete building structure. There is no upper support structure.

- Configuration 3 (SG)

  This SG support configuration, shown in Figure 2-6(b), includes vertical pinned-end columns attached to the SG channel head or tube sheet, a lower support including compression bumpers and/or snubbers, and an upper support consisting of a ring band with compression bumpers and/or snubbers. The columns provide vertical support to the SG. The lower support compression bumpers and beam and the upper support compression bumpers and snubbers transfer lateral SG loadings to the building structure.

- Configuration 4 (SG)

  The SG is hung from rod hangers in this configuration, which is similar to that shown in Figure 2-7. Lateral loads are resisted at the lower and upper support locations by compression bumpers and snubbers.

- Configuration 5 (SG)

  In the SG support configuration shown in Figure 2-8, the SG is supported by structural steel plates mounted on stationary columns and bolted to the reinforced concrete building structure and fastened to the SG support feet on the channel head. Vertical and lateral loadings from the SG are transferred to the building structure through the plates. An upper support, consisting of a ring girder and compression bumpers, is also used to transfer lateral loads from the SG to the building structure.

2.3.3 Reactor Coolant Pump Supports

RCP supports are shown in Figures 2-5, 2-7, 2-9, and 2-10. The support system for the RCP must restrain the RCP for all design loading conditions, while allowing free thermal expansion of the RCS piping and the RCP itself. Several support systems have been developed to accomplish these functions. To accommodate thermal growth of the RCS (approximately 2.00 in. radially from the RPV), the RCPs have been hung from rod hangers, supported by pinned-end columns, and attached via rigid frames that slide on the building structure. Lateral support of the RCP accounts for RCS thermal growth through the use of gapped interfaces or snubbers. Unrestrained thermal growth of the RCP is normally provided through gapped interfaces.

RCP supports provide vertical and lateral restraint to the RCP. Vertical support is provided through load brackets or feet attached to the pump casing. In most cases, lateral support for the RCP is provided at this same location. The types of vertical support configurations include rod hangers, pinned-end columns, and frame-type structures. Lateral support is provided through tension tie rods, compression struts, frame-type structures, and snubbers. Six basic configuration groups have been identified for the RCP:

- Configuration 1 (RCP)

  This support configuration consists of pinned-end columns that support the RCP vertically and a frame structure surrounding the pump that transfers lateral loadings to the reinforced concrete building structure through compression bumpers, snubbers, or tie rods. The pump support may be interconnected to the SG lower support frame structure (see SG configuration 1). Figure 2-5 shows this configuration interconnected with the SG support. An alternative configuration, used at Indian Point Units 2 and 3, consists of a frame structure with a sliding Lubrite base plate to accommodate reactor coolant loop expansion and a system of tie rods and anchor bolts to restrain movement beyond calculated limits. A second alternative configuration utilizes a stationary frame structure anchored at the basemat and sometimes braced at the primary and/or secondary shield walls. This frame structure is designed to allow the pump to slide on the frame during plant heatup.

- Configuration 2 (RCP)

  This support configuration, shown in Figure 2-9, consists of pinned-end columns to provide vertical support and tension tie rods, or compression bumpers or snubbers, to provide lateral support. Columns, tie rods, and compression bumpers are attached to the RCP at the pump casing feet and transfer loads to the building structure.

- Configuration 3 (RCP)

  This support configuration is similar to RCP configuration 2, except that a frame structure is used to provide lateral support. The frame structure is not interconnected to the SG support.

- Configuration 4 (RCP)

  In this configuration, no lateral supports are provided to transfer loads from the pump to the building structure. It is supported vertically by spring hangers.

- Configuration 5 (RCP)

  This configuration is shown in Figure 2-10 and is similar to the one described for the SG lower support system under SG configuration 5.

- Configuration 6 (RCP)

  This configuration, shown in Figure 2-7, uses an overhead hinged frame-type structure from which the pump is hung. Horizontal restraint is provided for dynamic loads while permitting thermal expansion with the use of snubbers for the hinged frame bracing in addition to horizontal snubbers.

2.3.4 Pressurizer

Two examples of PZR support are shown in Figures 2-11 and 2-12. The PZR support system must restrain the PZR for all design loading conditions, while allowing free movement of the PZR under the range of temperatures encountered during plant operation. The PZR is supported at the base of its support skirt either directly by the reinforced concrete building structure or by a structural steel frame supported by the building structure. This support interface transfers vertical and lateral PZR loads to the building structure. Lateral support near the center of gravity of the PZR is provided for plants with higher seismic input. A ring girder, as described below and shown in Figure 2-11, may or may not be used. For plants included in this study, three basic PZR support configurations have been identified:

- Configuration 1 (PZR)

  The pressurizer is supported at the base by the PZR skirt support and near the center of gravity of the PZR by an upper lateral support, consisting of either a ring girder with compression elements or compression and/or shear-carrying strut elements alone. The skirt is bolted onto the concrete floor or to a structural steel frame. See Figure 2-11.

- Configuration 2 (PZR)

  Similar to configuration 1 (Figure 2-11), the PZR is supported only at the base with the PZR skirt support and there is no upper support. The skirt is bolted onto the concrete floor or bolted to a structural steel frame.

- Configuration 3 (PZR)

  Shown in Figure 2-12, the PZR is supported at the PZR skirt support, which rests on a rigid ring girder that is suspended from the operating floor by four hanger columns. Anti-sway brackets welded to the shell of the PZR fit into striker plate assemblies embedded in the concrete floor close to the center of gravity of the PZR vessel.

2.3.5 Reactor Coolant System Pressurizer Surge Line Supports

The general design concept for the PZR surge line is similar for the Westinghouse two-, three-, and four-loop plants. The line generically features a pipe routed between a nozzle on the RCS hot leg and a nozzle on the PZR. The physical arrangement features five diameter bends and some form of horizontal offset (loop) to provide flexibility and account for terminal-end thermal displacements created by the RCS hot leg and PZR during plant operation.

The two-loop plants generically have 12-in. surge lines with a 12 x 14 in. reducer in the PZR nozzle. Three- and four-loop plants generically have 14-in. surge lines that connect directly to the PZR nozzle.

Thermal expansion considerations incorporated in pipe routing generally result in the use of mechanical/hydraulic snubbers (active components) and variable springs as support/restraint devices. These devices provide no restrictions on pipe expansion. Rigid structural members and/or component standard supports may also be used to support the pipe in directions where the pipe is sufficiently flexible to absorb thermal expansion without overstress. Lube plates may be used to minimize drag effects where high friction loads between the pipe and structural support members prevent the pipe from moving freely in unrestrained directions during thermal expansion.

2.4 ENGINEERING AND DESIGN DATA

Engineering and design data important to the aging evaluation of primary component supports include: structural materials; temperature; relative humidity; neutron fluency; fracture toughness; and codes, standards, and regulations. Environmental factors (temperature, humidity, and radioactivity) potentially affect the properties of the material and thus potentially contribute to the degradation of material properties in steels and concrete. Design and construction codes provide the technical basis and data for plant construction and operation. Regulatory requirements provide guidelines for managing safety-related issues. The data given herein are representative of a Westinghouse pressurized water reactor (PWR) plant and provide judged bounds.

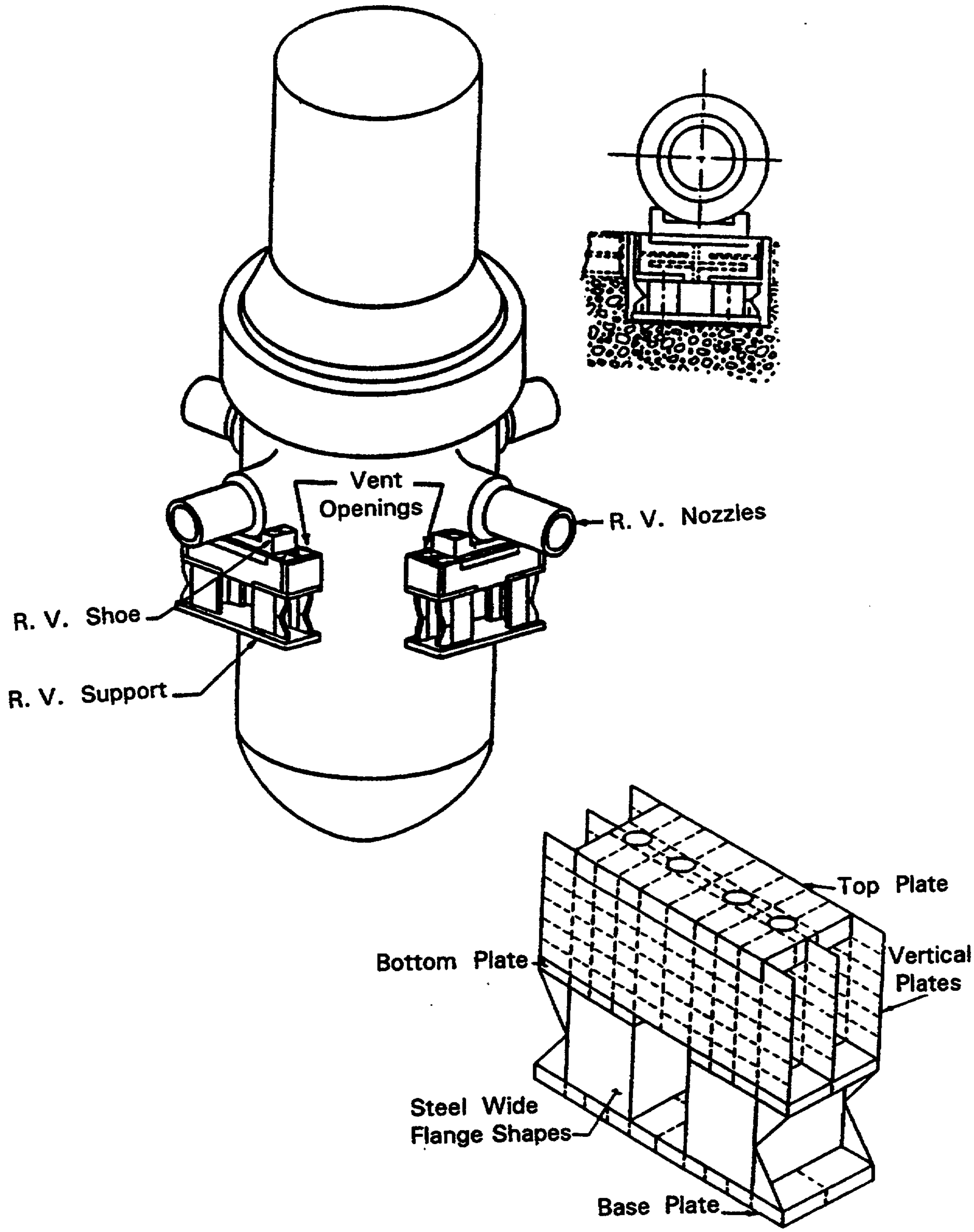


Figure 2-1 Reactor Pressure Vessel Configuration 1 Support

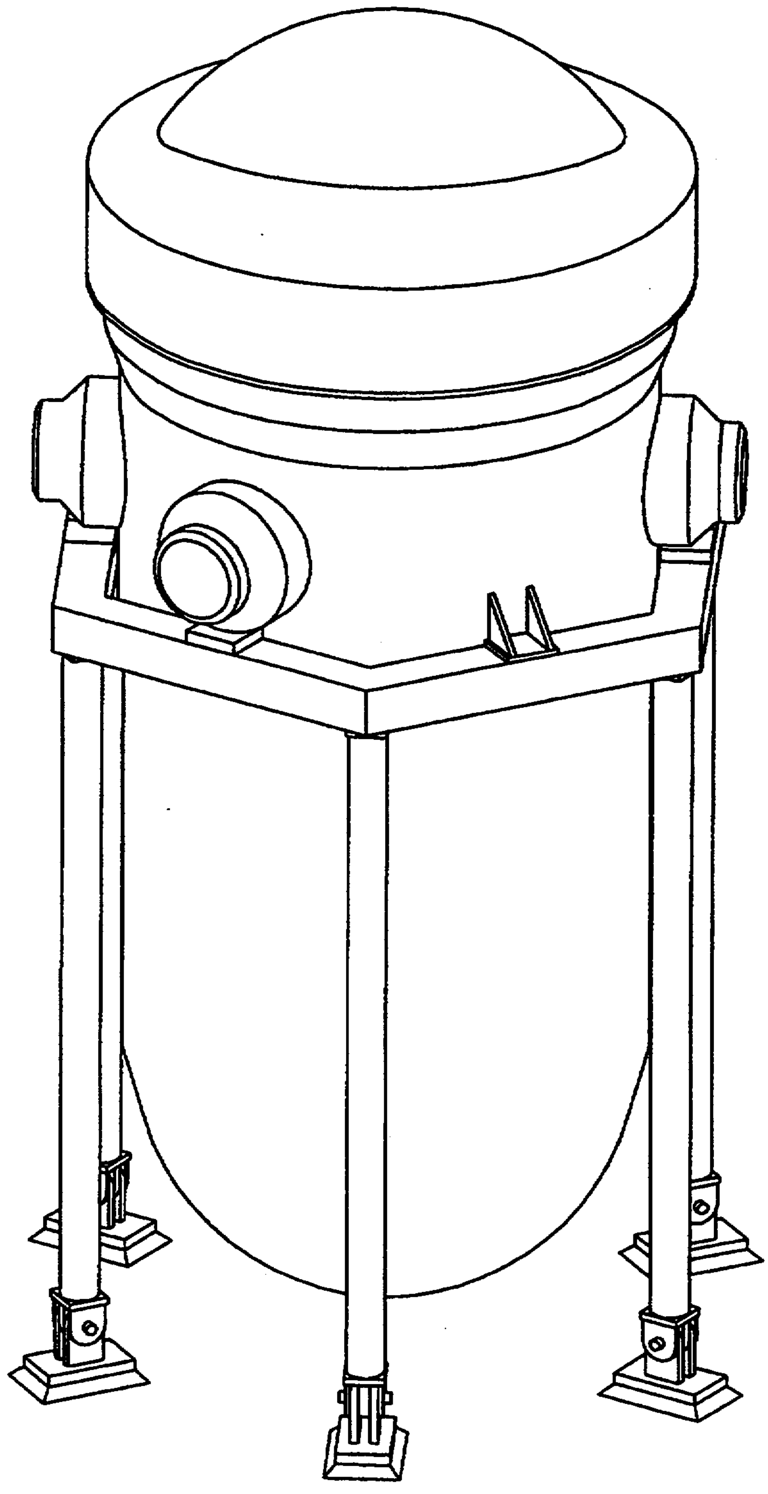


Figure 2-2 Reactor Pressure Vessel Configuration 2 Support

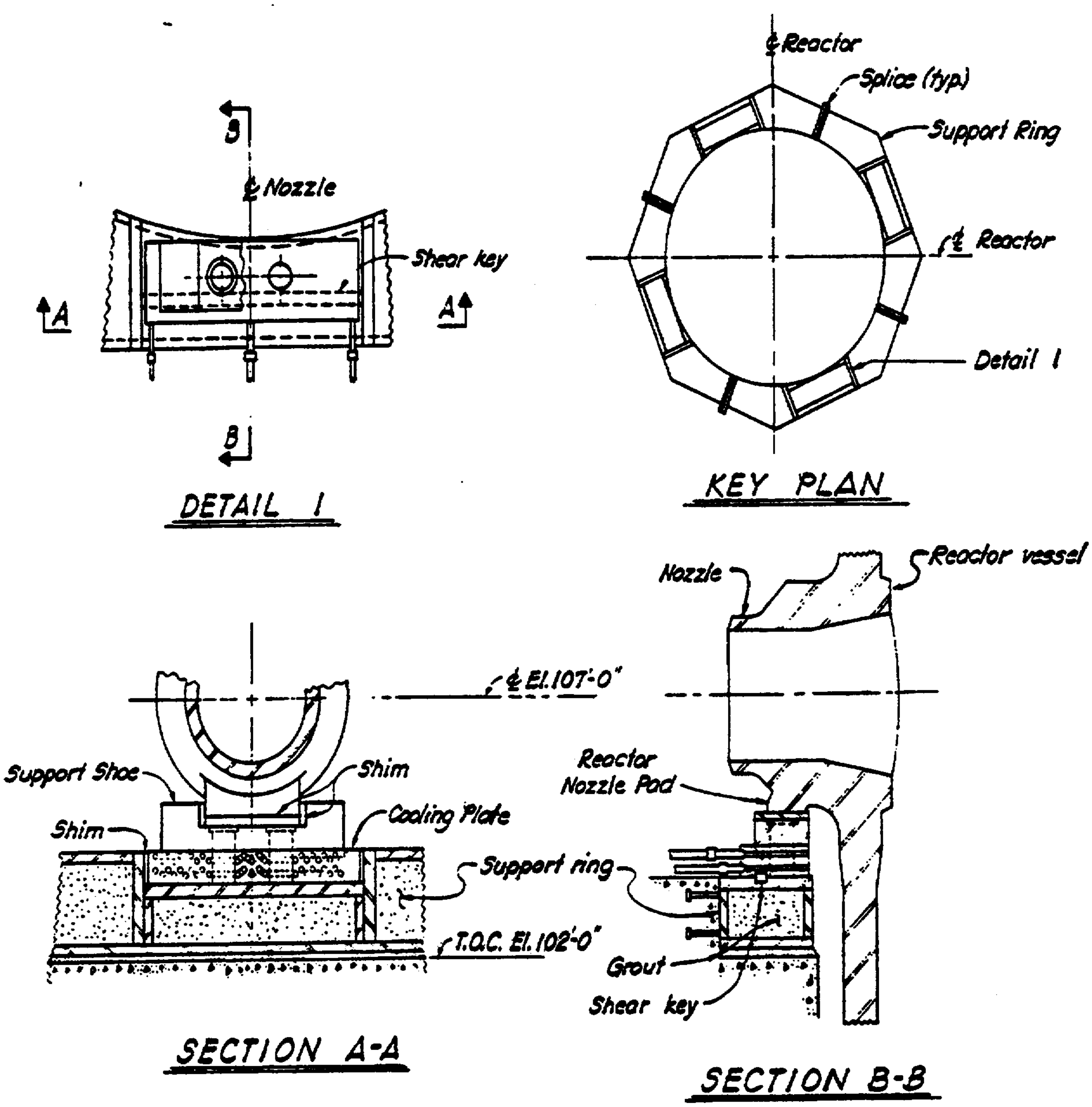


Figure 2-3 Reactor Pressure Vessel Configuration 3 Support

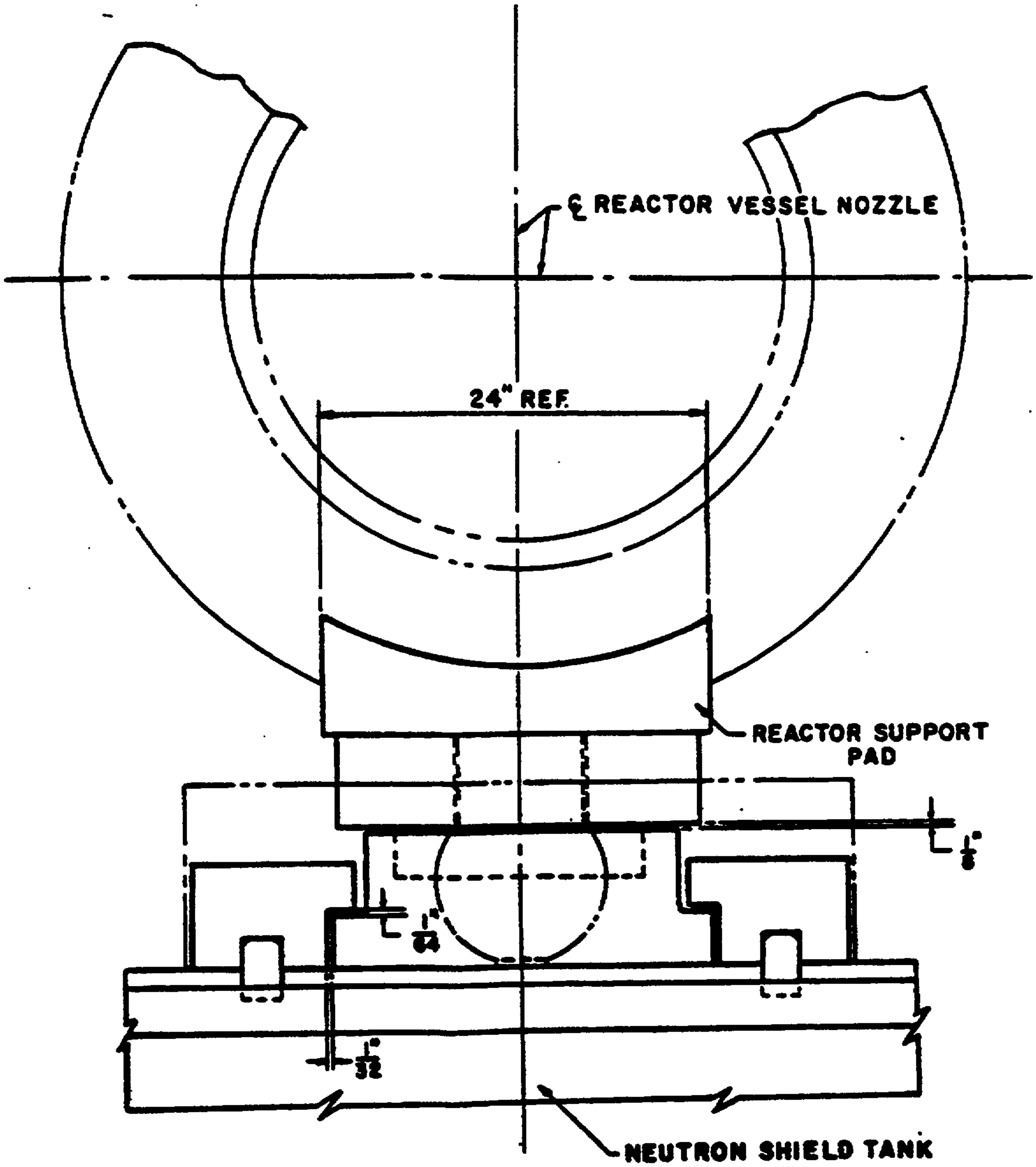


Figure 2-4(a) Reactor Pressure Vessel Configuration 4 Support

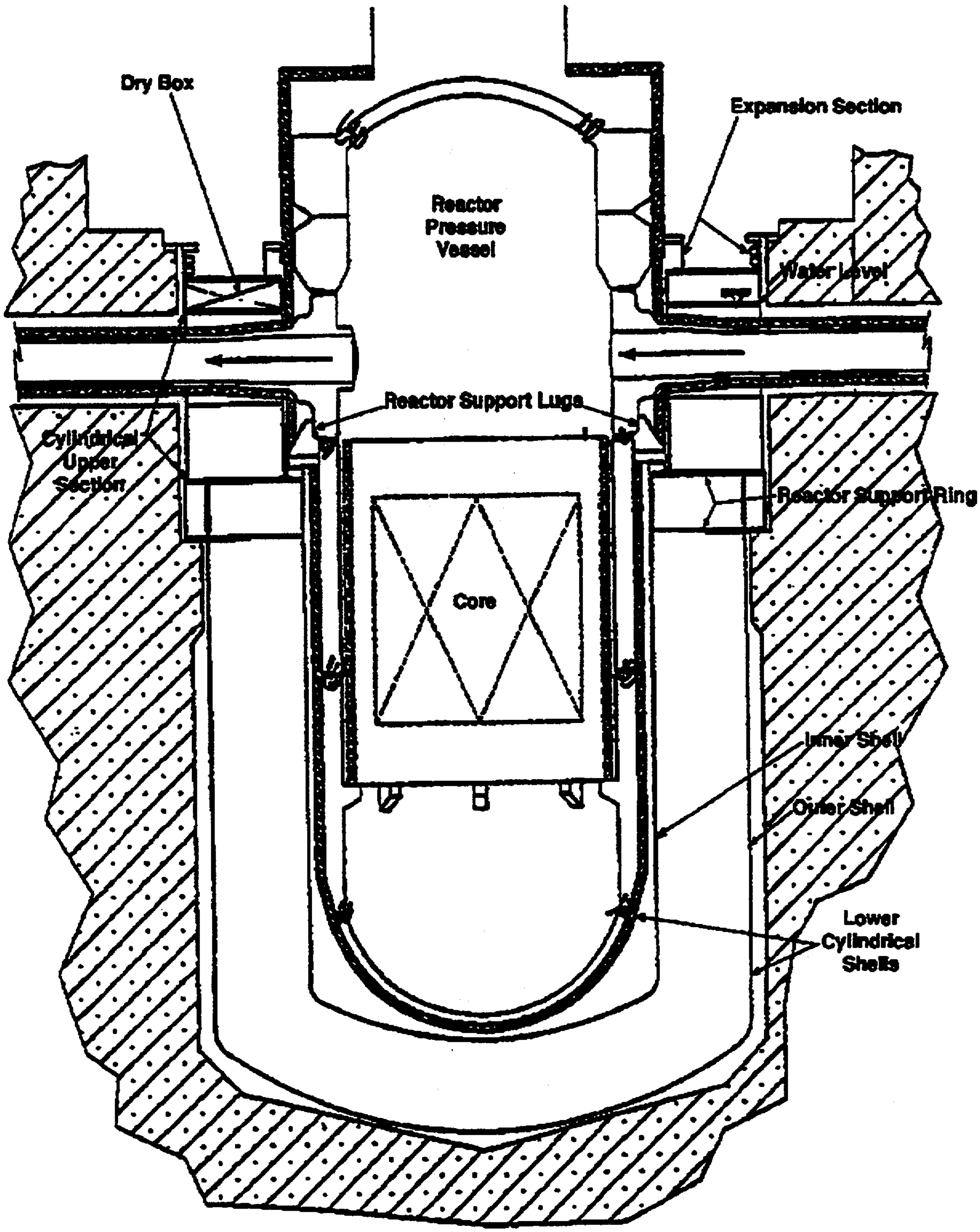
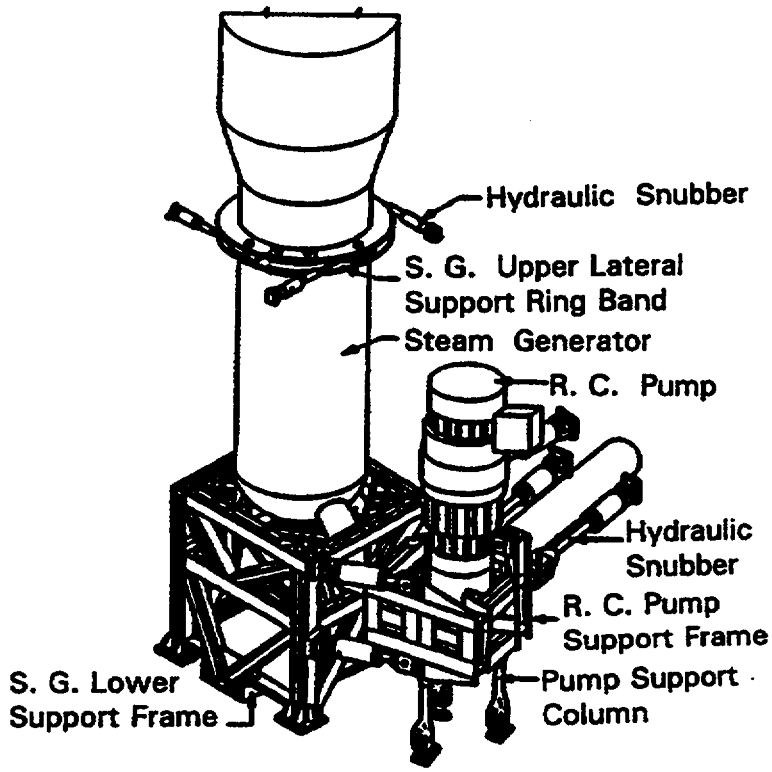


Figure 2-4(b) Reactor Pressure Vessel Configuration 4 Support



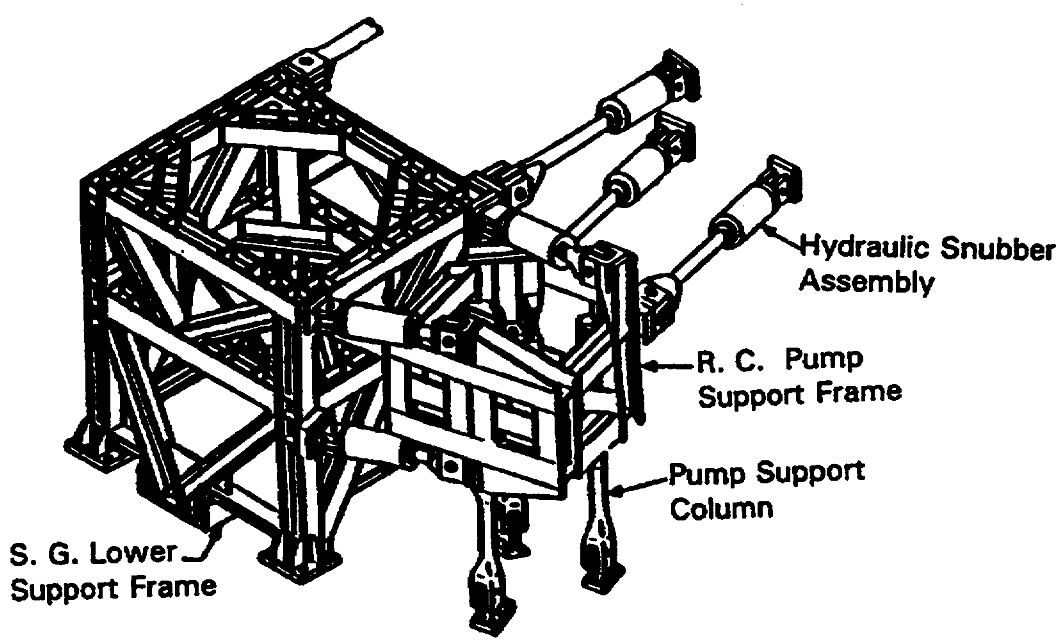
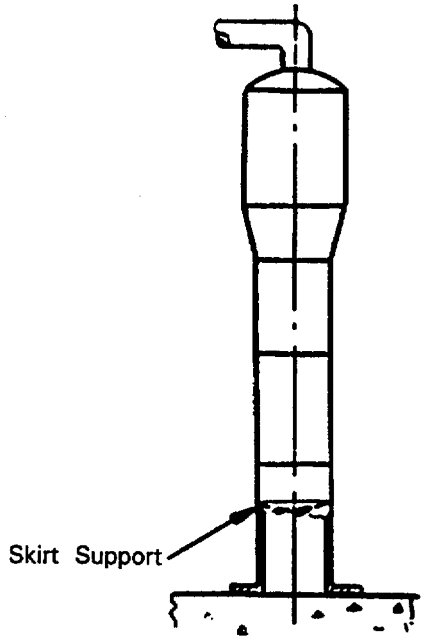
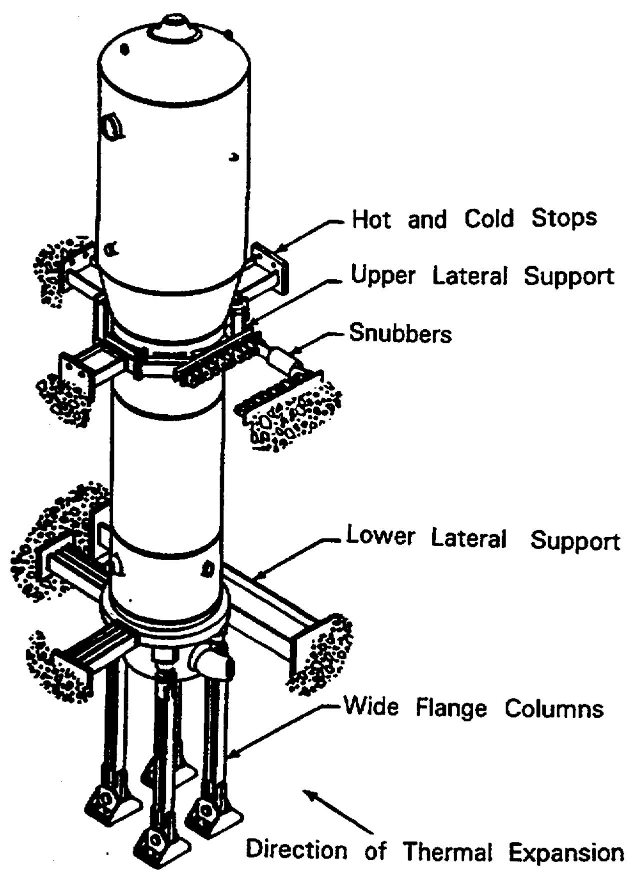


Figure 2-5 Reactor Coolant Pump and Steam Generator Supports, Configuration 1



(a) Configuration 2



(b) Configuration 3

Figure 2-6 Steam Generator Support Configurations 2 and 3

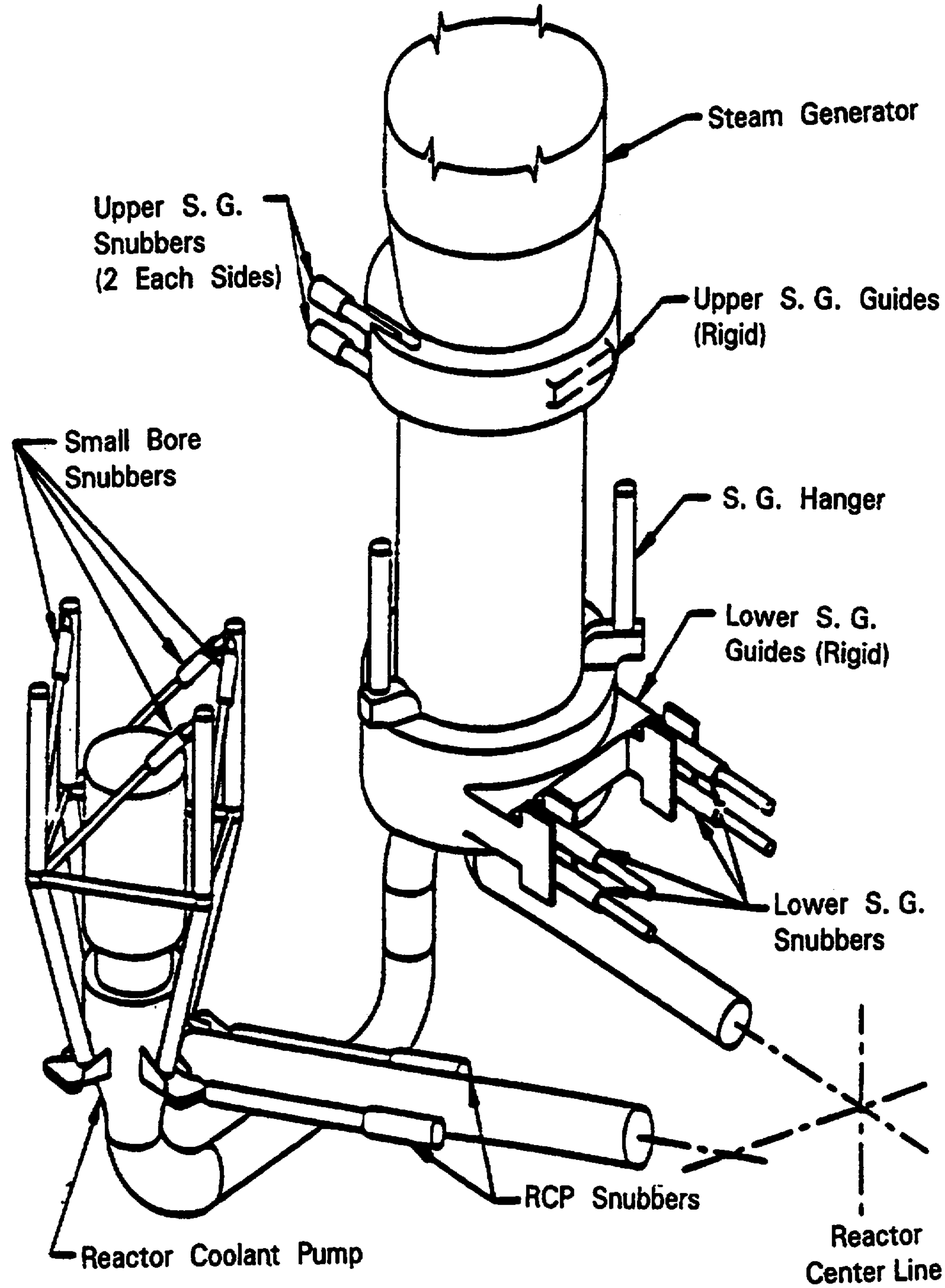


Figure 2-7 Steam Generator Support Configuration 4, and Reactor Coolant Pump Support Configuration 6

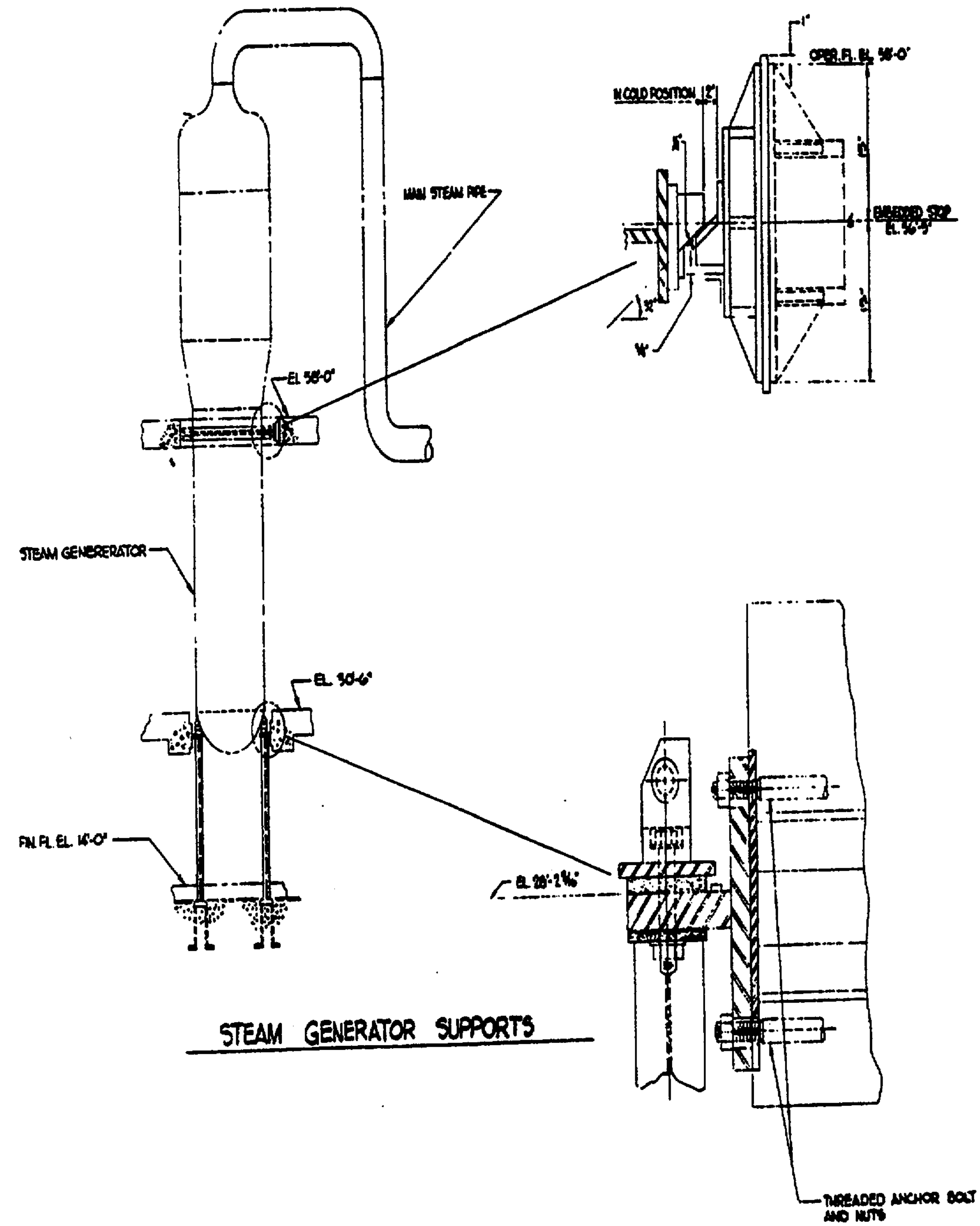
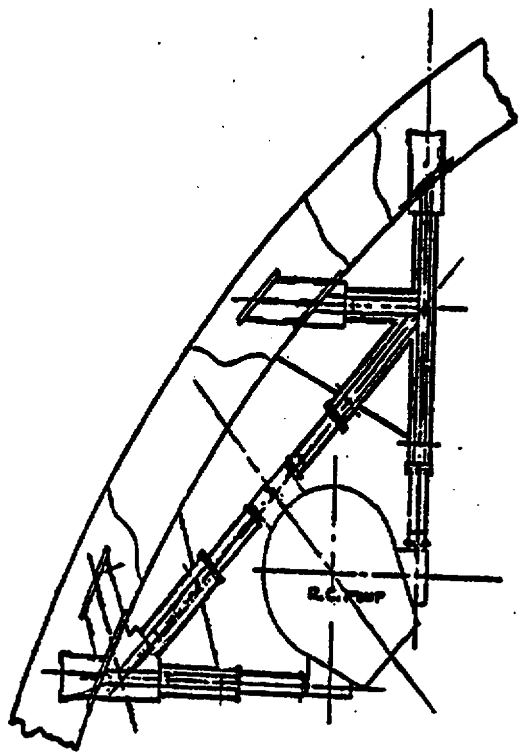
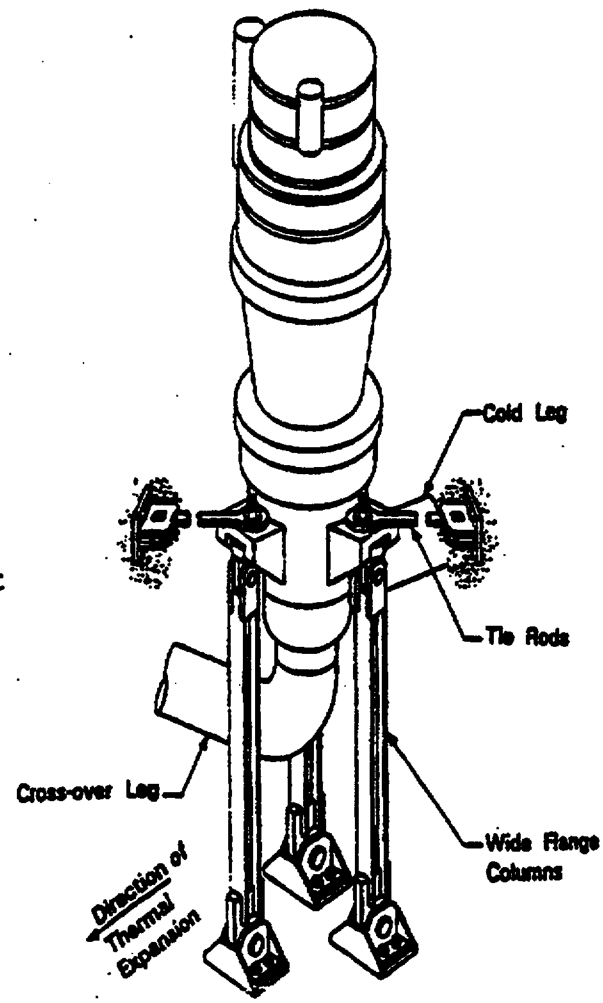


Figure 2-8 Steam Generator Support Configuration 5



Plan View of Lateral Restraint
Reactor Coolant Pump Support
Configuration 3



Reactor Coolant Pump Support
Configuration 2

Figure 2-9 Reactor Coolant Pump Support Configurations 2 and 3

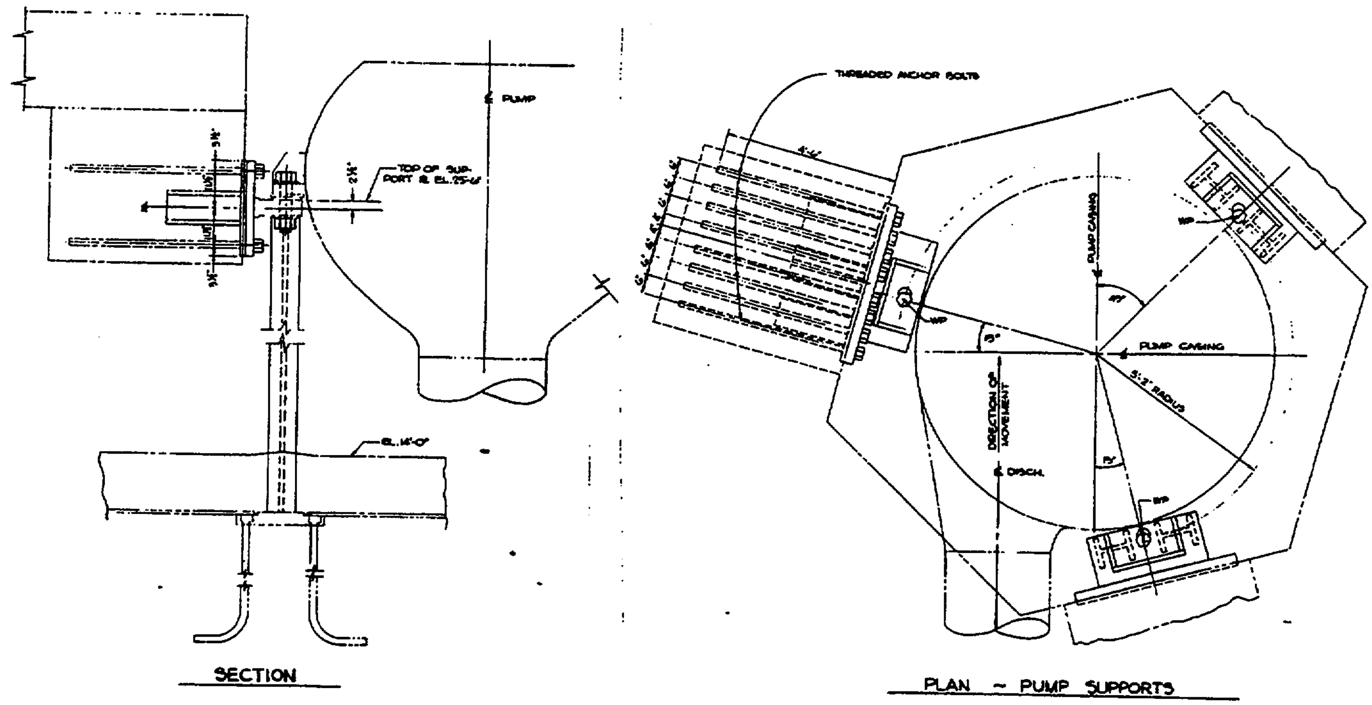


Figure 2-10 Reactor Coolant Pump Support Configuration 5

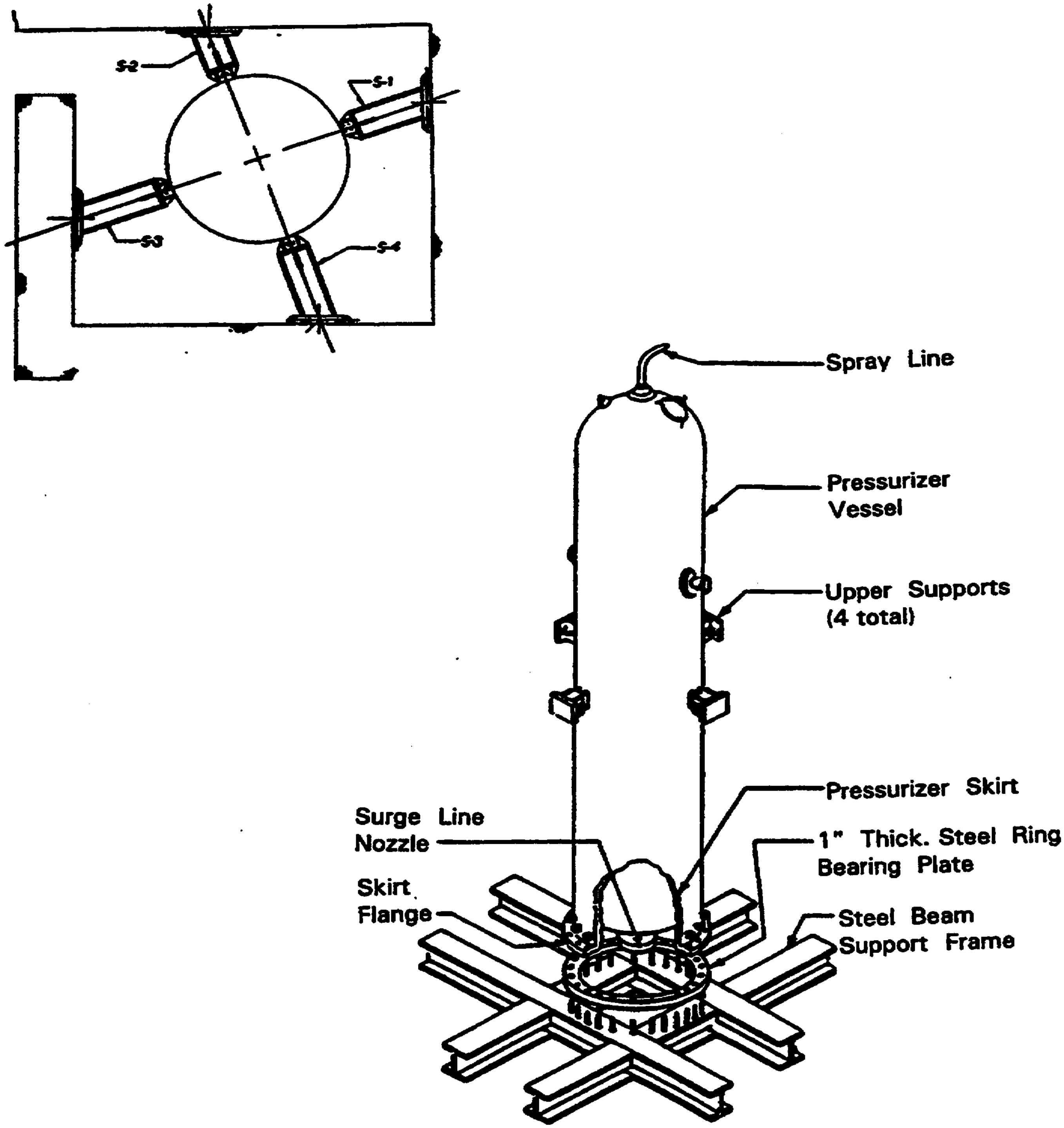


Figure 2-11 Pressurizer Support Configuration 1

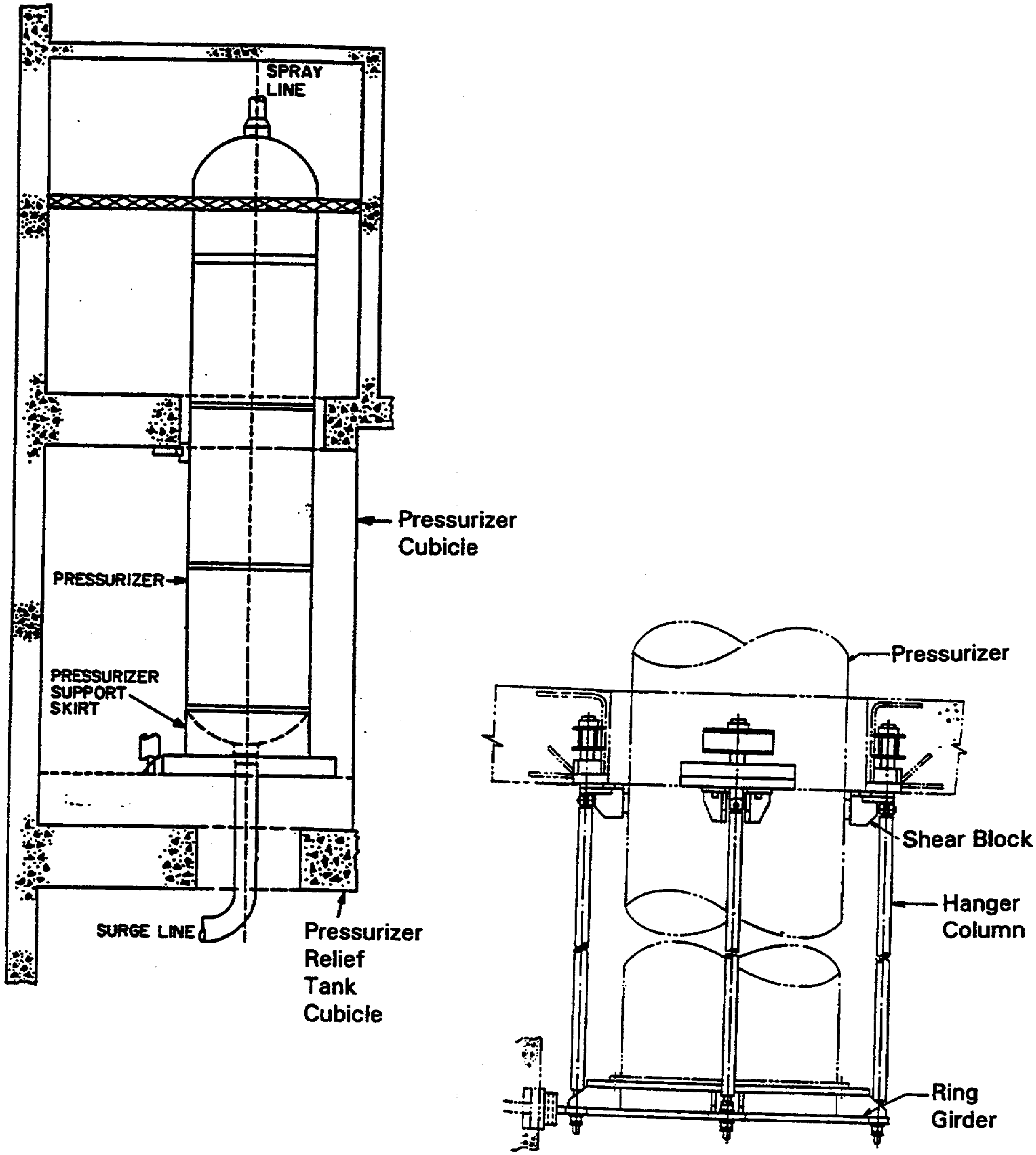


Figure 2-12 Pressurizer Support Configuration 3

2.4.1 Structural Materials

RCS component supports are constructed from various types of steel. The specific grade of steel may vary slightly from plant to plant. Table 2-4 provides a listing of the ASTM designation of the most commonly specified materials. Plants built to ASME III NF requirements use corresponding SA materials or ASTM materials meeting the requirements of ASME Code cases 1644, N-71, or N-249. The categories of materials include structural steels, high-strength low alloy steels, and carbon steel pipe. For the neutron shield tank and pressurizer skirt, material A516 plate, Grade 60 and 70, is used.

In general, the type of welding filler materials and fluxes used conform to ASME Boiler and Pressure Vessel Code, Section III, Paragraph NF 2400. For supports designed to the AISC Specification [Ref. 2], the standards of the American Welding Society (AWS) D1.0 [Ref. 3] are used to define welding requirements.

A variety of bolt materials are used for concrete embedments. Such materials are ASTM A36 or A588 threaded rod, to ASTM A354, A490, or A540 bolting material.

2.4.2 Temperature

The ambient temperature inside containment varies from 50°F to 150°F. The ambient pressure is 14.7 psia or lower. Temperatures at respective equipment/support interfaces are a function of normal operating temperatures of the supported equipment. Equipment temperatures that are to be considered at the support interface are given in Table 2-5. These are nominal temperatures for the two-, three-, or four-loop plants. These temperatures are used for design unless thermal analyses or tests are performed and documented to justify the use of lower values. Note that the temperature profile along a support member can significantly decrease as the distance between the support member and the component that is acting as the heat source increases. This reduction is a function of heat loss, distance, insulation, member shape, and material. The reduction in temperature is generally considered during the design of supports, so the higher temperatures reflected in Table 2-5 occur only locally within the support near the support/component interface.

The strength and durability of concrete is partially a function of temperature as well as other factors. Therefore, all supports are designed or cooled so that concrete temperatures at the support/concrete interface do not exceed 200°F. Deterioration may start at 300°F (Subsection 3.2.8). If cooling is required to maintain the concrete temperature within acceptable limits, the cooling air requirements are made by the plant owner and may be included in the support designs (e.g., RPV support configuration 1; see Figure 2-1).

2.4.3 Relative Humidity

Relative humidity inside the containment building varies between 15 and 70 percent during normal operation. During normal refueling or abnormal operation conditions, relative humidity may reach 100 percent.

TABLE 2-4
MATERIALS - PRIMARY COMPONENT SUPPORTS

Component	RPV	SG	RCP	PZR
Structural Shapes 1.5 in. < t ≤ 4.0 in.	A572(1)	A572(1)	A572(1)	A572(1)
	A588(2)	A588(2)	A588(2)	A588(2)
Structural Plates Thickness (t) ≤ 1.5 in.	A514(7) A572(1)	A514(7) A572(1)	A572(1)	A572(1)
1.5 in. < t ≤ 4.0 in.	A572(3)	A572(3)	A572(3)	A572(3)
t ≤ 4.0 in.	A588(2)	A588(2)	A588(2)	A588(2)
4.0 in. < t ≤ 8.0 in.	A588(2)	A588(2)	A588(2)	A588(2)
Pump Lateral Tie Bars	-	-	A514(7)	-
Structural Plates	-	-	A543(8)	-
Pipes and Tubes	A106(9)	A106(9)	A106(9)	A106(9)
	A618(10)	A618(10)	A618(10)	A618(10)
Steel Pins	A540(11)	A540(11)	A540(11)	A540(11)
	A434(12)	A434(12)	A434(12)	A434(12)
Bolting(4)	A490(13)	A490(13)	A490(13)	A490(13)
	A354(14)	A354(14)	A354(14)	A354(14)
	A540(11)	A540(11)	A540(11)	A540(11)
Forgings	A668(6)	A668(6)	A668(6)	A668(6)

Notes:

(1) ASTM Designation, high-strength low-alloy steel, Grade 50.
(2) ASTM Designation, corrosion-resistant high-strength, low-alloy steel of 50 ksi minimum yield strength, Grade A and B.
(3) ASTM Designation, high-strength low-alloy steel, Grade 42.
(4) Nuts used with the bolts are to be the same material as bolts or as specified in the ASTM specification for bolting material. Washers to be quenched and tempered carbon steel (0.40-percent minimum carbon content) or as specified in the ASTM specification for the bolting material.
(5) Deleted.
(6) ASTM Designation, steel forgings, carbon and alloy, Class N, American Institute of Steels and Iron (AISI) 4340 steel.
(7) ASTM Designation, quenched and tempered alloy, 90 ksi minimum yield stress.
(8) ASTM Designation, pressure vessel plates, alloy steel quenched and tempered, Class 2, 100 ksi minimum yield.
(9) ASTM Designation, seamless carbon steel pipe for high-temperature service, Grade C.
(10) ASTM Designation, hot-formed welded and seamless high-strength, low alloy structural tubing, Grade II or III.
(11) ASTM Designation, alloy steel bolting material for special applications, Grade B-23, Class 4.
(12) ASTM Designation, steel bars, alloy hot-wrought or cold-finished, quenched and tempered, Class BD, AISI 4340 steel.
(13) ASTM Designation, heat-treated, steel structural bolts, AISI 4140 steel.
(14) ASTM Designation, quenched and tempered alloy steel bolts and studs, Grade BC, AISI 4340 steel.

TABLE 2-5
EQUIPMENT/SUPPORT INTERFACE TEMPERATURES

Supported Component	Temperature
RPV (Outlet Nozzle)	625°F
RPV (Inlet Nozzle)	565°F
SG (Primary Side)	625°F
SG (Secondary Side)	565°F
RCP	565°F
PZR & Surge Line	653°F

2.4.4 Radiation Environment

In the assessment of the state of embrittlement of light water RPV supports, an evaluation of the neutron exposure of the materials comprising highly irradiated portions of the supports is required. This exposure evaluation must include an assessment of the entire neutron and gamma ray energy spectrum at the support locations as well as a calculation of the iron atom displacements or displacement per atom (dpa) experienced by the materials.

The characterization of the environment should also include a definition of radial, axial, and azimuthal distributions within the support components.

2.4.5 Fracture Toughness

Fracture toughness, commonly denoted by K_{IC}, is a material property associated with linear elastic fracture mechanics (LEFM). The fracture toughness of a material is a measure of its capability of resisting crack initiation or crack extension. K_{IC} is associated with brittle fracture, meaning that the plasticity at the crack tip is negligible when fracture occurs. For sufficiently constrained structures, such as heavy section plates and shells, stiffened plates, etc., the fracture behavior might be brittle even though the material is known to be quite ductile.

K_{IC} is a true material property determined from test specimens with sufficiently large thickness. The toughness measured from thin specimens is not a true material property because toughness measurements decrease with increasing thickness. The measurement approaches a limiting value, K_{IC}, at which the plane strain or the maximum constraint condition is established. The thickness requirement for the plane strain condition is satisfied if $\beta < 0.4$, using:

$$\beta = (K_{IC}/\sigma_{ys})^2/t$$

where:

σ_{ys} = yield strength
t = thickness of the specimen

Therefore, the K_{IC} concept offers a conservative approach to fracture problems. Under the theory of LEFM, an existing crack will not grow if the applied stress intensity factor, K, is less than K_{IC}. The method for determining K_{IC} is given in ASTM Specification E-399 [Ref. 4].

When deformation is inelastic, usually for cases where the thickness of the structure section is thin and the material is ductile, the R-curve (material resistance curve) or the J_{IC} approach should be considered because the actual fracture resistance capability of the material is greater than that predicted by the K_{IC} approach for this condition. The method for determining J_{IC} is given in ASTM Specification E-813 [Ref. 5].

Fracture toughness is an inherent material property that depends on the chemical composition, manufacturing process, heat treatment, and service temperature. Most of the structural steels, e.g., ASTM 36, 516, 517, 533, 572, 588, HY-80, HY-130, etc., exhibit a transition temperature, below which the fracture toughness is rapidly reduced. This is known as the nil-ductility temperature (NDT). Some structural steels may lose fracture toughness when performing prolonged service under high-flux neutron radiation or at high temperatures. This age-related deterioration is characterized by an increase in the NDTT.

Generally, the material has acceptable design toughness when, under given conditions of stress and temperature, the material can withstand loading to its design limit in the presence of flaws. Toughness also implies that under certain specified loading conditions, such as thermal loading, the material has the capability to arrest the growth of a flaw.

2.4.6 Codes, Standards, and Regulations

Table 2-6 lists the codes, standards, and regulations applicable to various areas of nuclear power plant operations related to the supports that are within the scope of this report. The licensing requirements or commitments applicable to each of the plants are established in the specific plant current licensing basis and may include these U.S. NRC Regulatory Guides and IE Bulletins.

As stated in Section 1.0, plants included in this evaluation cover the operating licenses from 1968 to 1993. Prior to 1974, as defined by contract date, the AISC Specification was used. Table 2-7 lists the code editions and which code or specification (ASME Code, Section III, Subsection NF or AISC Specification [Ref. 2]) applies for the different plants.

2.5 TIME-LIMITED AGING ANALYSES

TLAAs are those licensee calculations that:

- Consider 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 its 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

Per Part 54 requirements, it is necessary to demonstrate that the:

- TLAAs remain valid for the extended period of operation

- TLAAs have been projected to the end of the extended period of operation, or

- Effects of aging on the intended functions will be adequately managed for the extended period of operation.

TLAAs have been identified as part of the original design process for the supports that are within the scope of this report. Both ASME Section III, Subsection NF [Ref. 12] and AISC [Ref. 2] provide for fatigue design based on allowable stress range reductions when exceeding 20,000 cycles. However, fatigue is not part of design qualification analyses for the component supports within the scope of this report since they are not subject to high fatigue usage factors and significant stress cycles in excess of 20,000. In Subsection 3.2.6 and Section 3.3 of the report fatigue and TLAAs are further discussed. It is concluded that no additional analyses are required to be performed by the utility for demonstration that TLAAs are acceptable for the extended period of operation since all required demonstration analyses are contained in the report (Subsection 3.2.6 and Section 3.3).

2.6 GENERAL MAINTENANCE PRACTICES

Maintenance programs follow the ASME Code as well as the American Concrete Institute (ACI) recommendations. The regulations and rules that govern the inspection of primary component and surge line supports begin at the top level with the Code of Federal Regulations. Document 10 CFR 50.55a references Section XI of the ASME Code [Ref. 6]. Requirements are given in the following ASME Section XI, 1989 edition, subsections:

- IWF-1000, "Scope and Responsibility," includes Class 1 component supports that are within the scope of this GTR.

- IWF-2000 defines the examination and inspection requirements.

- IWF-2520 and IWA-2240 lead to VT-1 and VT-3 visual inspection methods defined in IWA-2200 or alternative methods of examination.

- IWF-2000, IWF-3000, IWF-7000, IWA-4000, and IWA-7000 address acceptance standards, repair, and replacement.

Concrete structures and surfaces, including coated areas, are visually examined for evidence of off-normal conditions using methods such as given in ACI 201.1R-68 [Ref. 9], "Guide for Making a Condition Survey of Concrete in Service." ACI 349.3R-96 [Ref. 27], "Evaluation of Existing Nuclear Safety-Related Concrete Structures," provides guidance for inspecting concrete embedments. Corrective actions may be taken following recommendations given in [Ref. 9]:

- ACI 201.2R, "Guide to Durable Concrete"

- ACI 222R, "Corrosion of Metals in Concrete"

- ACI 207.3R, "Practices for Evaluation of Concrete in Existing Massive Structures for Service Conditions"

- ACI 224.1R, "Causes, Evaluation, and Repair of Cracks in Concrete Structures"

- ACI 224R, "Control of Cracking in Concrete Structures"

TABLE 2-6
APPLICABLE CODES, STANDARDS, AND REGULATIONS

Codes/Standards/ Regulations	Subjects and Descriptions
AISC [Ref. 2]	Structural design code similar to the ASME Code Section III, Subsection NF but used for plants constructed before 1974.
ASME Section XI [Ref. 6]	Inservice inspection, Subsection IWF provides the requirements for inspection and acceptance of Class 1, 2, 3, and MC supports.
IEB 88-08 (U.S. NRC)	Thermal stresses in piping connected to RCS identifies thermal stratification potentials for unisolatable portions of the RCS and advises utilities to review their designs for potential impact.
IEB 88-11 (U.S. NRC)	PZR surge line thermal stratification requires plants with operating licenses to perform a VT-3 inspection on pipe, supports, whip restraints, and anchor bolts to determine gross discernible 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.
INPO SER 87-25	Surge line thermal cycling observed during RCS pressurization heatup and cooldown.
Reg. Guide 1.147 [Ref. 7]	Inservice inspection code case acceptability - ASME Section XI, Division 1.
ASME Code Case N-491 [Ref. 8]	Provides alternative rules for examination of Class 1, 2, 3, and MC component supports for light-water cooled power plants, Section XI, Division 1.
ACI Manual of Concrete Practices [Ref. 9]	Defines criteria, practices, and guidelines for the design and maintenance of concrete structures.
USI A-12 (U.S. NRC) NUREG-0577 [Refs. 10 and 11]	Addresses low fracture toughness and lamellar tearing.
Generic Letter 88-05 (U.S. NRC)	Corrosive effects of RCS leakage.
GSI-15 (U.S. NRC)	Radiation effects on reactor vessel supports.
GSI-29 and IEB 82-02 (U.S. NRC)	Bolting degradation.

TABLE 2-7

REACTOR COOLANT SYSTEM STRUCTURAL SUPPORT DESIGN CODE OR SPECIFICATION

Plants	Code/Year
Beaver Valley 1 & 2	AISC/'69
Braidwood 1 & 2	NF/'74
Byron 1 & 2	NF/'74
Callaway	NF/'74
Catawba 1 & 2	NF/'74
Comanche Peak 1 & 2	NF/'74
Cook 1 & 2	AISC/'69
Diablo Canyon 1 & 2	AISC/'69
Farley 1 & 2	AISC/'69
Ginna	AISC/'63
Haddam Neck	AISC/'63
Indian Point 2	AISC/'63
Indian Point 3	AISC/'63
Kewaunee	AISC/'69
McGuire 1 & 2	AISC/'69
Millstone 3	NF/'74
North Anna 1 & 2	AISC/'69
Point Beach 1 & 2	AISC/'63
Prairie Island 1 & 2	AISC/'69
Robinson 2	AISC/'63
Salem 1 & 2	AISC/'69
Seabrook	NF/'74
Sequoyah 1 & 2	AISC/'69
Shearon Harris	NF/'74(1)
South Texas 1 & 2	NF/'74
Summer	NF/'74(1)
Surry 1 & 2	AISC/'63
Turkey Point 3 & 4	AISC/'63
Vogtle 1 & 2	NF/'77
Watts Bar 1 & 2	NF/'74(1)
Wolf Creek	NF/'74
Zion 1 & 2	AISC/'63

Notes:

(1) AISC/'69, Evaluated per NF/'74

2.6.1 Inspections

Inspection data collected are used to make conclusions regarding actions needed to address degradation. Recordable indications provide documentation of certain conditions discovered during plant inservice inspection (ISI). Indications are generally reported and documented for evaluation. IWF-3000 [Ref. 6] gives specific standards for acceptance or rejection. Cause of degradation, along with historical failures, forms the basis to identify causes of damage, which include age-related degradation mechanisms. This includes the effects of irradiation (neutron embrittlement), environment, mishandling, improper maintenance, and the like.

2.6.2 Steel Supports

Visual inspection (VT-3, structural condition) is the primary method used; however, inspections using ultrasonic testing (UT), dye penetrant, magnetic particle, and radiation monitoring are sometimes employed. An inspection cycle using VT-3 methods is followed, consistent with the IWF-2400 [Ref. 6] inspection schedule. Recordable indications include corrosion, cracks, etc. In Table 2-8, acceptable inspection and monitoring methods used for steel structures to detect specific effects are given.

2.6.3 Concrete Supports

Concrete structures are mostly affected by the adverse environment in which they are contained: elevated temperatures, high humidity, irradiation, etc. In addition, aging of concrete structures in nuclear power plants contributes to the deterioration of structural materials. In-plant concrete degradation mechanisms require defining methods of inspection that will detect manifestations of these mechanisms on the structure.

The durability of concrete has historically been good. However, concrete degradation is a significant concern since it provides restraint for the support anchorage system. Concrete structural integrity is dependent on aging effects that are influenced by environmental conditions and construction materials. This includes the Portland cement included with the aggregate mix as well as the reinforcing steel.

In Table 2-9, a summary of maintenance methods and conditions for which inspections are made that pertain to the concrete supports is given. The indicated responses detail which methods detect particular conditions associated with concrete support degradation. ISI records these conditions at regular maintenance intervals. Inspection methods sometimes used include visual, measurement, radiation monitoring, etc. Recordable indications include cracks, spalling, staining, rebar corrosion, etc.

2.6.4 Piping Supports - Pressurizer Surge Line

Supports are subject to periodic visual examination (VT-3) through a sampling basis based on the plant code of record. Passive supports such as slide supports, box supports, and lube plates are examined to the acceptance standards defined in ASME Section XI, Subsection IWF-3400 to ensure the structural integrity and desired freedom of movement of the pipe.

2.6.5 Aging Degradation Operating Experience

No documentation related to industry operating experience associated with aging has been found for the supports within the scope of this report. However, aging management issues that may potentially cause degradation to these supports have been identified and are summarized in an EPRI report [Ref. 1]. This summary provides the basis of identifying aging mechanisms. This is discussed further in Section 3.1.

2.7 CONCLUSIONS - AGING MECHANISMS

From the review of industry issues and maintenance history, the following aging mechanisms are identified as potentially significant for the RCS supports within the scope of this generic technical report:

- SCC
- Corrosion and aggressive chemical attack
- Neutron embrittlement
- Thermal aging embrittlement
- Mechanical wear
- Fatigue
- Creep and stress relaxation
- Concrete degradation
- Low fracture toughness and lamellar tearing

These mechanisms cause effects that can result in degradation of structural integrity. The effects of these mechanisms and their significance to the RCS supports are evaluated in Section 3.0. Also provided in Section 3.0 is a discussion of industry issues that are related to aging of RCS supports.

TABLE 2-8
MONITORING OF REACTOR COOLANT SYSTEM STEEL SUPPORTS

Method	Periodic?	Detects Corrosion	Detects/Predicts General (Environmental) Deterioration	Detects External Damage	Detects Integrity at Bolted or Welded Connections	Detects Debris, Loose or Missing Parts	Detects Clearances, Settings, Physical Displacements
Visual (In-field)	Y	Y	Y	Y	Y	Y	Y
Visual (Remote)	Y	Y	Y	Y	N	Y	N
Radiation Monitoring	Y	N	Y	N	N	N	N
UT	S	N	N	Y	Y	N	N
Dye Penetrant/ Magnetic Particle	S	N	Y	Y	Y	N	N

Notes:

Y - Yes
N - No
S - Supplemental

TABLE 2-9
MONITORING OF REACTOR COOLANT SYSTEM CONCRETE SUPPORTS

Method	Periodic?	Detects Staining	Detects Rebar Corrosion	Detects Cracks	Detects General (Environmental) Deterioration, Spalling, Porosity	Detects Volume Change	Detects Compressive Strength	Detects/Predicts Concrete Material Characteristics due to Aging, Temperature Effects, Irradiation
Visual (In-field)	Y	Y	Y[1]	Y	Y	N	N	N
Measurement	Y	N	N	N	N	Y	N	N
Radiation Monitoring	Y	N	N	N	N	N	N	Y
Core Drilling	N	N	N	N	N	N	Y	Y
Rebound Hammer	N	N	N	Y	Y	N	N	Y

Notes:

Y - Yes

N - No

(1) Visual detection is possible only if rebar corrosion occurs

3.0 AGING MANAGEMENT REVIEW

This section describes each of the significant age-related degradation mechanisms that affect the reactor coolant system (RCS) supports and evaluates how the effects caused by these mechanisms can potentially degrade the intended functions of the RCS supports. No time-limited aging analyses (TLAAs) are identified that need to be evaluated. Where necessary, the means by which the aging effect can be managed have been identified. The specific aging management options are described in Section 4.0.

3.1 INDUSTRY ISSUES

No documentation related to industry operating experience associated with aging has been found for the RCS supports. Technical issues related to the original design basis have been identified and documented. The aging management issues in the nuclear industry have been compiled in an Electric Power Research Institute (EPRI) report [Ref. 1]. Two aging effect issues relevant to the RCS supports were identified from this EPRI report: (1) aggressive chemical attack, and (2) corrosion. There are three additional RCS primary component support issues that have been identified and are being addressed by utilities in the current licensing term: (1) low fracture toughness and lamellar tearing [Unsolved Safety Issue (USI) A-12 and NUREG-0577]; (2) stress corrosion cracking (SCC) of high-strength bolting materials; and (3) neutron embrittlement of reactor vessel supports by low-temperature, low-fluence irradiation. Pertinent references and cross-references to subsections within this report where the aging effects associated with these issues are discussed is given in Table 3-1.

Utilities need to develop implementation procedures to address seismic qualification of experience under USI A-46. However, it is noted that passive equipment such as piping as well as major pieces of equipment in the nuclear steam supply system (e.g., reactor pressure vessel (RPV), steam generator (SG), reactor coolant pump (RCP), pressurizer (PZR)) are excluded from the USI A-46 scope due to ruggedness. This issue is not related to aging.

Other issues related to the original design basis have been documented, such as:

- U.S. NRC IEB 88-08, thermal stresses in piping connected to RCS
- U.S. NRC IEB 88-11, pressurizer surge line thermal stratification

However, they are not directly related to the scope of this report.

It is also noted that there are several ongoing national and international programs [Refs. 40 to 42] on nuclear plant aging management research for concrete and other structural components.

TABLE 3-1

INDUSTRY ISSUES

Issue	Reference	Report Section
Aggressive Chemical Attack	License Renewal Industry Reports Summary, TR-104305 [Ref. 1] NUMARC Class 1 Structures Industry Report [Ref. 44] Generic Letter 88-05, U.S. NRC	Subsections 3.2.2, 3.2.8(b), 4.2.1
Corrosion	License Renewal Industry Reports Summary, TR-104305 [Ref. 1] Generic Letter 88-05 U.S. NRC ACI Report 222R-89 [Ref. 31]	Subsections 3.2.2, 3.2.8, 4.2.1
Low Fracture Toughness and Lamellar Tearing	USI A-12 (U.S. NRC) NUREG-0577 [Refs. 10 and 11]	Subsection 2.4.5, 3.2.9
Stress Corrosion Cracking (SCC) of High-Strength Bolting Materials	EPRI NP-5769 IEB 82-02 (U.S. NRC) GSI-29	Subsections 3.2.1, 4.2.2
Neutron Embrittlement of Reactor Vessel Supports by Low-Temperature, Low-Fluence Irradiation	GSI-15 (U.S. NRC) NUREG/CR-5320 [Ref. 17]	Subsections 2.4.4, 3.2.3, 3.2.8, 4.1

3.2 AGING EFFECT REVIEW

3.2.1 Stress Corrosion Cracking

Mechanism Description

SCC is a localized non-ductile cracking failure resulting from an unfavorable combination of sustained tensile stress (residual or externally applied), material condition, and environment. As illustrated in Figure 3-1, SCC can occur only when these three factors exist simultaneously. Materials respond differently to environmental and stress conditions. Under an appropriate combination of susceptible material condition and sufficient tensile stress in a corrosive environment, SCC can occur. The crack surface morphology can be intergranular (IGSCC) or transgranular (TGSCC). IGSCC proceeds along the grain boundaries, while TGSCC advances without apparent preference for the grain boundaries. IGSCC and TGSCC can occur in the same alloy, depending on the environment and the microstructure of the metal. In most cases, SCC involves crack initiation, subcritical crack growth, and failure when the crack reaches a critical size and the tensile strength of the remaining material is exceeded.

Tensile stresses greater than or equal to the yield strength of the material are required to induce SCC in a pressurized water reactor (PWR) environment. These stresses can result from applied loads, residual stresses (from cold work, welding, or other material process), or a combination of applied and residual stresses. The time required for a part to fail by SCC generally increases with decreasing stress. If the amount of stress is below the threshold stress level, the SCC occurs at such a slow rate that service life is not affected. For each alloy-environment combination, there is an effective minimum or threshold stress required for SCC to occur. If known for a specific material condition, this threshold value must be used with considerable caution since the environmental conditions may change during service.

At present, there appears to be no general environmental pattern that causes SCC in various metals. Generally, only a few chemical species in the environments are effective in causing SCC in a given material. The species responsible for SCC need not be present in high concentrations. SCC is well known in various aqueous mediums. The presence of oxidizers and the temperature of the environment appear to have pronounced influence on SCC tendencies. For example, in light-water reactor (LWR) environments, one of the most aggressive contributors to SCC is the dissolved oxygen concentration. Equally important is the level of halogens (i.e., chlorides and fluorides). In the primary water of a PWR, the oxygen concentration is controlled to below 5 ppb (parts per billion), and the level of halogens is controlled below 150 ppb during operation. As shown in Figure 3-2 [Ref. 13], these limits have been maintained to preclude the occurrence of SCC on austenitic stainless steels included in the pressure-retaining components of the RCS, such as the reactor pressure vessel (RPV).

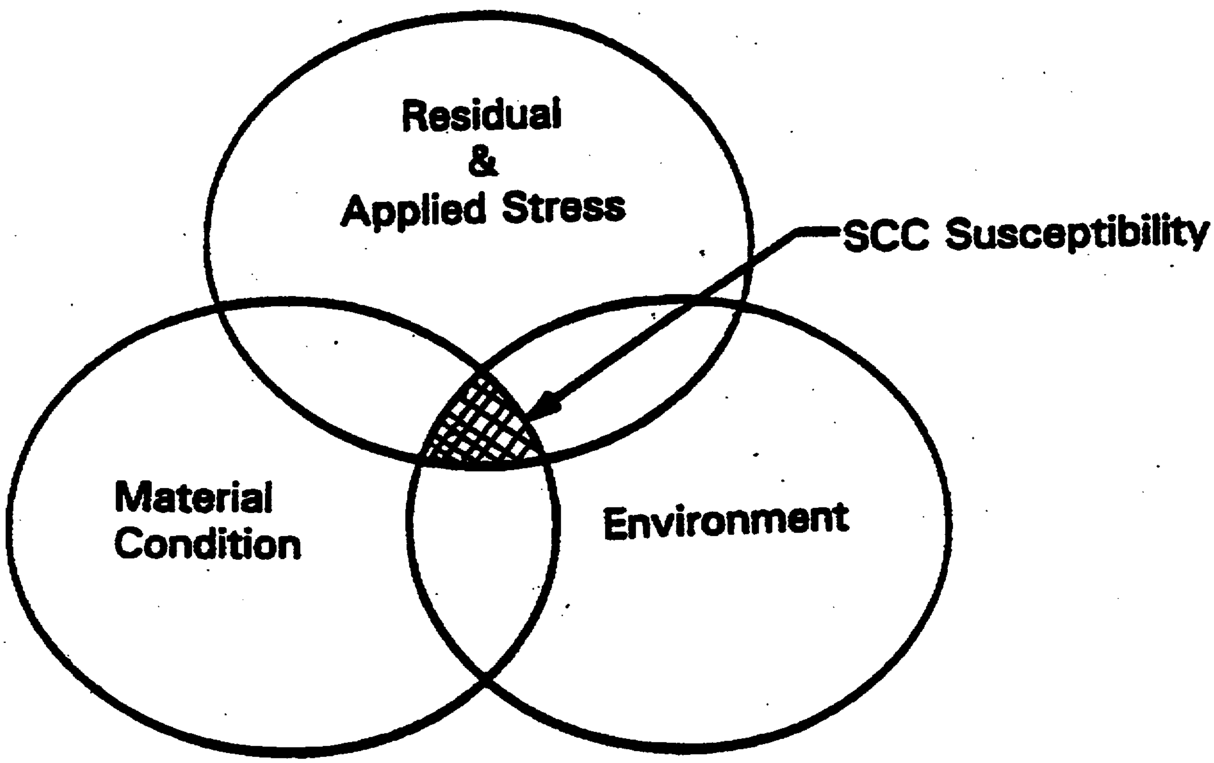


Figure 3-1. Parameters that Influence Stress Corrosion Cracking

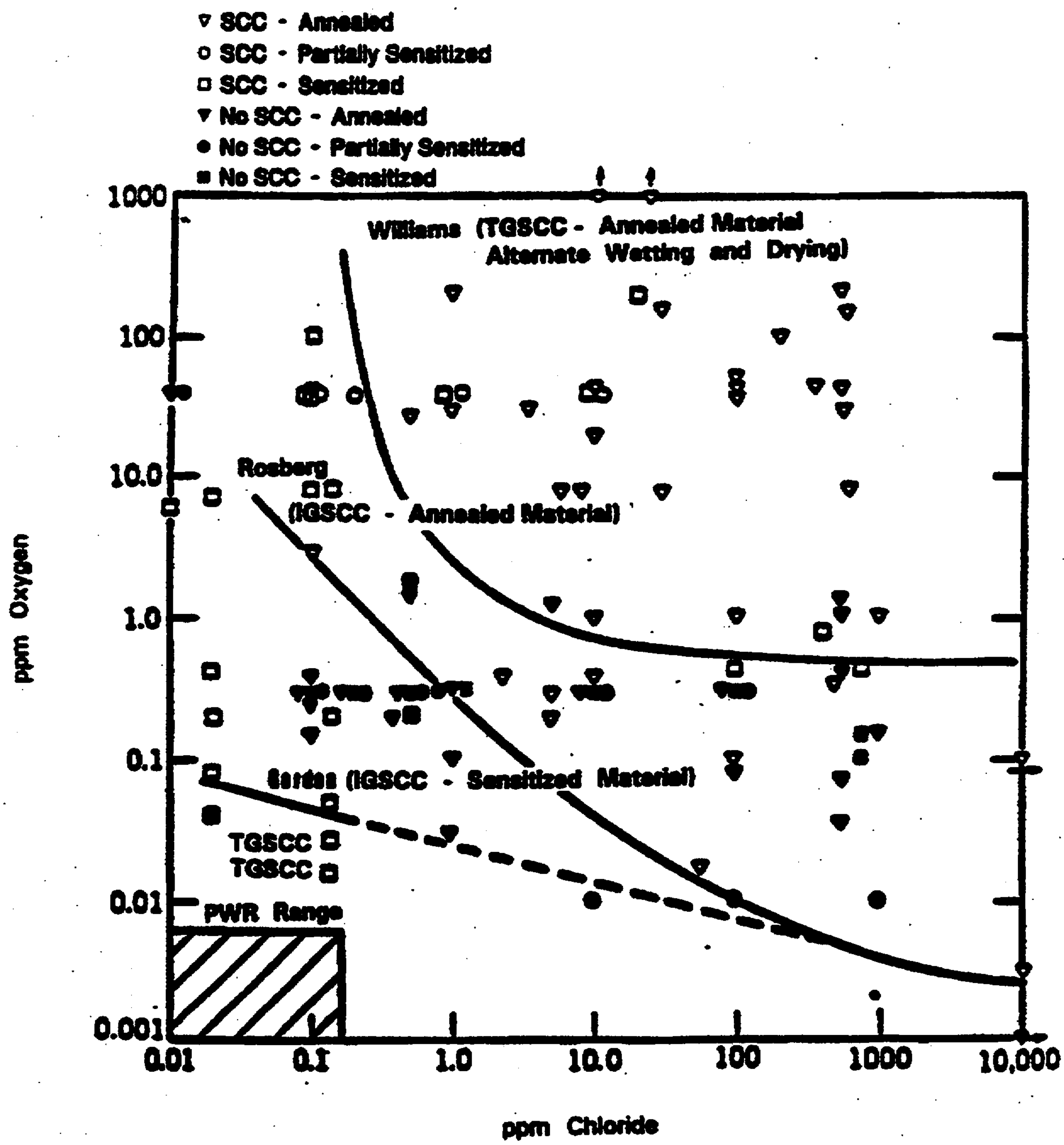


Figure 3-2 The Effect of Oxygen and Chloride on Stress Corrosion Cracking of Austenitic Stainless Steels in High-Temperature Water

The material susceptibility to SCC varies from alloy to alloy as well as with the metallurgical conditions of the material. SCC susceptibility is affected by the chemical composition, preferential orientation of grains, composition and distribution of precipitates, and the grain size of a given material. These factors interact with the environmental conditions and stresses to affect the rate of cracking.

Aging Effect Evaluation

The initiation and propagation of SCC requires a combination of susceptible material, an aggressive environment, and the presence of tensile stresses exceeding the threshold level. Case reports on reactor system support bolt failures due to SCC and excessive applied loads have been documented [Ref. 14]. The key factors include high-strength materials, moist environments, and a high level of sustained tensile stresses. Two general classes of steel have been involved: high-nickel maraging steels, and low-alloy quenched and tempered (LAQT) steels. A common characteristic of RCS support bolting SCC failures was the use of high-strength materials whose specified minimum yield strength exceeds 150 ksi.

Aging Effect Management

The only area of the RCS support system that can potentially be subject to SCC is bolting material with high-strength material. This potential aging effect can be managed by an inspection program as described in Subsection 4.2.2, aging management option AMP-1.3.

3.2.2 Corrosion and Aggressive Chemical Attack

Mechanism Description

Corrosion is an electrochemical reaction between a metal or alloy and its environment and is characterized by a deterioration of the material. The metal becomes thinner by chemical attack at the surface in an aggressive environment. The consequences of such damage are wall thinning, reduction of load-carrying capacity, roughened surface, and build-up of corrosion products.

The extent of corrosion is dependent on the environment and the type of materials used in a specific application. For example, corrosion can easily occur in bare plates, anchor bolts, and brackets due to standing water, whereas austenitic stainless steels resist corrosive attack in the primary coolant of a PWR by quickly oxidizing to form a protective film. Therefore, all internal surfaces of PWR RCS components are fabricated with austenitic stainless steel cladding so they are not susceptible to significant corrosion attack. However, corrosion degradation can occur in areas where cladding is absent or can result from leakage of borated primary coolant onto ferritic or low-alloy steel surfaces.

The primary effect of boric acid leakage is that it can concentrate, leading to corrosion of both carbon and low-alloy steels. Corrosion from water leakage is also a possible mechanism. The general corrosion rate (wastage) of carbon steel and low-alloy steels can be unacceptably high under conditions that prevail when primary coolant leaks onto surfaces and concentrates at operating temperatures. Under drying conditions, corrosion rates as high as 10 to 20 mils per year have been reported [Ref. 15]. Data regarding the corrosion rates of carbon steel and low-alloy steel in highly oxygenated boric acid environments are scattered and appear to be a function of test methodology, temperature, and other contaminants present. Brookhaven National Laboratory has reported corrosion rates of approximately 120 mils per year at 212°F in an unspecified boric acid concentration [Ref. 15].

Aging Effect Evaluation

Corrosion or aggressive chemical attack does not occur to the RCS supports unless there is a leakage event. The leakage of primary coolant and the subsequent evaporation and re-wetting cycles can lead to a concentrated boric acid slurry and the subsequent corrosion of low-alloy and carbon steel components. Note that once boric acid leaking is controlled and the wastage is cleaned, no long-term damage will result. Leakage of demineralizers is also possible; however, it is not as serious as that associated with borated water. Therefore, the effects of general corrosion of RCS support components are not significant, unless the surfaces are exposed to boric acid or demineralized water. Under these conditions, the potential aging effect is possible.

Aging Effect Management

The RCS support system can potentially be subject to corrosion and aggressive chemical attack only following a leakage event. By inspecting for leakage the effects can be identified and managed by subsequent evaluation and corrective action. Refer to Subsection 4.2.1, AMP-1.1 and AMP-1.2.

3.2.3 Neutron Embrittlement

Mechanism Description

Irradiation by neutrons results in embrittlement of ferritic steels. This irradiation embrittlement is manifested by changes in mechanical properties that include increasing tensile and yield strengths with corresponding decreases in fracture toughness and ductility. The extent of irradiation embrittlement is a function of irradiation temperature, neutron fluence, and trace material chemistry. The driving force for irradiation damage is all energy level neutrons that displace metal atoms from their normal lattice positions. Most of these displaced atoms return to normal lattice positions, but a small fraction remain displaced and produce defects in the microstructure and can cause embrittlement.

Most of the research was focused on high-energy neutrons (>1 MeV) and higher temperature conditions. Recently, the issue of embrittlement of ferritic and low-alloy steels at low fluence ($1\text{-}10 \times 10^{17}$ n/cm^2) and low temperatures (122°F) was raised with the publication of Oak Ridge National Laboratory (ORNL) high-flux isotope reactor (HFIR) test data [Ref. 17]. The data showed a higher degree of embrittlement than anticipated for several ferritic steels (American Society for Testing and Materials (ASTM) (A212-B, ASTM A350-LF3, and ASTM A105-II). Transition temperature shifts greater than 100°F were indicated. The data promoted concerns for the potential for neutron embrittlement of ferritic components exposed to low energy (<1 MeV), such as the RPV supports.

Generic Safety Issue (GSI) 15, "Radiation Effects on Reactor Vessel Supports," addresses this issue. The U.S. NRC issued NUREG 1509, "Radiation Effects on Reactor Pressure Vessel Supports" [Ref. 45]. Information contained in this report can be used to assist in the resolution of this issue. The use of this NUREG is discussed in Appendix Section 9.0.

Aging Effect Evaluation

Degradation due to radiation will be addressed by plant-specific evaluation.

Aging Effect Management

This issue cannot be closed at this time. Three options are available for the utility to resolve this issue:

- The utility could commit to incorporating the resolution of GSI-15 at the time of the renewal application,
- A utility may choose to commit to their own program, or
- The utility could reference a generic topical report that demonstrates neutron embrittlement does not cause detrimental aging effects and that there is no need to identify aging management options.

A plant-specific resolution or generic bounding evaluation could consist of the following activities:

- Identify location of support items within the irradiation field
- Define neutron fluence, dpa, and gamma ray dose for each affected support element
- Identify support material type
- Obtain nil-ductility transition temperature (NDTT) shift

- Determine material toughness
- Evaluate material toughness and material behavior under stress levels

3.2.4 Thermal Aging Embrittlement

Mechanism Description

Thermal aging embrittlement of materials is a time/temperature-dependent degradation mechanism that decreases material toughness. Various forms of embrittlement due to thermal aging have been observed. Aging of cast austenitic stainless steels at elevated temperatures (above 600°F), temper embrittlement, and strain aging embrittlement are the most common forms. Temper embrittlement [Ref. 21] and strain aging embrittlement are forms of thermal aging that are seen in ferritic materials. These materials are not commonly used for RCS components. More common to the RCS is the use of cast austenitic stainless steels for piping.

The microstructure of austenitic stainless steel castings is duplex ferrite in an austenite matrix. The ferrite phase increases the tensile strength and improves weldability, stress corrosion resistance, and casting soundness. Studies have shown that the ferrite phase becomes embrittled when the material is aged even at the relatively low temperature of 600°F, resulting in significant decreases in toughness. During service at an elevated temperature for a sufficient time, a brittle alpha prime phase (rich in chromium) can precipitate in the ferrite phase. The precipitation increases the strength and decreases the ductility of high-chromium and duplex stainless steels.

Studies [Refs. 22 to 25] have shown that the presence of a chromium-rich ferrite phase in austenitic stainless steel castings is responsible for thermal aging; the greater the ferrite phase content, the higher the susceptibility to thermal aging. The studies also show that alloying elements such as chromium, molybdenum, nitrogen, carbon, nickel, and silicon all contribute to thermal aging in stainless steel castings.

Aging Effect Evaluation

There is no cast austenitic stainless steel used in the supports that are within the scope of this report. Furthermore, in general, RCS supports are operated at temperatures below 450°F. Therefore, temper embrittlement is not a concern for the ferritic materials of RCS supports. Therefore, thermal aging embrittlement is not applicable.

Aging Effect Management

Due to the lack of detrimental aging effects caused by thermal aging embrittlement, there is no need for the identification of aging management options.

3.2.5 Mechanical Wear

Mechanism Description

Mechanical wear is defined as damage to a solid surface caused by the removal or plastic displacement of material by way of mechanical contact with solid, liquid, or gas. There are three primary types of wear: adhesive wear, abrasive wear, and erosive wear. Adhesive wear is characterized as the transference of material from one surface to another during relative motion or sliding, known as solid-phase welding. Abrasive wear or abrasion is characterized as plastic displacement or loss of material from a solid surface due to hard particles sliding against the surface. Erosive wear is a combined action of abrasion and corrosion. Erosive wear is characterized as an increase in rate of deterioration or attack on a metal because of the relative movement between a corrosive environment and the metal surface.

Aging Effect Evaluation

The RCS component supports are not susceptible to mechanical wear that would cause loss of the RCS intended function. This is because of the wear-resistant material used, the low frequency (number of times) of movement, and the slow movement between sliding surfaces. Note that lubricants are employed in some of the primary component supports, e.g., sliding foot assemblies associated with the RPV. Current inspection programs based on visual examination are employed to identify binding. To date, no binding has been noted.

As discussed in Section 2.6 of the report, maintenance programs follow the ASME Code. The regulations and rules that govern the inspection of primary component and surge line supports begin at the top level with the Code of Federal Regulations. Document 10 CFR 50.55a references Section XI of the American Society of Mechanical Engineers (ASME) Code. Component supports that are subject to examination are examined in accordance with Table IWF-2500-1.

As part of the initial hot functional startup testing, the thermal deflection of the primary loop piping and components is measured at several temperature plateaus from ambient conditions to hot standby. Measurements are taken at lateral support/restraint locations, and between components and piping and the building structure. These measured deflections are compared with the theoretical primary loop movements to ensure that the system is deflecting as it should without binding and to obtain accurate as-built data that will be used to determine shim sizes for the lateral equipment supports and restraints. Once the component support shims and pipe restraint shims have been installed, the support gaps are again monitored during initial heatup to criticality to demonstrate that the support shims were sized properly. Should there be any major construction programs during the life of the plant that could potentially affect the primary loop thermal expansion or support shim sizes (e.g., steam generator replacement), the thermal behavior of the loop is monitored after completion of the construction program to ensure that the loop deflects as it should, and the support shim sizes are adjusted if necessary. From all of the inspections and measurements made, assurance is obtained by the utility that the piping and components are responding properly and binding is not an issue.

Aging Effect Management

Due to the lack of detrimental aging effects caused by mechanical wear, there is no need for the identification of aging management options.

3.2.6 Fatigue

Mechanism Description

Fatigue is a progressive failure of a structural part under repeated, cyclic, or fluctuating loads. Almost all structural materials, ferrous or nonferrous, are subject to fatigue. When a structural part repeatedly experiences fluctuating stresses, damage at microscopic levels may be initiated and accumulated in the material, which eventually leads to cracking. The applied stress could be well below the yield point of the material.

The degree of damage is proportional to the applied stress and number of stress repetitions. However, fatigue will not occur below the endurance limit, the threshold stress level. The endurance limit is a material property that depends on the chemistry, method of manufacturing, heat treatment, etc. The endurance limit of a material is determined from a series of tests on the applied stress (S) versus the number of cycles (N) to failure. The endurance limit for ferrous alloys is defined as the stress value on the horizontal portion (parallel to the log N-axis) of the S-N curve, below which a test specimen will survive for unlimited repetitions, at least 10^8 cycles. For nonferrous alloys, the endurance limit is not well defined. However, in practice, the stress corresponding to a large N-value, say 10^8, in the S-N curve may be used as the endurance limit.

There are two types of fatigue evaluation possible. One is the so-called fatigue usage and the other is fatigue crack growth. Fatigue usage factor, U_f, is used for evaluating structural parts for which no detectable flaws are involved. This evaluation is based on Miner's linear cumulative damage rule, as defined by:

$$U_f = \sum_{j=1}^{p} \frac{n_j}{N_j}$$

where:

j = stress level

p = total number of stress levels considered

n_j = number of cycles fluctuating at level j

N_j = number of cycles or fatigue life corresponding to stress level j, as defined in the S-N curve

A structure is considered unacceptable when U_f is greater than one.

The standardized fatigue usage evaluation method is described in ASME Code Section III. Fatigue crack growth is used for evaluating structural parts for which a crack is found or postulated. This type of analysis is directly related to the linear elastic fracture mechanics (LEFM). The amount of crack growth can be calculated using Paris' law, which correlates crack growth and the driving force by a simple power function, as defined by:

$$\frac{dA}{dN} = C(\Delta K)^n$$

where:

dA/dN = amount of crack growth per cycle

ΔK = range of the applied stress intensity factor

C and n = material constants depending on stress and environmental conditions

Fatigue crack growth analysis is not performed for design or inspection of structures. This type of analysis would be performed only when warranted by special cases where fatigue crack growth would be suspected. For the supports within the scope of this report, this type of analysis is not necessary. Situations where this could occur would be in regions of integral attachments, lugs, nozzles, or welds on a component shell. These areas are not in the scope of this report, but addressed in specific component reports.

Aging Effect Evaluation

Within the RCS components, localized high stress at locations involving geometric discontinuities or heavy sections (for rapid temperature transients) could exist because of a direct interaction with the pressure and thermal loads. For most parts of the RCS supports, however, there is no localized high stress that can be induced by the transients. The stresses on the RCS supports are global reaction forces required to balance the RCS. Fatigue is not an issue for the RCS supports, where the stresses are usually below the endurance limit, and thus only small fatigue usage will result.

An estimate of fatigue for the RCS supports through 40 years of plant operation is detailed in Table 3-2. In this estimate, an enveloping stress of membrane plus bending components, σ_{p+b}, occurring in the RCS supports was used that represents the alternating stress for fatigue evaluation. It is based on upper-limit allowable stresses from ASME Section III, Subsection NF. The number of cycles represent heat-up and cooldown cycles as well as those due to seismic events.

From Table 3-2, the estimated maximum fatigue usage in the RCS supports is less than 0.1 for 40 years of plant operation. For 60 years of operation, the estimated fatigue is less than 0.15. Further, the number of cycles are much less than 20,000 cycles discussed in Section 2.5, recognized by ASME Section III, Subsection NF [Ref. 12] and the American Institute of Steel Construction (AISC) [Ref. 2] as the potential number of cycles where fatigue may need to be considered in design. Therefore, fatigue is not an effect that is a concern for the RCS support structures.

The concrete embedments that are part of the RCS supports are not subject to high stress and cycle combinations. This is supported by Reference 27, which states:

> ...There have been few documented cases of reinforcing fatigue failures in the concrete industry and those published have been produced by high stress/cycle combinations.... Since the safety-related concrete structures are designed for low probability/high consequence loadings, degradation due to fatigue is unlikely....

Aging Effect Management

Due to the lack of detrimental aging effects caused by fatigue, there is no need for the identification of aging management options.

3.2.7 Creep and Stress Relaxation

Mechanism Description

Creep is a continuous physical deformation with time for a metal under a constant applied stress. Just like fatigue, creep may lead to total failure of a structural part. Stress relaxation is similar to creep, but it is referred to as a reduction in stress with time under a given constant strain. Creep and stress relaxation are affected strongly by temperature and depend particularly on the ratio of the test or service temperature, T, to the melting point of a metal or alloy, T_m. Note that the temperature ratio T/T_m mentioned here is evaluated with respect to the absolute temperature scale (i.e., adding 460 to Fahrenheit to convert to Rankin). Thus, for a given T/T_m ratio, it is possible to obtain similar creep curves for iron and lead by properly adjusting the applied stress. Generally, creep is insignificant when the T/T_m ratio is less than 0.5. Some materials possess exceptionally good creep resistance. For example, nickel-based alloys can function satisfactorily in temperatures up to 1800°F without appreciable creep.

Aging Effect Evaluation

Since the temperature in the PWR RCS supports is generally below 650°F (1110°R), well below half of the melting point of steels ($T_m \approx 2410°F = 2870°R$, and $T/T_m \approx 0.39$), creep and stress relaxation are not issues for the RCS supports for extended operation.

Aging Effect Management

Due to the lack of detrimental aging effects caused by creep and stress relaxation, there is no need for the identification of aging management options.

TABLE 3-2
ESTIMATED FATIGUE USAGE AT REACTOR COOLANT SYSTEM SUPPORTS AFTER 40 YEARS

Plant Operating Condition	Stress Level σ_{p+b}	Max Stress Level (ksi)	Number of Cycles (n) Representative	Cycles to Failure[1] (N_f) Representative	Fatigue Usage (n/N_f)
Normal	0.6 Fy_{min}	30	200	25,000	0.008
Upset	Fy_{min}	50	400	5,000	0.08
Sum	-	-	600	-	0.088

__Notes:__

(1) Based on the stress shown in column 3.

3.2.8 Concrete Degradation

Concrete is the most common material used for foundations and large containers, etc., because of its high strength, formability, low cost, and ease of maintenance. Like most structural materials, concrete is susceptible to age-related deterioration during service due to exposure to unfavorable conditions such as weathering, ground water, high temperature, and radiation.

Mechanism Description and Evaluation

The potential degradation mechanisms for concrete structures are discussed in detail in References 28 to 30. The main causes of degradation include cracking and rebar corrosion, aggressive environments, elevated temperature, and radiation exposure. Each of these degradation mechanisms is described in the following subsections.

a. Cracking and Rebar Corrosion

Concrete cracking is common because of the inherently low tensile strength and NDTT. The presence of cracks is an indication of, and cause of, other structural problems. Cracks may form during initial curing or after it has hardened. Common causes of cracking during the initial setting stage are plastic shrinkage and settlement cracks. Any tensile stress may cause concrete to crack. For example, cracks may develop in the concrete containments under internal pressure during leak rate testing.

Cracking is the path to leaking and hostile environments, which in turn become a source for further damage, such as rebar corrosion and concrete leaching. American Concrete Institute (ACI) report 222R-89 [Ref. 31] provides some descriptions for cracking mechanisms. Under normal conditions, the highly alkaline environment of concrete provides a protective film to prevent corrosion of the steel rebar. The presence of cracks promotes the carbonation of concrete, resulting in the reduction of pH and the breaking down of the protective film, and leading to subsequent rebar corrosion. The rust expands and creates tensile stress on the concrete, thus causing spalling and more cracking.

b. Aggressive Environments

Leaching and aggressive chemicals can cause deterioration of concrete. Leaching is a phenomenon that occurs at those parts of concrete where water enters and passes through a concrete body, washing out the readily soluble calcium hydroxide and other solids. As a result, the porosity of the concrete is increased, boosting vulnerability to a hostile environment while reducing strength. The rate of leaching decreases as the amount of dissolved salts contained in the percolating water increases. The rate is higher in cold water due to higher solubility of calcium hydroxide at lower temperatures. Note that leaching is significant only when water flows into cracks or improperly constructed joints. Leaching also results from alternating and successive wetting and drying of concrete. Water flowing on a concrete surface will not cause leaching. The product of leaching, when dried and deposited on a concrete surface, is known as efflorescence. The presence of efflorescence is an indication of leaching.

Because of the alkalinity of hydrated cement paste, acidic material can attack concrete readily. However, note that a minimum concentration must be maintained to have a significant chemical attack. Sulfates of sodium, potassium, or magnesium are harmful chemicals that can attack the hydrated lime and hydrated calcium aluminate in cement paste. The calcium sulfate and calcium sulfoaluminate formation is associated with considerable expansion, which disrupts the concrete.

Based on the Nuclear Management and Resources Council (NUMARC) Class 1 structures industry report [Ref. 44]:

- Concrete exposed for extended periods to aggressive chemicals (<5.5 pH) or chloride or sulfate solutions beyond defined limits (500 ppm chlorides and 1500 ppm sulfates) can undergo significant chemical attack.

- Concrete exposed to above environment for intermittent periods only will not have significant degradation.

For the potential of corrosion to be significant, concrete must be exposed to the corrosive environment for extended periods of time. Cracks must occur to allow aggressive chemicals to reach the reinforcement. The use of ACI 318 and 349 design standards result in dense, well-cured concrete with low permeability with proper reinforcement.

c. Elevated Temperature

The production and handling of steam and the nuclear fission process generate large thermal loads on nuclear plant components. Sustained exposure to high temperatures (300°F or higher) or to numerous hot-cold cycles may cause the concrete to deteriorate, with surface scaling and cracking becoming visible. The key locations include hot process and steam piping penetrations, reactor biological shield, steam-driven equipment pedestals, locations in the turbine building, and certain equipment supports.

Elevated temperature causes concrete to lose the moisture content of its constituents and deteriorates the paste and aggregate. It also creates high local stresses and strains because of the non-homogeneity of the mixture. Elevated temperature will increase the rate of aging, a phenomenon characterized by loss of strength and stiffness with time. Elevated temperature will also reduce concrete's creep resistance, the strength of concrete-reinforcing steel bond, as well as radiation shielding effectiveness.

Concrete operating temperature should not exceed 150°F, and local area temperatures should be kept below 200°F. Reactor vessel supports could be subjected to high temperatures that could potentially result in local temperatures above 200°F if supplemental cooling is not provided. For those support configurations where the local temperature at concrete surfaces could exceed 200°F, special design features are incorporated based on air or water cooling to keep the local temperature below 200°F.

d. Radioactivity

Many studies on the effects of radiation on concrete properties have been conducted [Refs. 32 to 36]. It was generally concluded that damage to the concrete by radioactivity is far less than that by temperature effects. Petrographic examination found neither any recognizable visual degradation of the concrete core nor any cracking in the matrix of aggregate particles for concrete samples exposed to levels similar to those experienced in the most severe locations in a nuclear plant [Ref. 37].

Aging Effect Evaluation

Concrete degradation due to radiation will be addressed by plant-specific evaluation.

Aging Effect Management

See Subsection 3.2.3. This is a plant-specific management issue.

3.2.9 Low Fracture Toughness and Lamellar Tearing

Mechanism Description

During the U.S. NRC licensing review of this mechanism, several questions were raised regarding the potential for lamellar tearing and low fracture toughness of the SG and RCP supports [Ref. 10]. The specific technical concern was the capability of the supports to maintain their structural integrity under accident conditions. To address the low fracture toughness concern, the licensee undertook tests on the excess support steel samples that were not originally defined in the relevant ASTM specifications. The toughness of A 36 steel was found to be adequate, but the toughness of the A 572 steel was determined to be relatively poor at an operating temperature of 80°F or below. To alleviate the toughness concerns, the licensee agreed to raise the temperature of the A 572 steel beams by auxiliary electric heaters to a minimum temperature of 225°F any time the RCS was pressurized above 1000 psig.

Because materials of questionable fracture toughness had been used in other plants, the U.S. NRC issued NUREG-0577 [Ref. 10] in September 1979, revised October 1983 [Ref. 11], to address the concern. This NUREG identifies that a number of plain carbon steel specifications used for support material require no fracture toughness tests. Furthermore, the specifications permit the production of these steels as semi-killed or silicon-killed, which makes a steel inherently coarse-grained and of low fracture toughness. The materials of concern are listed in Table 3-3.

TABLE 3-3
POTENTIAL LOW-TOUGHNESS MATERIALS

Material Specification	Product or Structure Type
ASTM A 285, A 515, A 572	Plate or structural shapes
ASTM A 53, A 105, A 106	Tubular sections
ASTM A 27	Castings
ASTM A 307	Nuts or bolts

NUREG-0577 provides criteria for identifying structural materials, design features, or construction practices that may have resulted in supports that were not in full compliance with the necessary fracture toughness criteria. The criteria address fracture toughness and the minimum specified yield strength. A series of submittals and reviews were conducted to establish the final acceptance criteria.

A review was used to classify plants with respect to the fracture toughness adequacy of the supports. If the components within a plant met the U.S. NRC NUREG-0577 acceptance criteria, then their fracture toughness was considered acceptable. Structural members that exhibited deficiencies were identified. Utilities with potential problems were required to demonstrate that the suspect structures have adequate fracture toughness to comply with the criteria defined by the U.S. NRC in NUREG-0577.

The lamellar tearing issue was described in NUREG-0577. It addresses the failure of SG and RCP support materials that occurred in one utility plant. Lamellar tearing is a failure involving weldments, as characterized by a separation on the base metal that is generally parallel to the weld fusion line. Physically, lamellar tearing is a complex process involving several factors such as heat input, geometry of the joint, steel internal cleanliness, weld size, and post-weld stress relief heat treatment. The problem usually occurs in large welded structures involving a high degree of restraints. Mechanically, lamellar tearing is triggered by the high residual tensile stress produced in the welding process. Large joint sections with a high degree of structural stiffness reduce fracture toughness and enhance crack propagation.

Lamellar tearing can be controlled by careful welding procedures and adequate post-weld stress relief heat treatments. Ultrasonic testing (UT) should be performed to verify the soundness or flawlessness in the welded parts. NUREG-0577 concluded that the likelihood of support failure due to lamellar tearing is low because the applied stresses during plant operation are low.

Aging Effect Evaluation

This mechanism has been identified as an industry issue and addressed. The U.S. NRC provided acceptance criteria (NUREG-0577) for utilities with potential low fracture toughness problems to demonstrate that adequate fracture toughness exists. Further, in NUREG-0577 it was concluded that lamellar tearing was not likely.

Aging Effect Management

Low fracture toughness and lamellar tearing do not cause detrimental aging effects that must be addressed by maintenance programs.

3.3 TIME-LIMITED AGING ANALYSIS EVALUATION

Fatigue is the only mechanism that is considered to be associated with a time-limited aging analysis (TLAA) for the RCS supports. Fatigue was evaluated in Subsection 3.2.6. No fatigue calculations have been performed for the RCS supports as part of their design since the number of cycles is much less than 20,000. An estimate of fatigue usage was made for 60 years of operation for the most critical stress conditions in Subsection 3.2.6. It was found to be less than 0.15 where the allowable limit is 1.0. Therefore, no additional demonstration is required by the utility since the analysis performed in Subsection 3.2.6 serves this purpose. However, a renewal applicant still needs to identify and evaluate plant-specific TLAAs applicable to their supports, if any.

3.4 AGING EVALUATION SUMMARY

Degradation mechanisms causing aging effects that are not likely to be significant and are not a consideration in ongoing inspection and maintenance programs associated with RCS supports are:

- Neutron embrittlement
- Thermal aging embrittlement
- Mechanical wear
- Fatigue
- Creep and stress relaxation
- Low fracture toughness and lamellar tearing

Degradation mechanisms causing aging effects potentially requiring an inspection and maintenance program are:

- Aggressive chemical attack
- Corrosion
- SCC

The program attributes managing these effects are presented in Section 4.0. The attributes are based on existing maintenance and inspection programs. The attributes from the programs will remain adequate to manage aging effects during an extended period of operation since the degradation resulting from aging mechanisms does not increase significantly between inspection periods.

4.0 AGING MANAGEMENT ACTIVITIES AND PROGRAM ATTRIBUTES

In this section, options to manage the effects of aging are presented. Since this report is generically applicable to the plants identified in Section 1.1, only program attributes are given. These attributes are described, and their effectiveness during an extended period of operation is justified. This provides the generic demonstration that aging effects are managed so that intended functions will be maintained consistent with the current licensing basis (CLB) during an extended period of operation. Plant-specific details, based on these attributes, will be developed as part of the license renewal application and complete the demonstration process.

Aging management options are summarized by aging management program (AMP) tables (see Table 4-1). These tables summarize the program attributes and activities that can be implemented by utilities to manage aging affects during an extended period of operation. Details and implementation guidance are provided in the following text. Other options different from the attributes presented in this report can be made and described in the utility plant-specific license renewal application.

The inspection activities being performed and maintenance management programs being pursued to meet current licensing and industry issue requirements should continue. The aging management evaluations reported herein for license renewal have not resulted in any new requirements for utilities. Utilities follow American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) quality experience inspection and examination for their nuclear facility [Ref. 38 and 39]. A utility should provide the basis for implementing options different 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 (Section 2.4).
- The procedures required to address industry issues are implemented in a different manner.

4.1 CURRENT LICENSING BASIS PROGRAMS

Programs have been implemented within the CLB term by utilities to address technical issues resulting from industry practices and United States Nuclear Regulatory Commission (U.S. NRC) directives. Some of these programs are plant-specific aging management programs that also address the aging effects identified in Section 3.0 and satisfy the aging management and program attributes identified in Section 4.0. Such programs are plant-specific and since this report is generic, it is not within the scope and purpose of this report to list these programs for each of the plants. These current term programs would be extended into the license renewal term as required by the Rule. Based on Nuclear Energy Institute (NEI) 95-10, Subsection 4.4.2, the utility will identify these programs and identify any modifications to current term commitments with justification, in the plant-specific license renewal application. Further, if the utility does not follow the recommended program attributes given in the generic technical report

(GTR), the revised program with justification would be provided in the plant-specific license renewal application as well.

Industry issues important to the current term as well as potentially related to the extended period of operation have been identified and discussed in Section 3.1. As deemed necessary by the utility, the current term programs would be extended into the extended period of operation. For example, current utility commitments in response to Generic Safety Issue (GSI) 29 and Inspection and Enforcement Bulletin (IEB) 82-02 related to bolting degradation are adequate to manage aging; therefore, a utility will extend their existing commitments in response to IEB 82-02 into an extended period of operation, as required by 10 CFR 54.33(d), unless modifications are made, in which case a utility would address this in the plant-specific license renewal application. Note that IEB 82-02 does not apply to reactor coolant system (RCS) support bolts but to pressure boundary bolting.

The RCS component and surge line supports are not generally designed to specifically use bolted joint connections requiring preload. Therefore, the support connections designed using bolted joints for the RCS supports and surge line do not rely on preload to remain functional. In the event that preload is important for a specific support design, a locking mechanism can be used to ensure that the preload is not lost. If a support design depends on preload to remain functional, then this would be a plant-specific situation, and if a locking mechanism is not used, a CLB inspection program may include an inspection of the connection for loss of preload if deemed necessary; this would be a plant-specific program.

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.
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 effects may occur and actions, as necessary, to mitigate or eliminate the effect from occurring elsewhere.
Confirmation	Post-maintenance test or other techniques to confirm that the actions have been completed and are effective.

4.2 AGING MANAGEMENT PROGRAMS

Aggressive chemical attack, corrosion, and stress corrosion cracking (SCC) have been evaluated in Section 3.0 and found to have degradation mechanisms causing aging effects potentially requiring inspection and maintenance programs. For the RCS supports, aggressive chemical attack and corrosion have similar degradation effects and are addressed together.

Maintenance programs to manage these potential effects are given. The programs are based on ASME Section XI, 1989 code edition following Examination Category F-A. Examination Category F-A calls for visual (VT-3) inspections following the inspection schedule given in IWB-2000, "Examination and Inspection." For license renewal, inspections should be performed in accordance with Table IWB-2412-1, Inspection Program B, and IWF-2420. These inspections are supplemented by IWF-2430, "Additional Examinations." In addition, supports other than piping supports should receive a 100-percent inspection, and Class 1 piping supports should receive a 25-percent inspection at each interval [Refs. 7 and 8]. Component support conditions observed during the VT-3 inspection that are unacceptable for continued service include: improper hot and cold settings of spring supports; misalignment of supports; deformation or structural degradation of fasteners, springs, clamps, or other support items; and missing, detached, or loosened items. When such conditions are observed, supplemental examinations based on visual (VT-1) inspection or surface (dye penetrant or magnetic particle) or volumetric (radiographic or ultrasonic) examination may be used to determine the mechanism causing the flaw (effect). The extent to which individual utilities reference the mandatory Appendices VII and VIII of ASME Section XI in such augmented inspections will depend on the current inservice inspection (ISI) code of record at the plant. For codes of record prior to the 1989 edition of Section XI, these appendices may not apply. Individual license renewal applicants may reference the appendices in conjunction with a description of the plant code of record. The technical justification for programs that deviate from the 1989 edition of Section XI and Appendices VII and VIII should be provided in a plant's license renewal application.

(a) Ultrasonic examinations may be performed during the period of extended operation if the visual examinations detect surface flaws that exceed the established criteria. These examinations are performed to assist in determining the character of the flaw (size, shape, and orientation). Ultrasonic examinations are part of the supplemental examinations discussed in ASME Section XI, IWF-3200.

(b) The frequency of the periodic examinations using ultrasonic methods is consistent with ASME Section XI Code requirements since the need for such examinations is based on the results of the visual examinations that are performed consistent with ASME Section XI.

(c) The need of ultrasonic examinations is based on findings from visual examinations. The acceptance criteria for visual examinations are given in IWF-3400 for the steel component supports. To determine more information on a potential flaw, ultrasonic examinations may be performed following IWF-3200, as stated below (1989 Code Addenda):

Examinations that detect conditions that require evaluation in accordance with the requirements of IWF-3100 may be supplemented by other examination methods and techniques (IWA-2000) to determine the character of the flaw (that is, size, shape, and orientation). Visual examinations that detect surface flaws that exceed IWF-3400 criteria shall be supplemented by either surface or volumetric examinations.

Ultrasonic examination is a volumetric examination method in IWA-2000.

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

The program attributes are based on requirements that follow ASME Code Section XI and American Concrete Institute (ACI) recommended practices. These inspection practices have been defined by industry using experts knowledgeable in this area. As discussed in Section 2.6, during the current licensing term these practices have been followed by utilities. During the period of extended operation, there will be no change in the plant environment, inspection requirements, loading, design features, or operational procedure that would change the degradation behavior of the structures within the scope of this GTR. Further, the practices as defined by the program attributes have been demonstrated to manage any degradation that would be related to aging, since during the current licensing term industry operating experience there has been no documentation of recorded degradation associated with aging for the supports within the scope of this GTR (see Section 3.1). Further, the attributes retain any special regulatory requirements defined to address technical issues identified during the current licensing term. Such an example is SCC where the attributes retain the guidelines in Electric Power Research Institute (EPRI) report NP-5769, including the exceptions taken by NUREG-1339. Therefore, it can be concluded that the program attributes are based on methods and procedures that have been demonstrated to be capable of managing the aging effects so that the intended functions will be maintained consistent with the CLB during an extended period of operation.

The aging management review that is given in the report is applicable to the passive portion of the snubber support. A review was performed of ASME Section XI, Article IWF-5000. It was concluded that this section pertains primarily to the active portions of the snubber. Reference to the passive elements as described in Sections 2.1 and 2.2 are inspected following VT-3 visual examination methods described in IWA-2213 per Table 4-2, program attribute AMP-1.1. Therefore, IWF-5000 is not added within the attribute tables in the GTR. However, it is noted that the aging management programs to address the potential aging degradation effects do not relieve the utilities from following IWF-5000 requirements for snubbers.

4.2.1 Aggressive Chemical Attack and Corrosion (AMP-1.1 and AMP-1.2)

Those areas exposed to borated or demineralized water are potentially subject to deterioration caused by corrosion. The effects of corrosion could result in reduced load capacity, strength, or loss of movement between sliding surfaces. In this section, the management of these effects is discussed. Specifically:

- Attributes to manage effects as defined in Table 4-1
- Surveillance activities
- Inaccessible area programs
- Visual inspection methods
- Boric acid corrosion management programs
- Demineralized water corrosion management programs

Current ongoing utility plant programs are sufficient to demonstrate and document that these potentially adverse effects (loss of strength, binding) will be evaluated and managed for license renewal. These programs also address potential degradation effects in both steel and concrete structures. The attributes for these programs are defined in the following tables, which are shown at the end of this section:

- Table 4-2 Steel supports (AMP-1.1)
- Table 4-3 Concrete embedments (AMP-1.2)

These attributes have three separate surveillance activities that are linked, as shown in the referenced tables:

- Monitor the general structural conditions of the supports
- Identify leakage
- Monitor leakage

Note that inaccessible areas are subject to age-related degradation effects from these mechanisms. Therefore, the maintenance program should address inaccessible areas. Examples of inaccessible areas within the scope of this generic technical report are:

- Sliding surfaces
- Embedments within concrete
- Water-cooled reactor vessel supports (inside area)
- Reactor pressure vessel (RPV) supports (limited access)

For those designs where the reactor vessel supports are cooled by circulating water through or past bearing plates under the shoe, this is an inaccessible area. If corrosion damage is present, it would likely manifest itself as a leak prior to any significant weakening of the structural members of the supports.

Those areas that are inaccessible are currently excluded from the ASME inspection program. However, utilities must rely on visual examinations for evidence of degradation. Examples of visual evidence of degradation that would be used by a utility to identify potential problems within an inaccessible area are:

- Binding as evidence of local deflections or deformations that are unusual
- Leaking of fluid
- Discoloration or flaking of surface coating, indicating the presence of corrosion

Degradation of inaccessible areas may be managed by:

- Identification of inaccessible areas
- Defining indirect visual evidence that will alert an inspector to potential degradation
- Assessing degradation and need for a more detailed inspection that may take the form of:
  - Use of a remote device (e.g., camera) to assess degradation
  - Use nondestructive examination (NDE) methods to evaluate degradation
  - Use core borings to evaluate concrete degradation
  - Repair degradation if serious
  - Identify mechanism causing degradation effect, evaluate, and correct if feasible

The effects of general corrosion of RCS support component surfaces exposed to leaking primary coolant were found to be potentially significant in Subsection 3.2.2 of this report. These effects may include the following indirect visual evidence: loss of material; discoloration; or accumulated residues on surfaces of components, insulation, or floor areas. Periodic inservice inspections, in accordance with the ASME Code Section XI, Subsection IWF, plus any utility commitments in their CLB in response to Generic Letter 88-05, are capable of managing these effects for both the current and any license renewal term. Examination Category F-A calls for visual (VT-3) inspections that would include visual monitoring of the condition of any lubricant as well as checking for binding.

It is recognized that Generic Letter 88-05 is limited to RCS pressure boundary components. However, it is noted that for corrosion of the RCS support components, evaporation and rewetting cycles from leakage must occur. The source of this leakage would be the RCS pressure boundary components. The boric acid corrosion management programs have been developed in response to Generic Letter 88-05 requirements. Therefore, the commitments made by the utilities to address Generic Letter 88-05 would be part of any aging management program associated with components that may be affected by leakage from pressure boundary components.

Further, as part of a Generic Letter 88-05 management program, a potential path of leakage would be identified with the source. Therefore, the commitments in Generic Letter 88-05 would be credited to assist in the management of the aging effects on the RCS supports since a path of a potential leak to the RCS supports would be included in the utility Generic Letter 88-05 management program, with any corrective actions taken to prevent or control corrosion.

Component support conditions that are unacceptable for continued operation include general corrosion resulting in loss of intended functions of the RCS supports. If the external surfaces are inaccessible, the surrounding area, including floor areas or equipment surfaces located underneath the components, is visually inspected for evidence of leakage. The relevant conditions for visual inspections include (1) area of general corrosion of a component resulting from leakage, and (2) discoloration or accumulated residues on surfaces of components, insulation, or floor areas that may be evidence of borated water leakage.

Boric acid corrosion management programs have been developed in response to Generic Letter 88-05 requirements. Generally, such a program consists of the following:

- Inspect for presence of boric acid crystals via visual examination of the components.
- If boric acid is found, gather information for engineering evaluation and then clean.
- If a significant amount is present, as defined by the presence of crystal buildup, then:
  - Remove crystals and clean surface.
  - Take measurements for evaluation of degradation.
  - If functional integrity limits are exceeded due to corrosive deterioration, make necessary repairs.
  - Identify potential path of leak, identify source (if possible), and fix the leak. If source cannot be found or quickly repaired, it may be necessary to place detectors in critical areas to monitor buildup and take appropriate actions in a timely manner.
- If there are only traces of boric acid, which is defined as light tracks, then only cleaning is required.
- Perform evaluation of degradation and determine if refurbishment is necessary.

Leaks from demineralized water would be addressed in a similar program, as given below:

- Remove water
- Clean surface
- Identify path of leak, identify source (if possible), and fix the leak
- Evaluate degradation taking maintenance or repair actions, as required (e.g., paint surface, repair cracks)

This program can manage the aging effects related to leakage of borated water onto RCS supports because the program provides three diverse methods for detecting the aging effect. When unacceptable effects are detected, corrective actions would be initiated. The three diverse methods for detecting material wastage due to boric acid corrosion are:

- Inspection of supports as specified by ASME Section XI, Subsection IWF
- Identification of leakage during walkdowns
- Leakage monitoring

In terms of risk, aging effects caused by leaking borated water are most likely to occur during operation due to the high-pressure and high-temperature environment, or when the supports have not been inspected for a period of time. Experience has shown that leakage occurs at bolted connections, threaded fittings with seal welds, or branch line connections to piping or components. RCS support components are located away from these locations and the potential harsh environment caused by leakage. During this period, plant-specific leakage monitoring programs would identify that external system leakage was occurring and an appropriate inspection would be initiated.

Before, during, and after refueling outages, the plants also inspect the RCS, including the supports. This inspection would detect leakage too small for the plant's leakage monitoring program and too small to cause a loss of an intended function during the previous operating period of the plant, or when the supports were last inspected. Finally, inspections as specified by ASME Section XI, Subsection IWF would augment the refueling inspections. Evidence of unacceptable aging effects would be corrected as described above. The combination of detection and repair in a timely manner maintains the intended function of the RCS supports. Operating experience has demonstrated the effectiveness of this program (IWF) because no reported damage has occurred to the RCS supports due to leakage.

The program that manages the aging effects due to aggressive chemical attack and corrosion addresses:

- The condition of the supports

- The cause of the effects by identification of leakage causing corrosion and deterioration of the concrete

- The monitoring of the leakage if required

Surveillance or inspection techniques, frequency of inspection, acceptance criteria, corrective actions, and confirmation activities are defined following ASME Code Section XI, 1989 edition, IWF requirements for the steel components, and ACI procedures for the concrete structures. These procedures and methods are recognized as acceptable means to address inspection and maintenance issues by industry and regulatory domains. As an example, Section XI is recognized by the U.S. NRC for defining acceptable inspection and corrective procedures. In SECY-96-080, the U.S. NRC has incorporated subsections IWE and IWL, 1992 edition, into 10 CFR 50.55a "to assure that the critical areas of containments are routinely inspected to detect and take corrective action for defects that could compromise a containment's structural integrity." IWF is similar to IWE and IWL in provision of guidelines for the implementation of ISI, and therefore, the methods given in subsection IWF are adequate to ensure that the critical areas are inspected in a timely manner to detect and take corrective action for aging effects that could compromise intended functions. The inspection frequency is based on recognized industry practice. Leakage walkdowns are recommended at every refueling outage, and if necessary, leakage monitoring can be performed continuously. Corrective actions consist of repairs, replacement, or evaluation in a timely manner with the repairs meeting acceptable standards of ASME Section XI. Preservice examinations of all repairs and replacements are to be made prior to the return to service.

Inaccessible areas are not neglected. The management program recommends acceptable technical procedures using indirect visual evidence of degradation to identify potential aging degradation within these areas. The leakage causing corrosion or aggressive chemical attack is from boric acid or demineralized water. Management programs in response to Generic Letter 88-05, which is related to boric acid corrosion, have been developed and implemented by the utilities. The recommended management program retains these accepted regulatory technical procedures and methods for managing boric acid corrosion degradation. Further, leaks from demineralized water are recommended to be managed by similar means recognizing that the aging effects from demineralized water are less severe. As described above, the aging management program attributes (AMP-1.1 and AMP-1.2) will adequately manage aging during the period of extended operation resulting from potential chemical attack or corrosion.

4.2.2 Stress Corrosion Cracking (AMP-1.3)

The effects of stress corrosion cracking (SCC) have been determined to be potentially significant for structural bolting used in Class 1 component supports in Section 3.2.1 of this report. These effects may include bolting that is cracked or missing as the result of overly hardened or high-strength material operating in a moist environment under sustained tensile stress. Periodic inservice inspections, in accordance with the ASME Code Section XI, Subsection IWF, plus any utility commitments in their CLB in response to IEB 82-02, Unresolved Safety Issue (USI) A-12, or Generic Safety Issue 29 (GSI-29), are capable of managing these effects in the license renewal period. Utility commitments in their CLB to the resolution of USI A-12 or GSI-29 may include hardness testing of support bolting to determine items for augmented inspections.

Table 4-4 (AMP-1.3) is the attribute table describing an acceptable maintenance program to address the potential for SCC. EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," provides guidance for defining conditions for the potential of SCC.

It is noted that in EPRI NP-5769, Volume 1, identifies U.S. NRC generic issue B-29 (GI B-29) as pertaining to degradation and failure of bolting in nuclear power plants. To avoid confusion, this issue is referred to as U.S. NRC GSI-29 in the report.

The aging management program attributes in Section 4 of the report are intended to be implemented after completion of an initial baseline evaluation of the bolts in the RCS supports.

The initial baseline evaluation should follow the guidelines in EPRI report NP-5769 including the exceptions taken by NUREG-1339 and Generic Letter 91-17. Once the baseline evaluation is performed, structural integrity of the bolts' in the RCS supports is thoroughly checked. In other words, the elements that can influence the bolts' susceptibility to SCC are reviewed and satisfied with respect to the guidelines of EPRI report NP-5769.

The SCC baseline evaluation provides justification to eliminate (specific) SCC inservice inspection (ISI) for bolts in the RCS supports. The ASME Section XI requirements are still retained as defined in the other attribute tables. The visual examinations to ASME Section XI in the aging management program attributes are designed to detect conditions of any leakage or other contaminants that may cause degradation of bolts by SCC.

TABLE 4-2
AGING MANAGEMENT PROGRAM ATTRIBUTES PROGRAM AMP-1.1
AGGRESSIVE CHEMICAL ATTACK AND CORROSION (STEEL)

Code References to 1989 ASME Section XI Edition

Attribute	Description	Reactor Coolant Systems Supports Application	
Scope	Components and applicable aging effects.	<u>Component</u> Steel supports Including Embedments (per scope in Section 2.1)	<u>Effect</u> Corrosion due to borated or demineralized water • Reduced load-carrying capacity caused by loss of material • Loss of movement caused by roughened surface or corrosion product build-up
Surveillance Technique	Monitoring, inspection, or testing techniques used to detect the aging effect.	1. Inspect supports per Subsection IWF, Requirements for Class 1, 2, 3, and MC Component Supports of Light-Water Cooled Power Plants (Examination Category F-A, Visual, VT-3) • IWF-2500, Examination Requirements and Table IWF-2500-1, with IWF-2520 OR • IWA-2240, Alternative Examinations 2. Leakage identification walkdowns <u>AND</u> 3. Leakage monitoring • Increase in humidity level • Change in fluid volume • Increase in temperature <u>OR</u> • Increase in radioactivity	
Frequency	Time period between program performance or when a one-time inspection must be completed.	1. Inspection: IWF-2410, Inspection Program - Table IWB-2412-1, each 10-year interval follow first interval, 10-year inspection program, with IWB-2412 2. Leakage walkdown: each refueling outage 3. Leakage monitoring: as needed	

TABLE 4-2 (Continued)
AGING MANAGEMENT PROGRAM ATTRIBUTES PROGRAM AMP-1.1
AGGRESSIVE CHEMICAL ATTACK AND CORROSION (STEEL)

Code References to 1989 ASME Section XI Edition

Attribute	Description	Reactor Coolant System Supports Application
Acceptance Criteria	Qualitative or quantitative criteria that determine when corrective actions are needed.	1. Inspection: IWF-3410, Acceptance Standards - Component Support Structural Integrity 2. Leakage walkdown: identification of fluids 3. Leakage monitoring: plant-specific leakage monitoring criteria • Increase in humidity level • Change in fluid volume • Increase in temperature OR • Increase in radioactivity
Corrective Actions	Actions to further analyze, prevent, or correct the consequences of the effect.	1. Inspection: IWF-3112, acceptance during preservice examinations, with IWF-3200 OR IWF-3122, acceptance during inservice examinations, with IWF-3200 2. Leakage walkdown: remove standing fluid, evaluate boric acid buildup, clean and restore affected surface, and identify source of leak and repair 3. Leakage monitoring: same as 2
Confirmation	Post-maintenance test or other techniques to confirm that the actions have been completed and are effective.	1. Inspection: IWF-2200, preservice examination following adjustment, repair, or replacement prior to return of the system to service IWF-2420, Successive Inspections IWF-2430, Additional Examinations 2. Leakage walkdown: re-examine affected surfaces after cleaning or restoration AND Re-examine at next outage 3. Leakage monitoring: continue monitoring

TABLE 4-3
AGING MANAGEMENT PROGRAM ATTRIBUTES PROGRAM AMP-1.2
AGGRESSIVE CHEMICAL ATTACK AND CORROSION (CONCRETE EMBEDMENT)

Reference to ASME Code Refers to 1989 ASME Section XI Edition

Attribute	Description	Reactor Coolant System Supports Application	
Scope	Components and applicable aging effects	<u>Component</u> Concrete Embedments (per scope in Section 2.1)	<u>Effect</u> Acidic solution — Reduced strength caused by concrete degradation and rebar corrosion Leaching — Reduced strength caused by increased concrete porosity
Surveillance Technique	Monitoring, inspection, or testing techniques used to detect the aging effect	1. Inspect concrete embedments using ACI guidance: ACI 201.1R-68, "Guide for Making a Condition Survey of Concrete in Service" ACI-207.3R-79, "Practices for Evaluation of Concrete in Existing Massive Structures for Service Conditions" ACI 224.1R-89, "Causes, Evaluation, and Repair of Cracks in Concrete Structures" ACI 349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures" 2. Leakage identification walkdowns 3. Leakage monitoring program	
Frequency	Time period between program performance or when a one-time inspection must be completed	1. Inspection: IWF-2410, Inspection Program — Table IWB-2412-1, each 10-year interval follow first interval, 10-year inspection program, with IWB-2412 2. Leakage walkdown: each refueling outage 3. Leakage monitoring: continuous	

TABLE 4-3 (Continued)
AGING MANAGEMENT PROGRAM ATTRIBUTES PROGRAM AMP-1.2
AGGRESSIVE CHEMICAL ATTACK AND CORROSION (CONCRETE EMBEDMENT)

Reference to ASME Code Refers to 1989 ASME Section XI Edition

Attribute	Description	Reactor Coolant System Supports Application
Acceptance Criteria	Qualitative or quantitative criteria that determines when corrective actions are needed	1. Inspection: The following references may be used as a guide for establishing acceptance criteria: ACI 201.2R-77, "Guide to Durable Concrete" ACI 224.1R, "Causes, Evaluation, and Repair of Cracks in Concrete Structures" ACI 224R-89, "Control of Cracking in Concrete Structures" 2. Leakage walkdown: identification of fluids 3. Leakage monitoring: plant-specific leakage monitoring criteria • Increase in humidity level • Change in fluid volume • Increase in temperature <u>OR</u> • Increase in radioactivity
Corrective Actions	Actions to further analyze, prevent, or correct the consequences of the effect	1. Inspection: The following references may be used as a guide for establishing acceptance corrective actions: ACI 201.2R-77, "Guide to Durable Concrete" ACI 207.3R-79, "Practices for Evaluation of Concrete in Existing Massive Structures for Service Conditions" ACI 222R-89, "Corrosion of Metals in Concrete" ACI 224.1R-89, "Causes, Evaluation, and Repair of Cracks in Concrete Structures" ACI 224R-89, "Control of Cracking in Concrete Structures" <u>With</u> IWF-3112, acceptance during preservice examinations <u>OR</u> IWF-3122, acceptance during inservice examinations 2. Leakage walkdown: remove standing fluid, clean and restore affected surface, and identify source of leak and repair 3. Leakage monitoring: same as 2

TABLE 4-3 (Continued)
AGING MANAGEMENT PROGRAM ATTRIBUTES PROGRAM AMP-1.2
AGGRESSIVE CHEMICAL ATTACK AND CORROSION (CONCRETE EMBEDMENT)

Reference to ASME Code Refers to 1989 ASME Section XI Edition

Attribute	Description	Reactor Coolant System Supports Application
Confirmation	Post-maintenance test or other techniques to confirm that the actions have been completed and are effective	1. Inspection: IWF-2200, preservice examination following adjustment, repair, or replacement prior to return of the system to service IWF-2420, Successive Inspections - intervals IWF-2430, Additional Examinations 2. Leakage walkdown: re-examine affected surfaces after cleaning or restoration <u>AND</u> Re-examine at next outage 3. Leakage monitoring: continue monitoring

TABLE 4-4
AGING MANAGEMENT PROGRAM ATTRIBUTES PROGRAM AMP-1.3
STRESS CORROSION CRACKING (BOLTING)

Code References to 1989 ASME Section XI Edition

Attribute	Description	Reactor Coolant System Supports Application	
Scope	Components and applicable aging effects	<u>Component</u> RCS Support Bolting (per scope per Section 2.1) Conditions (per EPRI guideline, NP-5769) <u>AND</u> • Bolt or stud sizes > 1-in. nominal diameter	<u>Effect</u> Stress Corrosion Cracking • Crack initiation • Localized cracking failure
Surveillance Technique	Monitoring, inspection, or testing techniques used to detect the effect	Inspect bolting per ASME Section XI, Subsection IWF • IWF-2500, Examination Requirements and Table IWF-2500-1, with IWF-2520 <u>OR</u> • IWA-2240, Alternative Examinations	
Frequency	Time period between program performance or when a one-time inspection must be completed	IWF-2410, Inspection Program — Table IWB-2412-1, each 10-year interval, follow first interval, 10-year inspection program, with IWB-2412	
Acceptance Criteria	Qualitative or quantitative criteria that determine when preventive or corrective actions are needed	IWF-3410, IWF-3200, AND IWA-2000, criteria based on VT-1 and VT-3 visual examinations	
Corrective Actions	Actions to mitigate or reverse the consequences of the effect	1. Evaluate existing materials and design, modify susceptible materials or design <u>OR</u> 2. Replace defective bolts	
Confirmation	Post-maintenance test or other techniques to confirm that the actions have been completed and are effective	Re-examine replaced bolts at next inspection interval if still susceptible to SCC	

5.0 SUMMARY AND CONCLUSIONS

The reactor coolant system (RCS) supports, associated with the plants listed in Table 1-1, have been reviewed for aging management as part of the Westinghouse Owners Group (WOG) Life Cycle Management/License Renewal (LCM/LR) program, with financial support from the WOG and Electric Power Research Institute (EPRI). The RCS supports are subject to an aging management review because they maintain system intended functions, support these intended functions in a passive manner, and are long-lived. This aging management review has identified aging effects caused by degradation mechanisms and evaluated the aging effects to determine which require management during an extended period of operation. For those effects that require management, options have been provided. No further demonstration is required by the utility related to time-limited aging analyses (TLAAs). All necessary demonstration requirements are contained in this report. However, a renewal applicant still needs to identify and evaluate plant-specific TLAAs applicable to their RCS supports, if any.

5.1 SUMMARY

The RCS supports maintain the system intended functions of:

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

The intended function of the RCS supports is to maintain the RCS components in equilibrium within spatial positions prescribed by design. This ensures the structural integrity and safe operation of the RCS piping and primary components under design conditions (the system intended function, as defined in 10 CFR 54).

The scope of the report covers age-related degradation issues associated with the support systems of the major components of the RCS, including the reactor pressure vessel (RPV), the steam generator (SG), the reactor coolant pump (RCP), the pressurizer (PZR), as well as the PZR surge line supports.

The mechanisms identified from review of design limits and aging are:

- Stress corrosion cracking (SCC)
- Corrosion and aggressive chemical attack
- Neutron embrittlement
- Thermal aging embrittlement

- Mechanical wear
- Fatigue
- Creep and stress relaxation
- Concrete degradation
- Low fracture toughness and lamellar tearing

Mechanisms that can cause aging effects are identified in Section 3.0.

The aging effects of these mechanisms have been evaluated, and the following require management during an extended period of operation:

- Aggressive chemical attack
- Corrosion
- SCC

Options to manage these aging effects have been provided in AMP-1.1, AMP-1.2, and AMP-1.3. The aging effects of the mechanisms listed above can be managed by current industry programs. All options are described in Section 4.0. These aging management programs are representative of current practices and have been generically demonstrated to be acceptable for an extended period of operation. They are based on ASME Code Section XI, 1989 edition. A utility may follow a different program; however, a utility is then required to provide a description of their program as part of their plant-specific license renewal application. Also, a utility, as deemed necessary, would continue into the extended period of operation established programs that address industry issues in the plant's current term.

In an appendix, Section 8.0, guidance is provided for the applicant with respect to the use and application of this generic technical report (GTR).

Listed below are those items that the applicants are to address in their application as defined within this GTR:

- Identification and evaluation of any plant-specific TLAAs applicable to their RCS supports (see Section 3.3).

- Identification and evaluation of current-term programs implemented within the current licensing basis term to address technical issues from industry practices and United States Nuclear Regulatory Commission (U.S. NRC) directives should be continued into the license renewal term. Modifications to or elimination of these programs have to be justified (see Section 4.1).

- Identification and justification of plant-specific programs that deviate from the recommended aging management programs (see Section 4.2).

- Technical justification for programs that deviate from the 1989 edition of Section XI and Appendices VII and VIII should be provided in a plant's license renewal application (see Section 4.2).

- Identification of any specific program necessary to ensure that proper preload is retained for the component supports within the scope of this report (see Section 4.1).

- Identification of any evidence of aging degradation in inaccessible areas during the current licensing term that is considered to potentially affect system intended functions. A plan of action to address any identified potential degradation should be provided (see Section 4.2 and RAI #11 of Section 7.2).

- Verification that the plant is bounded by the GTR. The actions applicants must take to verify their plant is bounded will be described in a generic implementation procedure (see RAI #24 of Section 7.2).

- Plant-specific evaluation of potential degradation due to irradiation of the components within the scope of this report (see Subsections 3.2.3 and 3.2.8(d)).

5.2 CONCLUSIONS

Implementation of the aging management options will manage the identified aging effects. Therefore, it is concluded that the RCS intended functions maintained by the RCS and surgeline supports can be ensured during the extended period of operation for the plants identified in Table 1-1, in accordance with the current licensing basis. System-level intended functions supported by the RCS supports will also be maintained.

6.0 REFERENCES

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7.0 APPENDICES

7.1 U.S. NUCLEAR REGULATORY COMMISSION REQUEST FOR ADDITIONAL INFORMATION - WCAP-14422, REV. 1

This appendix provides the questions and responses associated with the United States Nuclear Regulatory Commission (U.S. NRC) request for additional information (RAI) related to the Westinghouse Owners Group Topical Report WCAP-14422, Rev. 1, March 1996. Also provided is a table correlating the RAI with the section in the report that was changed due to the RAI. Note that section numbers given in the RAIs refer to section and page numbers in Revision 1 of the WCAP-14422.

REQUEST FOR ADDITIONAL INFORMATION
WESTINGHOUSE OWNERS GROUP TOPICAL REPORT WCAP-14422, Rev. 1
"LICENSE RENEWAL EVALUATION: AGING MANAGEMENT FOR REACTOR COOLANT SYSTEM SUPPORTS"

RAI and Report Section Cross-Reference Table

RAI NUMBER	DESCRIPTION	REPORT SECTION
1	WOG report guidance	8.0
2	Program attribute technical basis	4.2, 4.2.1, 4.2.2
3	Generic Letter 88-05 commitments	4.2.1
4	Current term programs	4.1, 5
5	ASME Sections VII and VIII	4.2, 5
6	Code Case N-491	--
7	Bolt preload	4.1, 5
8	Generically bounding fatigue analysis	--
9	SCC & ASME Section XI	Table 4-4
10	EPRI NP-5769 & NUREG-1339	4.2.2
11	Inspection of inaccessible areas	5
12	Snubber supports	2.1, 2.2, 4.2
13	Embedments	2.3
14	SCC surveillance	4.2.2
15	Support anchorage system	--
16	Ultrasonic examinations	4.2 Tables 2-8, 4-2 and 4-4
17	Wear-resistant material	2.3
18	Binding	3.2.5
19	Scope, base plates, embedments, and anchor bolts	2.1, Table 4-2
20	Leakage walkdowns and monitoring	4.2.1
21	Concrete embedments	2.3, 2.4.1
22	Attribute action description	Table 4-1
23	ACI 349.3R-96	2.6, Table 4-3
24	Judged bounds	5
25	General maintenance practices	--
26	Aging mechanism/effect evaluation	2.6.5 (new)
27	Temper embrittlement	3.2.4
28	Fatigue crack growth analysis	3.2.6

RAI and Report Section Cross-Reference Table - Continued

RAI NUMBER	DESCRIPTION	REPORT SECTION
29	Radiation concrete degradation	3.2.8
30	Demonstration/technical basis	4.0
31	Omitted inspection/procedures	--
32	Fatigue	--

RAI and Report Section Cross-Reference Table - Continued

RAI NUMBER	DESCRIPTION	REPORT SECTION
EDITORIAL COMMENTS		
1	Table 2-14 change to Table 2-7	2.4.6
2	Surry design code	Table 2-7
3	Specific standards	2.6.1
4	Appendix A of ASME Code	3.2.6
5	NUREG-0577	3.2.9
6	AISC Code	--
7	ACI 349.3R-95	6.0
8	"Modes"	2.1
9	Inspection	2.6.1
10	TLAA - plant-specific	3.3, 5.0
11	"dry" to "drying"	3.2.2
12	Piping support inspection	4.2
13	10 CFR 54.33(d)	4.1
Supplementary Editorial Comments from NRC Meetings		
1	Embedment scope	Table 4-2
2	EPRI NP-5769	Table 4-4

Request for Additional Information

1. Provide guidance for the use of the WOG report by reference in a renewal application. For example, discuss how an applicant can justify that its plant is bounded by the assumptions in the report, or determine if a plant specific program includes the attributes described in the report.

Response

A WOG GTR presenting an aging management evaluation that has been reviewed and approved by the U.S. NRC could be directly referenced in a renewal application. In this way, the applicant claims the same approval that was given by the U.S. NRC to the generic technical report, that is, that the aging management program attributes described in the report adequately manage the identified aging effects for the subject structure or component.

To claim that the report is applicable, the applicant must first evaluate that the plant-specific structure or component has the same characteristics as, or is bounded by, the structure or component included in the GTR. This includes configuration, functions, materials, service conditions, and design parameters. In addition, the applicant must also evaluate that the plant-specific component includes any protective measures assumed in the GTR (coatings, cathodic protection, etc.).

The applicant must then evaluate that the assumptions and bases for determining the aging effects identified in the report are applicable to the plant-specific structure or component being evaluated. This will require some review of the plant's operating and maintenance history to confirm that all aging effects apply. Any differences should be justified.

The specific plant programs that correspond to the aging management program(s) (AMP) should be identified. Individual features of the plant-specific programs should be compared with the attributes of the GTR aging management programs to ensure that the aspects of the plant-specific program (scope, frequency, activities, verifications, etc.) meet the requirements of the applicable GTR aging management program. Any differences should be identified and evaluated, so that the conclusions regarding the aging management program(s) still apply.

Finally, the applicant should provide some objective evidence that the plant-specific program has demonstrated effectiveness in managing the identified aging effects. This evidence can be obtained from plant operating and maintenance records.

If the plant-specific program is a "new" program, then the objective evidence can be inferred from an analogous existing program to demonstrate that the aging effect can be managed. Where there is no analogous program, demonstration of the effectiveness of the new or revised program requires two separate efforts. The first involves describing the new program with its attributes. The second effort involves providing objective evidence of the effectiveness at some future time.

Response to RAI #1 (Cont.)

The evaluations described above will reside in plant documents, with appropriate material extracted and summarized in the Application FSAR Supplements.

A new Section 8 will be added to this GTR to reflect the above process providing guidance to the utility.

Request for Additional Information

2. In response to RAIs #2 and #26, the WOG cites the revised Section 4.0 as the section of the report that includes a description and justification of aging management programs. Although Section 4.0 provides additional information on aging management programs in the form of "aging management program attributes," the WOG has not provided any justification for the program attributes. The aging management program attributes in Section 4 of the report must be supported by a technical basis to justify why these specific attributes will adequately manage aging during the period of extended operation. Provide the technical basis.

   For example, Program AMP-1.3 on stress corrosion cracking of bolting consists of inspection attributes which are visual inspections "VT-3" per Table IWF-2500-1 of Section XI of the ASME Code. Describe the technical basis for why a "VT-3" inspection is capable of detecting stress corrosion in the bolting configuration in a timely manner (i.e., prior to a loss of intended function).

Response

The following will be added to the report to address comment. See also RAI #20.

Added at the end of Section 4.2, Aging Management Programs:

> The program attributes are based on requirements that follow ASME Code Section XI and American Concrete Institute recommended practices. These inspection practices have been defined by industry using experts knowledgeable in these areas. As discussed in Section 2.6, during the current licensing term these practices have been followed by utilities. During the extended period of operation, there will be no change in the plant environment, inspection requirements, loading, design features, or operational procedure that would change the degradation behavior of the structures within the scope of this GTR. Further, the practices as defined by the program attributes have been demonstrated to manage any degradation that would be related to aging, since during the current licensing term industry operating experience there has been no

Response to RAI #2 (Cont.)

documentation of recorded degradation associated with aging for the supports within the scope of this GTR (see Section 3.1). Further, the attributes retain any special regulatory requirements defined to address technical issues identified during the current licensing term. Such an example is stress corrosion cracking where the attributes retain the guidelines in EPRI report NP-5769, including the exceptions taken by NUREG-1339. Therefore, it can be concluded that the program attributes are based on methods and procedures that have been demonstrated to be capable of managing the aging effects so that the intended functions will be maintained consistent with the CLB during an extended period of operation.

In addition to the above, the following has been added to the GTR at the end of Subsection 4.2.1, Aggressive Chemical Attack and Corrosion (AMP-1.1 and AMP-1.2):

The program that manages the aging effects due to aggressive chemical attack and corrosion addresses:

- The condition of the supports
- The cause of the effects by identification of leakage causing corrosion and deterioration of the concrete
- The monitoring of the leakage if required

Surveillance or inspection techniques, frequency of inspection, acceptance criteria, corrective actions, and confirmation activities are defined following ASME Code Section XI, 1989 edition, IWF, requirements for the steel components, and ACI procedures for the concrete structures. These procedures and methods are recognized as acceptable means to address inspection and maintenance issues by industry and regulatory domains. As an example, Section XI is recognized by the U.S. NRC for defining acceptable inspection and corrective procedures. In SECY-96-080 the U.S. NRC has incorporated subsections IWE and IWL, 1992 edition, into 10 CFR 50.55a "to assure that the critical areas of containments are routinely inspected to detect and take corrective action for defects that could compromise a containment's structural integrity." IWF is similar to IWE and IWL in provision of guidelines for the implementation of ISI, and therefore, the methods given in subsection IWF are adequate to ensure that the critical areas are inspected in a timely manner to detect and take corrective action for aging effects that could compromise intended functions. The inspection frequency is based on recognized industry practice. Leakage walkdowns are recommended at every refueling outage, and if necessary, leakage monitoring can be performed continuously.

Response to RAI #2 (Cont.)

Corrective actions consist of repairs, replacement, or evaluation in a timely manner with the repairs meeting acceptance standards of ASME Section XI. Preservice examinations of all repairs and replacements are to be made prior to the return to service.

Inaccessible areas are not neglected. The management program recommends acceptable technical procedures using indirect visual evidence of degradation to identify potential aging degradation within these areas. The leakage causing corrosion or aggressive chemical attack are from boric acid or demineralized water. Management programs in response to Generic Letter 88-05, which is related to boric acid corrosion, have been developed and implemented by the utilities. The recommended management program retains these accepted regulatory technical procedures and methods for managing boric acid corrosion degradation. Further, leaks from demineralized water are recommended to be managed by similar means recognizing that the aging effects from demineralized water are less severe. As described above, the aging management program attributes (AMP-1.1 and AMP-1.2) adequately manage aging resulting from potential chemical attack or corrosion during the period of extended operation.

Added at the end of Subsection 4.2.2, Stress Corrosion Cracking (AMP-1.3)

The aging management program attributes in Section 4 of the report are intended to be implemented after completion of an initial baseline evaluation of the bolts in the RCS supports.

The initial baseline evaluation should follow the guidelines in EPRI report NP-5769 including the exceptions taken by NUREG-1339 and Generic Letter 91-17. Once the baseline evaluation is performed, structural integrity of the bolts in the RCS supports is thoroughly checked. In other words, the elements that can influence the bolts susceptibility to SCC are reviewed and satisfied with respect to the guidelines of EPRI report NP-5769.

The SCC baseline evaluation provides justification to eliminate (specific) SCC ISI for bolts in the RCS supports. The ASME Section XI requirements are still retained as defined in the other attribute tables. The visual examinations to ASME Section XI in the aging management program attributes are designed to detect conditions of any leakage or other contaminants that may cause degradation of bolts by SCC.

Request for Additional Information

3. In response to RAIs #3, #28, and #29, the WOG indicates that CLB commitments in response to Generic Letter 88-05 would be capable of assisting in the management of general corrosion of the RCS support components. However, Generic Letter 88-05 is limited to RCS pressure boundary components. In order to credit the commitments in Generic Letter 88-05 to assist in the management of the effects aging on the RCS supports, verify that those commitments include the RCS supports, such as tracing RCS leakage to the RCS supports.

Response

Consistent with the description provided in the response to RAI # 1, the licensee would verify that their Generic Letter 88-05 program includes leakage on RCS supports. The response given below will be added to Subsection 4.2.1.

> It is recognized that Generic Letter 88-05 is limited to RCS pressure boundary components. However, it is noted that for corrosion of the RCS support components, evaporation and rewetting cycles from leakage must occur. The source of this leakage would be the RCS pressure boundary components. The boric acid corrosion management programs have been developed in response to Generic Letter 88-05 requirements. Therefore, the commitments made by the utilities to address Generic Letter 88-05 would be part of any aging management program associated with components that may be affected by leakage from pressure boundary components.
>
> Further, as part of a Generic Letter 88-05 management program, a potential path of leakage would be identified with the source. Therefore, the commitments in Generic Letter 88-05 would be credited to assist in the management of the aging effects on the RCS supports since a path of a potential leak to the RCS supports would be included in the utility Generic Letter 88-05 management program, with any corrective actions taken to prevent or control corrosion.

Request for Additional Information

4. In response to RAI #5, the WOG indicates that programs resulting from response to generic communications are followed by utilities during the current term and as necessary into renewal. Page 57 (Section 4.1) of the report further states, "As deemed necessary by the utility, the current term programs would be extended into the extended period of operation." To achieve closure on these issues, the WOG should determine which program(s) is relied on to manage the applicable aging effects for the RCS Supports. If the program is determined to be necessary to manage the effects of aging during the extended period of operation, then it should be clearly identified and justified in the report.

Response to RAI #4 (Cont.)

Response

Programs have been implemented within the current licensing basis term by utilities to address technical issues resulting from industry practices and U.S. NRC directives. Some of these programs are plant-specific aging management programs that also address the aging effects identified in Section 3 and satisfy the aging management and program attributes identified in Section 4. Such programs are plant-specific, and since this report is generic, it is not within the scope and purpose of the generic reports to list these programs for each of the plants. These current-term programs would be extended into the license renewal term as required by the Rule. Based on NEI 95-10, Section 4.4.2, the utility will identify these programs and identify any modifications to current-term commitments with justification in the plant-specific license renewal application. Further, if the utility does not follow the recommended program attributes given in the GTR, the revised program with justification would be provided in the plant-specific license renewal application as well.

This discussion has been incorporated into Section 4.1 and Section 5.

Request for Additional Information

5. In response to RAI #8, the WOG indicates that the extent a utility would reference Appendices VII and VIII to Section XI depends on the plant's code of record. However, to provide assurance that inservice inspections performed during the period of extended operation would be reasonably based on the 1989 edition of Section XI and Appendices VII and VIII. A utility may propose a different program, however, the technical justification for that program must be provided in the renewal application.

Response

The WOG is in agreement, and the following sentence has been added to the end of the second paragraph of Section 4.2 and Section 5: "The technical justification for programs that deviate from the 1989 edition of Section XI and Appendices VII and VIII should be provided in a plant's license renewal application."

Request for Additional Information

6. In response to RAI #19, the WOG indicates that the RCS support inspection is based on NRC-endorsed ASME Code Case N-491 (March 14, 1991) which eventually became the 1992 edition of ASME XI, Subsection IWF. However, in response to RAI #7, the WOG indicates that the RCS support program is based on the 1989 edition of ASME

Response to RAI #6 (Cont.)

Section XI. Clarify whether the WOG is relying on the 1989 edition of ASME Section XI, ASME Code Case N-491, or both to manage the effects of aging on the RCS Supports. If Code Case N-491 is being relied on, please identify the differences between this code case and the 1989 edition of ASME Section XI, and identify which components and aging effects Code Case N-491 will be used to manage.

Response

The RCS GTR license renewal evaluation support program is based on the 1989 edition of ASME Section XI. However, there are cases when it is necessary, for clarification purposes, to supplement the ASME Section XI requirements. The case in question, the only one in this GTR, is such a situation. It is stated in Section 4.2:... "In addition, supports other than piping should receive a 100-percent inspection, and Class 1 piping supports should receive a

25-percent inspection at each interval ..." It was recognized that the 25-percent criterion is not in the 1989 edition of ASME Section XI, and therefore, Code Case N-491 (Reference 8 of WCAP-14422, Rev. 1) and Regulatory Guide 1.147 (Reference 7 of WCAP-14422, Rev. 1) have appropriate guidance and were provided as references for the technical justification. The 25-percent criterion is given in Table-2500-1 of the cited code case. Further, the U.S. NRC has reviewed Code Case N-491 (3/14/91) and has documented it to be acceptable for application in the inservice inspection of components and their supports for water-cooled nuclear power plants in Regulatory Guide 1.147, Revision 11, October 1994.

Request for Additional Information

7. In response to RAI # 27, the WOG indicates the visual (VT-3) examinations of supports can detect the "loss of preload sufficient to cause bolts or fasteners to become loose." However, the response did not discuss how loss of preload prior to bolts becoming loose will be detected such that the intended function of the bolted joints will be maintained under CLB design loads (e.g., seismic events or LOCA). Please discuss how loss of preload prior to bolts becoming loose will be detected and managed. (This comment also applies to RAI # 39.)

Response

The RCS component and surge line supports are not generally designed to specifically use bolted joint connections requiring preload. Therefore, the support connections designed using bolted joints for the RCS supports and surge line do not rely on preload to remain functional. In the event that preload is important for a specific support design, a locking mechanism can be used to assure that the preload has not been lost. If a support design depends on preload to remain functional, then this would be a plant-specific situation, and if a locking mechanism is

Response to RAI #7 (Cont.)

not used, a CLB inspection program may include an inspection of the connection for loss of preload if deemed necessary; this would be a plant-specific program.

This discussion has been added in Section 4.1. In Section 5 it is identified as one of the items that the applicant must address in their application if they have a plant-specific program necessary to assure that they retain proper preload for the component supports within the scope of this GTR.

Request for Additional Information

8. In response to RAI #35, the WOG should provide additional information on the fatigue cumulative usage factor evaluation result presented in the report. For example, if appropriate, the WOG could indicate that the fatigue analysis is generically bounding

because it is performed using conservatively bounding stresses assuming bounding cycles based on reviewing WOG plant transients, etc.

Response

The fatigue analysis reported in Section 3.2.6 of the report is generically bounding. It is stated in the second paragraph under "Aging Effect Evaluation":

> An estimate of fatigue for the RCS supports ... is detailed in Table 3-2. In this estimate, an enveloping stress of membrane plus bending components, ..., occurring in the RCS supports was used that represents the alternating stress for fatigue evaluation. It is based on upper-limit allowable stresses from ASME Section III, Subsection NF.

As noted in this paragraph, the stress is based on an upper-limit allowable stress. This stress is generically bounding since it is based on the highest allowable stress that a component support could experience (based on ASME Section III, Subsection NF, but also representative of earlier plant vintages that use the AISC code). Further, in the second paragraph it is stated, "The number of cycles represent heat up and cool down cycles as well as those due to seismic event." These are bounding cycles for the Westinghouse plants.

Request for Additional Information

9. In response to RAI #37, the WOG indicates that the current Section XI program is adequate for managing stress corrosion cracking (SCC) of all RCS support bolting for renewal. In addition, the response indicates that for bolting other than the steam generator supports and their anchor bolting, reactor coolant pump support anchor bolting, pressurizer support skirt anchor bolting, and pressurizer support bolting, the SCC management program should be based on EPRI NP-5769 "Degradation and Failure of Bolting in Nuclear Power Plants."

   The above program characterization appears to be inconsistent with the program attributes in Table 4-4, Program AMP-1.3, of the WOG report. Table 4-4 indicates that steam generator support bolting, reactor coolant pump bolting, and pressurizer support and skirt bolting that meet certain conditions specified in the EPRI guideline and that are greater than 1 inch nominal diameter will be managed for SCC, during the period of extended operation, using the current Section XI program.

   Clarify the WOG program for managing SCC of all RCS support bolting. Is it the WOG intent to use the ASME Section XI program for the RCS support bolts that meet the three conditions in Table 4-4, or for all the RCS support bolting.

Response

It is the intent of the WOG program to use the ASME Section XI program to manage stress corrosion cracking of only the RCS support bolts identified in Table 4-4.

Request for Additional Information

10. In response to RAIs #37, #40, #41, and #42, the WOG references EPRI NP-5769. This EPRI report is discussed in Generic Letter 91-17 and NUREG-1339. Verify that the application of this EPRI report is consistent with discussions in Generic Letter 91-17 and NUREG-1339.

Response

Recommendation of the baseline evaluation of the RCS support bolts to EPRI NP-5769 includes exceptions taken by NUREG-1339 and Generic Letter 91-17. This has been added in Subsection 4.2.2. See also RAI #2.

Request for Additional Information

11. In response to RAI #13, the WOG discusses indirect inspection of inaccessible areas, that is, the inspection of surrounding areas to infer the condition of the inaccessible areas. However, there are major RCS supports which have not been inspected during the current license term because of inaccessibility, such as the reactor vessel supports. Discuss the WOG plans for a direct inspection of inaccessible supports (including inspection areas, examination technique, frequency, and acceptance criteria), at least on a one-time basis, to directly assess the condition of the inaccessible supports for renewal.

Response

The need for a direct inspection of inaccessible supports was assessed for the supports within the scope of this GTR, and it was concluded that they are necessary only if there are indications of degradation as defined in Section 4.2. Inspections of inaccessible areas are not necessary in load bearing areas since:

- No significant aging effect has occurred or is expected within the areas of inaccessible load bearing portions of the supports.

- Potential degradation due to wear is not considered a significant aging mechanism because of the wear-resistant material used, and any potential binding is not significant (see response to RAIs #17 and #18 given below).

Further, the inspection program given in the GTR for inaccessible areas is adequate to manage the potential aging degradation identified for these supports (see previous RAI #13 response given in Section 7.1 of the report). If a utility has evidence of aging degradation in inaccessible areas during the current licensing term which they may deem as potentially affecting system intended function, then the utility should so identify this situation in their plant-specific application. This is a plant-specific item that may result in a need for a one-time direct inspection of an inaccessible area prior to the extended licensing term. If such an inspection is needed, it should be performed in a manner that will not compromise the structural integrity of the supports and structures. In Section 5, this has been added in the list of potential plant-specific issues that must be addressed in the applicant's submittal.

Request for Additional Information

12. Page 7 (first paragraph) states, "Since the RCS supports (excluding snubbers) perform the intended function in a passive manner and are long-live, they are subject to an aging management review."

    Clarify whether the scope of this report include snubber supports, if not explain why, otherwise provide a discussion on aging management review of the RCS supports to which snubbers are attached.

Response

The passive support components associated with the snubber supports are included in the scope. This report will be modified to clarify this (Sections 2.1 and 2.2). The aging management review that is given in the report is applicable to the passive portion of the snubber support.

A review was performed of ASME Section XI, Article IWF-5000. It was concluded that this section pertains primarily to the active portions of the snubber. Reference to the passive elements as described in Section 2.1 and 2.2 are inspected following VT-3 visual examination methods described in IWA-2213 per Table 4-2, program attribute AMP-1.1. Therefore, IWF-5000 is not added within the attribute tables in the GTR. However, in Section 4.2 it is noted that the aging management programs to address the potential aging degradation effects do not relieve utilities from following the IWF-5000 requirements for snubbers.

Request for Additional Information

13. Page 6, Table 2-1, lists part or subcomponent (e.g., embedments) that requires aging management review.

    (a) please describe the "Embedments" that are subject to aging management review.
    (b) are these embedments identical to that of "concrete embedments" shown on Page 64?

Response

(a) The embedments subjected to aging management review per this GTR (see Section 2.1 of GTR) are those that are between the interface of the structural member of the support and the concrete. Clarification will be added to Section 2.3. See also response to RAI #21.

(b) Yes. Clarification will be added to the report (see response to RAI #19).

Request for Additional Information

14. Chapter 4, Table 4-4 provides surveillance techniques to detect the effects of SCC on bolting.

Please discuss how bolting degradation of RCS support system is monitored/evaluated by the ASME Code, Section XI requirements as stated in Table 4-4. Also discuss which utility commitments to IEB 82-02 or GI B-29 are capable of managing the effects of SCC on the RCS supports bolting. Please verify whether the "Generic Issue B-29" is associated with bolting degradation.

Response

It is noted that in EPRI NP-5769, Volume 1, identifies U.S. NRC Generic Issue B-29 (GI B-29) as pertaining to degradation and failure of bolting in nuclear power plants. To avoid confusion, this issue is referred to as U.S. NRC Generic Safety Issue 29 (GSI-29) in the report.

The attribute table given in GTR Table 4-4 identifies ASME Section XI, Subsection IWF as the reference for surveillance techniques. In the response given for RAI #2, the use of ASME Section XI to monitor/evaluate bolting degradation of RCS supports is discussed.

A baseline evaluation of the RCS support bolts to the guideline of EPRI NP-5769 including NUREG-1339 is recommended as commitments to IEB 82-02 or GSI-29 to manage the effects of SCC on the RCS support bolting. This will be added to Subsection 4.2.2.

GSI-29 is associated with bolting degradation. In NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants" (June 1990), this issue is discussed.

Request for Additional Information

15. Page 35, Subsection 2.6.3, states, "However, concrete degradation is a significant concern since it provides restraint for the support anchorage system."

Please describe the RCS support anchorage system to the concrete structure that is the integral part of the support system.

Response

See response given for RAI #21.

Request for Additional Information

16 Page 37, Table 2-8, shows that ultrasonic examinations are periodically used for detecting integrity at RCS steel support bolted or welded connections.

(a) describe when and how ultrasonic examinations are intended to be performed during the period of extended operation, (b) is the frequency of the periodic examination consistent with the ASME Section XI Code requirements? (c) discuss the use of ultrasonic examinations in specific aging management programs (e.g., ultrasonic examinations are not indicated in Table 4-2 through Table 4-4).

Response

(a) Ultrasonic examinations may be performed during the period of extended operation if the visual examinations detect surface flaws that exceed the established criteria. These examinations are performed to assist in determining the character of the flaw (size, shape, and orientation). Ultrasonic examinations are part of the supplemental examinations discussed in ASME Section XI, IWF-3200.

(b) The frequency of the periodic examinations using ultrasonic methods is consistent with ASME Section XI Code requirements since the need for such examinations is based on the results of the visual examinations that are performed consistent with ASME Section XI.

(c) The need of ultrasonic examinations is based on findings from visual examinations. The acceptance criteria for visual examinations are given in IWF-3400 for the steel component supports. To determine more information on a potential flaw, ultrasonic examinations may be performed following IWF-3200, as stated below (1989 Code Addenda):

> Examinations that detect conditions that require evaluation in accordance with the requirements of IWF-3100 may be supplemented by other examination methods and techniques (IWA-2000) to determine the character of the flaw (that is, size, shape, and orientation). Visual examinations that detect surface flaws that exceed IWF-3400 criteria shall be supplemented by either surface or volumetric examinations.

Ultrasonic examination is a volumetric examination method in IWA-2000. The above will be added to Section 4.2. Further, for clarification purposes, Section IWF-3200 will be added to Tables 4-2 and 4-4, which pertain to steel supports.

(d) In Table 2-8, the rows labeled UT and Dye Penetrate/Magnetic Particle methods in the column labeled Periodic? will be changed to supplemental (identified as S in the table) periodicity.

Request for Additional Information

17. Page 47 (fourth paragraph) states, "RCS component supports are not susceptible to mechanical wear. This is because of the wear-resistant material used, ..."

    Provide a discussion of the wear-resistant material used for the RCS supports.

Response

In cases where a primary component is designed to slide on a support structure to accommodate thermal movement during heatup, special materials are used as wear plates at the support interface with the component. Wear plate materials are: 1) self-lubricated "Lubrite" plate, and 2) Timken Graph-Air tool steel. The type of base material used for the Lubrite plate is ASTM A-48. It is noted that the extent of relative movement between a component and its support over the lifetime of the plant is quite small. A reactor vessel nozzle pad moves about 3/8-in. during plant heatup, which works out to less than 1 in. movement per year of plant operation.

The above will be added at the end of Section 2.3 of the report.

Request for Additional Information

18. Page 47 (fourth paragraph) states, "Current inspection programs based on visual examination are employed to identify binding."

    Identify and describe the current inspection programs used to identify binding.

Response to RAI #18 (Cont.)

Response

As discussed in Section 2.6 of the report, maintenance programs follow the ASME Code. The regulations and rules that govern the inspection of primary component and surge line supports begin at the top level with the Code of Federal Regulations. Document 10 CFR 50.55a references Section XI of the ASME Code. Component supports that are subject to examination are examined in accordance with Table IWF-2500-1.

As part of the initial hot functional startup testing, the thermal deflection of the primary loop piping and components is measured at several temperature plateaus from ambient conditions to hot standby. Measurements are taken at lateral support/restraint locations, and between components and piping and the building structure. These measured deflections are compared with the theoretical primary loop movements to ensure that the system is deflecting as it should without binding and to obtain accurate as-built data that will be used to determine shim sizes for the lateral equipment supports and restraints. Once the component support shims and pipe restraint shims have been installed, the support gaps are again monitored during initial heatup to criticality to demonstrate that the support shims were sized properly. Should there be any major construction programs during the life of the plant that could potentially affect the primary loop thermal expansion or support shim sizes (e.g., steam generator replacement), the thermal behavior of the loop is monitored after completion of the construction program to ensure that the loop deflects as it should, and the support shim sizes are adjusted if necessary. From all of the inspections and measurements made, assurance is obtained by the utility that the piping and components are responding properly and binding is not an issue.

The above discussion will be added to Subsection 3.2.5.

Request for Additional Information

19. Page 62, Table 4-2, "Steel supports" are shown under Component for RCS supports application.

    Clarify whether the scope of steel supports also includes base plates, or embedded plates, and anchor bolts. If not, provide the justification for exclusion.

Response

Base plates, embedded plates, and anchor bolts are included as part of the local embedment, which is within the scope of this report. This will be clarified in Section 2.1, which addresses scope boundary, and also in Table 4-2.

Request for Additional Information

20. Page 62, Table 4-2, lists "Leakage identification walkdowns and leakage monitoring" under surveillance techniques attribute to detect the aging effect.

    Please provide an additional description and justification on leakage identification walkdowns and leakage monitoring for detecting and managing the aging effect.

Response

Leakage identification walkdowns and leakage monitoring as pertaining to Table 4-2 are discussed in Subsection 4.2.1. There is no aging effect unless there is an event (leakage) that has the potential to cause an aging effect. The recommended procedures are based on acceptable methods currently used by utilities to address U.S. NRC Generic Letter 88-05 issues concerned with corrosive effects of RCS leakage. Leakage identification walkdowns are performed in accordance with ASME Code Section XI, Subsection IWF and the plant-specific commitments in response to Generic Letter 88-05. This information is stated in the third paragraph on page 60 (Rev. 1) of Subsection 4.2.1, Aggressive Chemical Attack and Corrosion (AMP-1.1 and AMP-1.2). It is felt that this information is provided at an appropriate level of detail for a generic report.

Leakage monitoring occurs by monitoring any or all of the parameters listed in Table 4-2, Acceptance Criterion 3. These parameters will be copied to the Surveillance Technique attribute. No additional descriptions of the attribute will be provided since it is felt that monitoring of any of the plant parameters is clear.

Provided below is a discussion of leakage identification and walkdown along with corrective actions. Subsection 4.2.1 should also be consulted.

The external surface of an RCS support is potentially exposed to borated water (in some cases demineralized water) the event 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 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.

The following text will be added to the end of Subsection 4.2.1 to explain why leakage identification walkdowns and leakage monitoring adequately detect and manage aging effects.

This program can manage the aging effects related to leakage of borated water onto RCS supports because the program provides three diverse methods for detecting the aging effect.

When unacceptable effects are detected, corrective actions would be initiated. The three diverse methods for detecting material wastage due to boric acid corrosion are:

- Inspection of supports as specified by ASME Section XI, Subsection IWF
- Identification of leakage during walkdowns
- Leakage monitoring

In terms of risk, aging effects caused by leaking borated water are most likely to occur during operation due to the high-pressure and high-temperature environment, or when the supports have not been inspected for a period of time. Experience has shown that leakage occurs at bolted connections, threaded fittings with seal welds, or branch line connections to piping or components. RCS support components are located away from these locations and the potential harsh environment caused by leakage. During this period, plant-specific leakage monitoring programs would identify that external system leakage was occurring and an appropriate inspection would be initiated.

Before, during, and after refueling outages, the plants also inspect the RCS, including the supports. This inspection would detect leakage too small for the a plant's leakage monitoring program and too small to cause a loss of an intended function during the previous operating period of the plant, or when the supports were last inspected. Finally, inspections as specified by ASME Section XI, Subsection IWF would augment the refueling inspections. Evidence of unacceptable aging effects would be corrected as described above. The combination of detection and repair in a timely manner maintains the intended function of the RCS supports. Operating experience has demonstrated the effectiveness of this program (IWF) because no reported damage has occurred to the RCS supports due to leakage.

Request for Additional Information

21. Page 64, Table 4-3, lists "Concrete Embedments" under RCS supports application.

    Please describe the concrete embedments and state whether they include embedded plates, or base plates, or embedded structural steel members (e.g., structural steel channels), or embedded bolts, or concrete expansion anchor bolts.

Response

See also response given for RAI #19.

The primary equipment support embedments are typically cast-in-place anchor bolts, through-wall anchor bolts, or cast-in-place weldments.

Response to RAI #21 (Cont.)

Anchor bolt designs include hook bolts, threaded bolts with individual washer plates and nuts, and groups of bolts sharing a common washer plate. Bolt sizes may range as large as 4 in.

diameter and may have lengths up to 7 or 8 ft. A variety of bolt materials are used, such as ASTM A36 or A588 threaded rod, to ASTM A354, A490, or A540 bolting material. Concrete expansion anchors are generally not used for the primary equipment supports (they may be used for surge line hangers in some cases).

Embedded weldments are typically fabricated using structural plate material, structural shapes, or a combination of both. The embedded weldments are constructed of the same materials as are the support structures.

This discussion will be added to Section 2.3 and Subsection 2.4.1, which pertain to description and materials, respectively.

Request for Additional Information

22. Page 58, Table 4-1, under the Attribute ACTIONS, the description states ".... Preventive actions should include evaluation of failures to determine where similar effects may occur and actions, if practical, to mitigate or ...."

    Please describe what actions will be taken, if it is determined that it is impractical to mitigate or eliminate the effects from occurring or delete the words if practical.

Response

The words "if practical" will be changed to "as necessary" in Table 4-1. Also, the title for this attribute should be "Corrective Action," and the second sentence will be changed to the following:

> Corrective actions should include evaluation of failures to determine where similar effects may occur and actions, as necessary, to mitigate or eliminate the effect from occurring elsewhere.

Request for Additional Information

23. Page 64, Table 4-3, under surveillance techniques to detect aggressive chemical attack and corrosion for concrete embedments, ACI 349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," Revision 1, March 1996 should be added to the list of ACI guidance for inspecting concrete embedments. This standard provides inspection guidance for concrete structures in addition to that in the three ACI standards that are listed in Table 4-3.

Response

This reference will be added to Table 4-3 under the Surveillance Technique attribute as well as Section 2.6.

Request for Additional Information

24. Page 14 (Section 2.4) states, "The data given herein ... provide judged bounds." Discuss the meaning of "judged bounds." Also, discuss what are renewal applicant actions to ensure that they are within these "judged bounds" when referencing the WOG report.

Response

Judged bounds refers to the generically applicable boundary defined by the author based on available data, experience, and engineering judgment. These data are intended to "bound" a majority of domestic Westinghouse plants. These judged bounds have been reviewed and accepted by utilities participating in the WOG LCM/LR program.

The actions an applicant must take to verify their plant is bounded by a GTR will be described as stated in the response to RAI #1.

Request for Additional Information

25. Page 34 (Section 2.6) lists "General Maintenance Practices" for the RCS supports. These are programs to monitor the RCS supports such that unanticipated aging degradations developed during plant service would be detected and managed. Clarify whether the WOG proposes to include these listed programs as a part of the aging management of the RCS supports during the period of extended operation.

Response

The maintenance practices used by the utilities follow ASME Section XI and ACI practices, or for earlier vintage plants something similar. The recommended practices to manage the significant aging effects as defined in the aging management program attribute tables follow the practices described in Section 2.6. Therefore, these listed programs are part of the aging management of the RCS supports during the period of extended operation as defined in Tables 4-2 to 4-4 of the report.

Request for Additional Information

26. Page 34 (Section 2.6) discusses "General Maintenance Practices" and then, in the next section (2.7) jumps into "Conclusions-Aging Mechanisms." There is no evaluation of how the applicable aging mechanisms and effects are identified. Further, this format may not be consistent with the WOG "Report Template." Provide the aging mechanism/effect evaluation.

Response

It is stated in Section 2.7 that the applicable aging mechanisms are identified from a review of the industry issues and maintenance history. The industry issues that are related to the aging mechanisms and effects are discussed in Section 3.0. The aging mechanism/effect evaluation is provided in Section 3.0, which is consistent with the GTR template. Furthermore, the content in Sections 2.6 and 2.7 follows the format of the template with the exception that the titles are different:

Section 2.6 is titled "General Maintenance Practices," whereas the template title is "Maintenance History."

Section 2.7 is titled "Conclusions - Aging Mechanisms," whereas the template title is "Aging Effects."

This is a group 1 report that preceded the latest GTR template. The format of the GTR was not necessarily modified to be identical to the template. Changes were made when beneficial with respect to content and clarity and not necessarily for style. The intent of Section 2.7 is to identify aging mechanisms or effects that require evaluation in Section 3. Section 2.6 has been revised to introduce the industry issues that are the source of all known aging mechanisms to the supports within the scope of the GTR.

Request for Additional Information

27. Page 46 (Section 3.2.4) states, "Temper embrittlement ... are seen in ferritic materials." Because RCS support components are fabricated from ferritic materials as listed in Table 2-4, provide a discussion on the potential of thermal aging embrittlement of RCS supports.

Response

In general, RCS supports are operated at the temperature below 450EF. Therefore, temper embrittlement is not a concern for the ferritic materials of RCS supports. This will be clarified in Subsection 3.2.4.

Request for Additional Information

28. Page 49 (Section 3.2.6) states, "... to characterize a defect or flaw found as the result of an unacceptable visual inspection condition Paragraph IWF-3122.3 ... would include standard procedures for fatigue crack growth analysis ..." However, the next paragraph of the WOG report states, "For the supports within the scope of this report, this type of analysis is not necessary." Clarify what actions are to be taken when defects or flaws are found by visual inspections of the RCS supports during the period of extended operation.

Response

The referenced section of the report pertains to fatigue. For the RCS supports, the stresses will be of such low value that fatigue cracks or growths will not occur making it unnecessary to perform fatigue crack growth analyses. This will be clarified in Subsection 3.2.6 by the deletion of the discussion related to the characterization of a defect or flaw.

Request for Additional Information

29. Page 53 (Section 3.2.8) discusses the potential degradation of concrete due to neutron irradiation and references it to the resolution of GSI-15. NUREG-1509 was issued in May 1996 to resolve GSI-15, and it did not address concrete degradation due to radiation.

    Provide justification of aging management of concrete degradation due to radiation for renewal.

Response

Since Revision 1 of the report has been issued, the WOG has decided to address GSI-15 plant-specifically. Concrete degradation due to radiation will also be addressed in this manner. A utility would submit a plant-specific evaluation at the time of the renewal application or reference a generic evaluation subsequent to this GTR. Aging Effect Evaluation of Subsection 3.2.8 has been removed and replaced by the following sentence:

> Concrete degradation due to radiation will be addressed by plant-specific evaluation.

Request for Additional Information

30. Page 57 (Section 4.0) states, "The plant-specific programs developed by utilities will demonstrate that aging effects are managed ..." The staff believes the intent of the WOG report is to provide the technical basis to justify why specific aging management program attributes will result in programs that adequately manage aging of the RCS supports for renewal. The utilities are to implement their plant-specific programs in accordance with the attributes. The "demonstration" or technical basis for the program attributes should be provided in the WOG report. See related question #2.

Response

The first paragraph will be modified as follows:

In this section, options to manage the effects of aging are presented. Since this report is generically applicable to the plants identified in Section 1.1, only program attributes are given.

These attributes are described, and their effectiveness during an extended period of operation is justified. This provides the generic demonstration that aging effects are managed so that intended functions will be maintained consistent with the current licensing basis (CLB) during an extended period of operation. Plant-specific details, based on these attributes, will be developed as part of the license renewal application~~s~~, and complete the demonstration process. ~~The plant-specific programs developed by utilities will demonstrate that aging effects are managed so that intended functions supported by the reactor coolant system (RCS) supports are maintained for an extended period of operation.~~

Request for Additional Information

31. Many of the attributes included in the program attributes tables (beginning on page 62) are based on Subsection IWF of Section XI. Discuss if there are any inspections/procedures in Subsection IWF that are omitted from these attribute tables. Also, identify the IWF inservice inspections that the WOG believes is not necessary to manage the effects of aging on the RCS supports.

Response

Inspections/procedures that are shown specifically address the required attribute actions as defined in Table 4-1. The Section XI subsections referenced make up a complete program to address the aging effects to which the attribute tables pertain; none have been omitted that are related to the aging effect to be managed. It is not necessary to identify IWF inservice inspections that are not necessary to manage the aging effects. It is noted that within these subsections, other code sections are referenced that would also apply. Some of these subsections pertain to alternate and supplemental examination methods. It is not necessary to list these referenced subsections since they can be directly obtained from the referenced IWF subsections. Therefore, the inservice inspections/procedures that are necessary to manage the effects of aging of the RCS supports are those that are referenced.

Request for Additional Information

32. The discussion on fatigue in Section 3.2.6 contains an estimate of the limiting stresses and number of cycles for supports. Verify that the discussed fatigue evaluation has considered these effects on the supports, or alternatively, consider these effects at the most limiting locations.

Response

As noted in the report, fatigue is not an issue for the RCS supports since they have small usage factors. This is the basic reason why a design fatigue evaluation is not performed forThe supports within the scope of this report. An estimate of the maximum fatigue usage in the RCS supports is performed to demonstrate that fatigue is not an effect that is a concern for the support structures. The reported fatigue evaluation is performed using the stress at the most limiting location. The stress is defined by the upper-limit allowable stresses.

Editorial Comments

1. Page 31 (third paragraph), "Table 2-14 lists the code ... " Table 2-14 should be Table 2-7.

Response

Agree.

Editorial Comments

2. Page 33, Surry 1 & 2, NF/'63 is incorrect. either NF/'74 or AISC/'63.

Response

The design code for Surry 1 & 2 is AISC/'63.

Editorial Comments

3. Page 35, Section 2.6.1, "However, IWF-3000 gives general criteria for acceptance or rejection." Change "general criteria" to "specific standards."

Response

Agree.

Editorial Comments

4. Page 49 (first paragraph), "which would include standard procedures for fatigue crack growth analysis contained elsewhere in ASME Code, Section XI,? Change "elsewhere in" to "in Appendix A of."

Response

Agree, however, this portion has been deleted. See RAI #28.

Editorial Comments

5. Page 54 (second paragraph), "US NRC issued NUREG-0577 in September 1979, revised October 1993." Please verify 1993 with Reference 11 (it should be 1983).

Response

Agree.

Editorial Comments

6. Page 33, Table 2-7, AISC normally does not have the year of publication but goes by editions, such as sixth edition (copyright 1963, 1964, 1965, 1966, and 1967), seventh edition (copyright 1970 and 1973), eight edition (copyright 1980), etc. The designation of AISC/'69 and AISC/'63 are seldom used.

Response

The AISC codes were identified by the year of publication rather than the particular edition since they are generally referenced in that manner in the various plant FSARs and stress reports. However, AISC '63 corresponds to the 6th Edition, and AISC '69 corresponds to the 7th Edition.

Editorial Comments

7. Page 72, Reference 27 of ACI 349.3R-95 which has been published in March 1996.

Response

So reflected in revised report.

Editorial Comments

8. Page 5, Section 2.1, line 8: mode should be modes.

Response

Agree.

Editorial Comments

9. Page 35 (Section 2.6.1) discusses inservice inspections and states, "Indications are generally reported but not dispositioned as acceptable or rejectable." However, Section XI requires inspection indications to be documented and evaluated. Revise the statement as appropriate.

Response

The statement has been revised to state: " ... Indications are generally reported and documente for evaluation. IWF-3000 [Ref. 6] gives specific standards for acceptance or rejection. ... ."

Editorial Comments

10. Page 55 (Section 3.3) indicates that fatigue is the only TLAA applicable to the RCS supports. Although this TLAA is generic to the RCS supports, there may be plant-specific TLAA applicable to the RCS supports which are not addressed by the WOG report.

    Revise the WOG report to indicate that a renewal applicant need not to perform additional evaluation relating to fatigue as a TLAA. However, a renewal applicant still needs to identify and evaluate plant-specific TLAA applicable to their RCS supports, if any.

Response

At the end of Sections 3.3 and 5.0 the following statement has been added: "However, a renewal applicant still needs to identify and evaluate plant-specific TLAAs applicable to their RCS supports, if any.

Editorial Comments

11. Page 44, Section 3.2.2, paragraph 3, line 5, change the word "dry" to "drying."

Response

Agree.

Editorial Comments

12. Page 58, Section 4.2, second paragraph, line 7, add the word "supports" after the word "piping" because the piping supports, not the piping itself (presumably), will be inspected.

Response

Agree.

Editorial Comments

13. Page 57, Section 4.1, the reference to 10 CFR 54.33(c) should be changed to 10 CFR 54.33(d).

Response

Agree.

Supplementary Editorial Comments from NRC Meetings

1. In Table 4-2, scope attribute, under component the wording should be "Steel supports including embedments"

Response

Agree.

Supplementary Editorial Comments from NRC Meetings

2. In Table 4-4, under Scope attribute, the EPRI guideline should be NP-5769 and not NP-5796.

Response

Agree.

7.2 U.S. NUCLEAR REGULATORY COMMISSION REQUEST FOR ADDITIONAL INFORMATION

This appendix provides questions and responses associated with the United States Nuclear Regulatory Commission (U.S. NRC) requests for additional information (RAIs). Also provided is a table correlating the RAI with the section in the report that was changed due to the RAI. Note that section numbers given in the RAIs refer to section and page numbers in the original release of WCAP-14422.

RAI and Report Section Cross-Reference Table

RAI NUMBER	DESCRIPTION	REPORT SECTION
1	Maintenance Programs	2.6, 4.0, 4.2
	Stress Corrosion Cracking (SCC)	4.2.2
2	Adequate Aging Management in Extended Period of Operation	4.0
3a	Maintenance Practices and Observations	2.6, 4.2
3b	Leaks	3.2.2, 4.2.1
3c	Mechanical Wear, Binding, and Erosive Wear	3.2.5, 4.2.1
3d	Additional Aging Management Programs for Renewal	4.0
4	Scope	2.1
5	Technical Issues and Operating Experience	2.4.6, 3.1, 4.1
6	References	6.0
7	Section XI Code Edition	2.4.6, 4.2, 5.1
8	Section VII and VIII	4.2
9	Editorial	Executive Summary
10	Scope	2.1
11	Code Editions	2.1, 2.4.6
12	Allowable Capacity Definitions	2.1
13	Inaccessible Areas	4.2.1
14	Support Movements (e.g., sliding)	2.3
15	RPV Cooling Water Corrosion	4.2.1
16	Compression Bumper Aging	
17	Snubbers (Active Function)	
18	Springs	2.1, 4.2
19	Sampling	4.2
20	Welds B AISC Designs	2.4.1
21	GSI 15	3.2.3, 3.2.8, 4.1

RAI and Report Section Cross-Reference Table (Continued)

RAI NUMBER	DESCRIPTION	REPORT SECTION
22	Radiation Data - Clarification	2.4.4
23	Full Energy Spectrum	2.4.4
24	Editorial - Clarification	3.1
25	USI A-12	4.2, 4.2.2, 5.1
26	Aging Management	4
27	Loose Bolts	4.2
28	Boric Acid Management	4.2.1
29	Boric Acid Management	4.2.1
30	Water Leakage	4.2.1
31	Thermal Aging of Cast Stainless Steel	3.2.4
32	Thermal Embrittlement Temperature	3.2.4
33	Fatigue Crack Growth	3.2.6
34	Fatigue Design Requirements	3.2.6
35	Request for more Fatigue Data	–
36	Concrete Temperature	3.2.8(c)
37	SCC Support Bolting	4.2.2
38	Information Bulletin 82-02	4.1
39	Loose Bolts Due to Vibration	–
40	Type (1) SCC Analysis	–
41	Type (1) and (2) SCC Analyses	–
42	NRC Review of Analyses	–
43	Concrete Scope	2.1
44	Masonry Walls	2.1
45	Clarification & Specific Maintenance Programs	2.6, 4.2.1
46	Prestress	–
47	pH	3.2.8(b), 4.2.1
48	TLAA B Fatigue	2.5, 3.2.6, 3.3
49	Use of ASME Section NB for Fatigue Evaluation	–
50	References	6.0
51	Editorial	Throughout

Request for Additional Information

1. Provide additional detail regarding aging management programs in terms of actions taken, results, and validity for the period of extended operation. The report must clearly identify and describe any programs used to manage the effects of aging during the period of extended operation such that an applicant for license renewal can readily verify whether or not its plant is within the conditions evaluated in this report.

   The following are some examples of lack of information:

   (a) Page 47 (Section 4.8.2) states, "The effects of cracking and rebar corrosion and aggressive environments can be managed by current inspection and repair programs." However, there is no substantive discussion of these programs.

   (b) Page 48 (Section 4.9.2) states, "Another strategy involves performing inspections at regular intervals to assure that SCC is not an active mechanism." However, there is no substantive discussion of these inspections.

   (c) Page 48 (Section 4.9.3) states, "As part of the ASME Section XI inspection requirements, the concrete ... is visually inspected." However, there is no discussion relating to the specific inspection requirements. There is not even a code subsection citation, such as IWF or IWL.

   (d) Page 50 (Section 4.11) and page 3 (Section 1.0) states "... functions ... are assured by existing programs and practices." However, there is no identification of specific programs necessary for renewal.

Response

The report has been modified to provide an additional level of detail regarding the programs that manage the effects of aging for a period of extended operation. A new Section 4 that identifies and describes the program attributes to be used for managing potentially significant age-related degradation effects so that a license renewal applicant can readily verify whether or not its plant conforms to the evaluation criteria has been added.

Attribute tables are provided describing acceptable maintenance programs to manage the potential aging effect. These programs are based on ASME Section XI, 1989 code edition, following Examination Category F-A. In Section 4.2, the following words are added to provide more detail in addition to the attribute tables:

> Examination Category F-A calls for the visual (VT-3) inspections following the inspection schedule given in IWB-2000, "Examination and Inspection." For license renewal, inspections should be performed in accordance with Table IWB-2412-1, Inspection Program B, and IWF-2420. These inspections are supplemented by IWF-2430 "Additional Examinations." In addition, supports other than piping should receive a 100-percent inspection, and Class 1 piping supports should receive a

Response to RAI #1 (Cont.)

> 25-percent inspection at each interval. Component support conditions observed during the VT-3 inspection that are unacceptable for continued service include: improper hot and cold settings of spring supports; misalignment of supports; deformation or structural degradation of fasteners, springs, clamps, or other support items; and missing, detached, or loosened items. When such conditions are observed, supplemental examinations based on visual (VT-1) inspection, or surface (dye penetrant or magnetic particle), or volumetric (radiographic or ultrasonic) examination may be used to determine the mechanism causing the flaw (effect).

In the new Section 4, Section 4.2.2, stress corrosion cracking (SSC) is addressed as follows:

> The effects of SCC have been determined to be potentially significant for structural bolting used in Class 1 component supports in Subsection 3.2.1 of this report. These effects may include bolting that is cracked or missing, as the result of overly hardened or high-strength material operating in a moist environment under sustained tensile stress. Periodic inservice inspections, in accordance with the ASME Code Section XI, Subsection IWF, plus any utility commitments in their CLB in response to IEB 82-02, Unresolved Safety Issue (USI) A-12, or GSI B-29, are capable of managing these effects in the license renewal period. Utility commitments in their CLB to the resolution of USI A-12 or GI B-29 may include hardness testing of support bolting to determine items for augmented inspections.

Request for Additional Information

2. Provide additional justification 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." For example, if the American Society of Mechanical Engineers (ASME) Section XI Subsection IWF inspections are relied on to manage aging for renewal, technical justification that such inspections would manage the effects of aging to ensure the structure or component's intended function for the period of extended operation should be provided in Section 4 of the report where it discusses aging management.

Response

In the new Section 4.0, the actions that a utility is to take as part of the license renewal application are summarized. The utilities would follow the acceptable aging management activities and program attributes given in the GTR, or they would provide the basis for implementing different options. Further, they would provide the basis of their maintenance program for the license renewal term if:

- Their aging management activities are different from the methods given in the GTR.
- Their plant falls outside the parameter ranges given in the GTR.

- The procedures required to address industry issues are implemented in a different manner.

See also response to RAI #1.

Request for Additional Information

3. Clearly state what programs (existing or additional) are relied on to manage aging for renewal. Consistent with the present format of the topical report, these programs should be discussed in Section 4.9, "Component Aging Degradation Management," and summarized in Section 4.11, "Aging Evaluation Summary."

   The following are some examples of lack of clarity:

   (a) Page 16 in the "Support System Description" section is the only place that mentions ASME Section XI Subsection IWF inspections. Is Subsection IWF relied on as an aging management program for reactor coolant system (RCS) supports during the period of extended operation?

   (b) Page 39 (Section 4.2.2) states, "... no further evaluation is required beyond assuring the absence of reactor coolant leaks." Is the Westinghouse Owners Group (WOG) committing to a leakage monitoring program for renewal?

   (c) Page 42 (Section 4.5.2) states, "Visual monitoring can be employed to identify binding without augmenting the current inspection programs." Is WOG committing to a visual monitoring program for renewal? Also, what are visual monitoring and current inspection programs?

   (d) Page 50 (Section 4.11, "Aging Evaluation Summary") states, "... it is not necessary to modify the existing maintenance and inspection programs for these structures." It also states, "... functions expected to be performed by the subject supports, are assured by existing programs and practices." Thus, the WOG appears to say that no additional program is needed for renewal. However, for example, page 47 (Section 4.9.1) appears to describe a specific additional program to manage aging of bolts for renewal. Is WOG committing to any additional aging management programs for renewal?

Response

A new Section 4 was added to the report to describe and evaluate the program attributes necessary to manage potentially significant effects of age-related degradation.

(a) The discussion of periodic inservice inspections carried out in accordance with Subsection IWF of the ASME Code Section XI is emphasized (see the response to RAI #1).

(b) The discussion in Subsection 3.2.2, Aging Effect Evaluation, will be revised to read:

> Corrosion or aggressive chemical attack does not occur to the RCS supports unless there is a leakage event. The leakage of primary coolant and the subsequent evaporation and rewetting cycles can lead to a concentrated boric acid slurry and subsequent corrosion of low-alloy and carbon steel components. Note that once boric acid leakage is controlled and the wastage is cleaned, no long-term damage will result. Leakage of demineralizers is also possible; however, it is not as serious as that associated with borated water. Therefore, the effects of general corrosion of RCS support components are not significant, unless the surfaces are exposed to boric acid or demineralized water. Under these conditions, the potential aging effect is possible.

Then, in Subsection 4.2.1, the following discussion will be added:

> The effects of general corrosion of RCS support component surfaces exposed to leaking primary coolant were found to be potentially significant in Subsection 3.2.2 of this report. These effects may include loss of material, discoloration, or accumulated residues on surfaces of components, insulation, or floor areas. Periodic inservice inspections, in accordance with the ASME Code Section XI, Subsection IWF, plus any utility commitments in their CLB in response to Generic Letter 88-05, are capable of managing these effects for both the current and any license renewal term. Examination Category F-A calls for visual (VT-3) inspections that would include visual monitoring of the condition of any lubricant as well as checking for binding.
>
> Component support conditions that are unacceptable for continued operation include general corrosion resulting in loss of intended functions of the RCS supports. If the external surfaces are inaccessible, the surrounding area, including floor areas or equipment surfaces located underneath the components, is visually inspected for evidence of leakage. The relevant conditions for visual inspections include: (1) area of general corrosion of a component resulting from leakage, and (2) discoloration or accumulated residues on surfaces of components, insulation, or floor areas that may be evidence of borated water leakage.

Response to RAI #3 (Cont.)

(c) Wear is not a significant aging effect to the RCS supports because of: the wear-resistant material used; the low frequency (number of times) of movement; and the slow movement between sliding surfaces. However, as noted in Subsection 3.2.5, in a corrosive environment, corrosion may affect the performance of sliding surfaces where binding could potentially occur. The following material will be added to Subsection 4.2.1:

> Periodic inservice inspections, in accordance with the ASME Code Section XI, Subsection IWF, plus any utility commitments in their CLB in response to Generic Letter 88-05, are capable of managing these effects for both the current and any license renewal term. Examination Category F-A calls for visual (VT-3) inspections that would include visual monitoring of the condition of any lubricant as well as checking for binding.

(d) For the components within the scope of this generic report, no additional maintenance techniques or methods are needed than those used within the current plant operating term (see response to RAI #1). There are additional programs that have specific actions that utilities must follow in the current term and are recommended for continuation into the extended term. These programs are the result of agreed actions necessary to address industry issues. One example is stress corrosion cracking of bolts. See response to RAI #37.

Options to manage the effects of aging are presented in Section 4.

Request for Additional Information

4. Clearly define the scope of the report. Although Section 3.0 (beginning on page 7) describes the RCS supports, the specific parts of the RCS supports that are within the scope of the report are not described. Some examples: Is the entire shield tank within the scope? Are integral attachments within the scope of the topical report? Address base plates, anchor bolts, adjacent concrete, and associated building structures (such as reactor cavity wall and basemat). Is the scope the same as that inspected under ASME Section XI Subsection IWF? In any case, the staff believes that concrete areas which affect the function of the RCS supports should be within the scope of the report.

Response

The scope will be clarified in Section 2.1. The following will be added:

The scope of this includes:

- Primary component supports for:
  - Reactor coolant pump (RCP)
  - Reactor pressure vessel (RPV) (note that the neutron shield tank is included in the scope and is described in RPV configuration 4; also the support ring is included, as described in configuration 3)
  - Steam generator (SG)
  - Pressurizer (PZR)
- PZR surge line supports, including springs

References made within this report to RCS supports include the above scope.

The boundary between the components and structures is:

- Up to, but not including, integral attachments that are on the components (integral attachments are discussed in the specific component generic reports, e.g., PZR support skirt boundary at PZR).
- Lugs, nozzles, or welds on component shells are also not included. They are also discussed in the specific component generic reports.
- Concrete local to embedment is included, but not the concrete adjacent to embedment (this portion is included in the generic report associated with seismic Class 1 structures). Base plates, embedded plates, and anchor bolts are included as part of the local embedment that is within the scope of this report.

Excluded are:

- Pipe whip restraints - covered in another GTR
- Masonry walls - none related to RCS supports
- Portions of snubber supports that perform intended function in an active manner

Request for Additional Information

5. Provide a discussion of RCS supports operating experience relating to aging, including applicable generic communications. Operating experience should be factored in when evaluating aging management programs. Also, responses to generic communications may contain aging management programs necessary for the period of extended operation.

Response

No documentation related to operating experience associated with aging of GTR supports has been found. Technical issues related to the original design basis have been identified and documented. Examples of such cases are:

- USI A-12 and NUREG-0577, low fracture toughness and lamellar tearing
- NRC Bulletin 88-11, pressurizer surge line thermal stratification B potential high thermal stresses may occur affecting supports and piping
- NRC IEB 88-08, thermal stresses in piping connectors
- Generic Letter 88-05, corrosive effects of reactor coolant system leakage
- SCC, stress corrosion cracking-localized cracking failure

The technical issues are addressed through specific actions that are required to address U.S. NRC directives. The programs are followed by utilities during the current term and as necessary into an extended period of operation.

This is discussed in Subsection 2.4.6, and Sections 3.1 and 4.1 in the report.

Request for Additional Information

6. All references in the report must be publicly available. Submit the necessary references that are not in the public domain. If any reference report is proprietary, the proprietary report must be submitted to NRC along with a nonproprietary version and a justification demonstrating that the report meets the criteria established in 10 CFR Part 2, Section 2.790.

Response

The references are publicly available. No proprietary references are used.

Request for Additional Information

7. National codes and standards are mentioned in the report, such as ASME Section XI, but without identifying specific editions. The staff needs to review technical elements of specific aging management programs. Identify specific code editions in the report. (License renewal aging management programs that are based on a specific edition of the ASME Section XI code may be changed by the licensee using 10 CFR 50.59 during the renewal term).

Response

Acceptable aging management programs for license renewal given in the GTR are based on ASME Code Section XI, 1989 edition or other codes (e.g., AISC) as identified in the GTR. A utility may follow a different program; however, they must provide a description of their program as part of their license renewal application. This is discussed in Subsection 2.4.6 and Sections 4.2 and 5.1 of the GTR.

Request for Additional Information

8. If ASME Section XI is to be relied on to manage aging during the period of extended operation, describe commitments to Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," (1989 edition of Section XI), and Appendix VIII "Performance Demonstration for Ultrasonic Examination systems:" (1989 Addenda to the 1989 edition of Section XI).

Response

The following will be added to the GTR (Section 4.2):

> Component support conditions observed during the VT-3 inspection that are unacceptable for continued service include: improper hot and cold settings of spring supports; misalignment of supports; deformation or structural degradation of fasteners, springs, clamps, or other support items; and missing, detached, or loosened items. When such conditions are observed, supplemental examinations based on visual (VT-1) inspection or surface (dye penetrant or magnetic particle) or volumetric (radiographic or ultrasonic) examination may be used to determine the mechanism causing the flaw (effect). The extent to which individual utilities reference the mandatory Appendices VII and VIII of ASME Section XI in such augmented inspections will depend on the current inservice inspection (ISI) code of record at the plant. For codes of record prior to the 1989 Edition of Section XI, these appendices may not apply. Individual license renewal applicants may reference the appendices in conjunction with a description of the plant code of record.

Request for Additional Information

9. Page iii states, "Management procedures have been developed ..."for five issues. However, the report only discusses management procedures for two issues: bolting and concrete. The other issues appear to be evaluated and found acceptable. The statement should be revised.

Response

The discussion on Page iii was removed to avoid confusion.

Request for Additional Information

10. Page 1 states, "The scope of the report covers ... support systems of the major components of the RCS, including the reactor pressure vessel (RPV), the steam generator (SG), the reactor coolant pump (RCP), and the pressurizer (PZR), as well as the RCS pressurizer surge line supports." Because Section 3 of the report describes only supports for the RPV, SG, RCP, PZR, and surge line, does the report only cover these supports? The report mentions the reactor support ring. Is the reactor support ring within the scope of the report? Also, are there other supports for the RCS, such as for the main loop piping, which are excluded from the scope of the report? The scope description should be definitive and comprehensive.

Response

It is true that the report covers only the RPV, SG, RCP, PZR, and surge line. The reactor support ring is part of the RPV support (RPV configuration 3) and therefore, is included in the scope of the report. Further, note that there are no supports on the primary coolant loop piping. See also the response to RAI #4.

Request for Additional Information

11. Page 7 (Section 3.1.1) states, "... the supports are designed to meet ASME Code (AISC Specifications for plants prior to 1974) ..." Identify the specific design requirements for each plant listed in Table 2-1 or 3-2. Also, define "prior to 1974" in relation to date of construction permit or other suitable milestones.

Response

The code editions and which code or specification (ASME Code or AISC Specification) applies are given in Subsection 2.4.6. Prior to 1974, as defined by the contract date, the American Institute of Steel Construction (AISC) Specification was used.

Request for Additional Information

12. Page 7 (Section 3.1.1) states, "For the purposes of this discussion, margin is defined as the ratio of the allowable capacity load to the design condition load." Define "allowable capacity load" and discuss any relationship with code allowable, yield stress, or other code terms. Are the margins discussed in the report the same as in existing codes?

Response

The paragraph on page 8 will be rewritten to clarify the use of the terms. Margin has been replaced by factor of safety. The factors of safety are those reflected in the existing codes. They are well known and do not have to be discussed within the report. The term "allowable capacity load" has been removed. The replacement for the subject paragraph is given below (Section 2.1):

> ...The supports are designed to meet the ASME Code Section III, Subsection NF, or the AISC Specifications for plants whose contract date is prior to 1974. Code-allowable stress limits reflect a factor of safety against capacity. Different code factors of safety exist for the loading conditions (i.e., service level A, B, C, D) and mode of failure (e.g., buckling, shear, tension, bearing).

Request for Additional Information

13. Section 3.0 (beginning on page 7) describes RCS supports. The supports have areas of limited accessibility. Describe aging management programs relied on for renewal for inaccessible areas of RCS supports, such as for the reactor vessel supports and inaccessible sliding surfaces excluded from inspection by ASME Section XI, Subsection IWF (Item No. F1.60 in Table IWF-2500-1).

Response

The following is added to the report in Section 4.2.1:

> Examples of inaccessible areas within the scope of this generic technical report are:
>
> - Sliding surfaces
> - Embedments within concrete
> - Water-cooled reactor vessel supports (inside area)
> - Reactor pressure vessel (RPV) supports (limited access)

Response to #13 (Cont.)

For those designs where the reactor vessel supports are cooled by circulating water through or past bearing plates under the shoe, this is an inaccessible area. If corrosion is present, it would likely manifest itself as a leak prior to any significant weakening of the structural members of the supports.

Those areas that are inaccessible are currently excluded from the ASME inspection program. However, utilities must rely on visual examinations for evidence of degradation. Examples of visual evidence of degradation that would be used by a utility to identify potential problems within an inaccessible area are:

- Binding as evidence of local deflections or deformations that are unusual
- Leaking of fluid
- Discoloration or flaking of surface coating, indicating the presence of corrosion

Degradation of inaccessible areas may be managed by:

- Identify inaccessible areas
- Define visual evidence that will alert an inspector to potential degradation
- Assessing degradation and need for a more detailed inspection that may take the form of:
  - Use of a remote device (e.g., camera) to assess degradation
  - Use nondestructive examination (NDE) methods to evaluate degradation
  - Use core borings to evaluate concrete degradation
  - Repair degradation if serious
  - Identify mechanism causing degradation effect, evaluate and correct if feasible

Note that a utility may follow a different procedure, but they would be obligated per 10 CFR 54.21 to so indicate in their plant-specific application.

Request for Additional Information

14. Section 3.0 (beginning on page 7) describes the RCS supports. Provide additional information on how these supports slide or otherwise permit the thermal growth of the RCS, such as for steam generator support "configurations 2 and 5" and pressurizer supports.

Response

The following will be added to Section 2, providing additional information on how these supports slide or otherwise permit the thermal growth of the RCS, such as for steam generator support "configurations 2 and 5" and pressurizer supports.

> There are various primary equipment support concepts used to accommodate the thermal expansion of the reactor coolant loop and equipment. A large number of plants use pinned-end columns to support the SGs and RCPs, which rotate a small amount as the plant heats up. For support configuration 2, the skirt is mated to a stationary skirt ring girder anchored to the building structure floor using a number of roller assemblies. These roller assemblies permit horizontal movement of the support skirt relative to the stationary ring girder in the hot leg direction, while preventing movement perpendicular to the hot leg and vertical direction. A set of stops engage the skirt at the end of its thermal travel to provide restraint for seismic and pipe rupture loads. For support configuration 5, SGs and RCPs slide on Lubrite bearing pads located between the components and stationary supports. A large holddown bolt at each pad, or foot, fits in a slot in the stationary support plate oriented in the direction of thermal motion of the component. The holddown bolts also have shear load capability for seismic and pipe rupture restraint.
>
> Since pressurizers remain stationary during plant heatup, the thermal growth that must be accommodated is in the vertical and radial direction at the upper lateral support. This growth is generally permitted by the use of jaw-type supports to engage the pressurizer lugs in the tangential, and sometimes radial, directions. The supports have small gaps in the cold condition, and these gaps are shimmed to nominal zero clearance in the hot condition. These jaw-type supports permit unrestrained vertical movement of the pressurizer lugs.

Request for Additional Information

15. Pages 8 and 10 indicate that some reactor vessel supports are cooled by water. Discuss the potential of support corrosion from this cooling water and associated aging management program.

Response

The reactor vessel supports are cooled by circulating water through or past bearing plates under the shoe. If corrosion damage is present, it would likely manifest itself as a leak prior to any significant weakening of the structural members of the support. Inspection will be visual with action taken following the management process described for inaccessible areas. See the response for RAI #13.

Request for Additional Information

16. Page 11 indicates that there are compression bumpers in steam generator support "configuration 3." Describe applicable aging effects associated with these compression bumpers and how they would be managed for renewal.

Response

The compression bumpers are structural steel components made of plates and W shapes. The applicable aging effects will be no different than those identified for other structural steel components that make up the supports.

Request for Additional Information

17. Pages 14 (Section 3.1.2.5) through 16 contain extensive discussion relating to aging management for snubbers. Pursuant to 10 CFR 54.21(a)(1)(i), snubbers are excluded from an integrated plant assessment (IPA) for renewal. Discussion relating to snubbers should be removed or minimized.

Response

It is agreed that snubbers are excluded from integrated plant assessment for renewal since they perform the intended function in an active function. Therefore, discussion relating to snubbers in the report have been removed.

Request for Additional Information

18. Page 16 states, "Spring supports and snubbers perform an active function." Although 10 CFR 54.21(a)(1)(i) excludes snubbers from an IPA, spring supports should require an IPA because they have no moving parts. Compression of the spring merely allows for component movement during heatup and cool down without changing the spring basic configuration, that is, the spring supports weight whether in the cold or hot condition with only marginal changes in loading. Provide an assessment of aging management of spring supports for renewal.

Response

Spring supports are included in the generic topical report. Note that no new age-related degradation effects will apply to spring supports other than those already described in the report. Reference is made to Examination Category F-A visual (VT-3) inspection requirements and unacceptable conditions, such as: improper hot or cold settings of spring supports; and deformation or structural degradation of springs.

Request for Additional Information

19. Page 16 states, "All supports ... are subjected to period visual examination (VT-3) through a sampling basis (25%) ..." However, the "25%" criterion is not in the latest ASME Section XI edition endorsed in 10 CFR 50.55a, that is, the 1989 edition of Section XI. Revise the report to be consistent with 10 CFR 50.55a.

Response

Regulatory Guide 1.147, Rev. 11, October 1994 endorses ASME Code Case N-491, March 14, 1991. The 25-percent sampling criterion is given in this code case. This is reflected in Section 4.2 of the GTR.

Request for Additional Information

20. Page 16 (Section 3.1.3) indicates that for RCS supports welding filler material and fluxes used conform to ASME Section III. Provide similar information on welding for RCS supports designed to the American Institute of Steel Construction (AISC) specifications.

Response

For supports designed to the American Institute of Steel Construction (AISC) Specification, the standards of the American Welding Society (AWS D1.0) are used to define the welding requirements. This has been added to the generic technical report (Subsection 2.4.1).

Request for Additional Information

21. Pages 33 and 40 discuss neutron embrittlement. Generic Safety Issue (GSI) 15, "Radiation Effects on Reactor Vessel Supports," addresses this issue. The NRC staff made a presentation to the Advisory Committee on Reactor Safeguards on July 7, 1994. The final resolution of GSI-15 should be available shortly. If GSI-15 is resolved prior to the completion of the review of this topical report, the resolution can be incorporated into the topical report. If GSI-15 is not resolved by the time of completion of the review of this topical report, the WOG has several options as follows:

    (a) The WOG could "submit a technical rationale which demonstrates that the CLB will be maintained until some later point in time in the period of extended operation, at which point one or more reasonable options ... would be available to adequately manage the effects of aging. (An applicant would have to describe its basis for concluding that the CLB is maintained, in the license renewal application, and briefly describe options that are technically feasible during the period of extended operation to manage the effects of aging, but would not have to preselect which option would be used.)" [60 FR 22484]

    (b) The WOG could provide a plant-specific resolution of GSI-15.

Additional Information for #21 (Cont.)

(c) Because it is likely that GSI-15 will be resolved before a WOG plant submits a renewal application, the WOG could commit to incorporating the resolution of GSI-15 when available. Should GSI-15 remain unresolved at the time an applicant references the topical report in its renewal application, the WOG could indicate that the applicant would have to provide item (a) or (b) above.

Indicate WOG preference.

Response

Since the report providing final resolution of GSI-15 is not available with the issuance of this topical report, three options are available for the utility to resolve this issue:

- The utility could commit to incorporating the resolution of GSI-15 at the time of the license renewal,
- A utility may choose to commit to their own program, or
- The utility could reference a generic topical report that demonstrates neutron embrittlement does not cause detrimental aging effects and there is no need for the identification of aging management options.

Request for Additional Information

22. Page 33 (Section 3.2.3) states, "Radiation data is considered in the design to determine the material properties necessary to assure that there is no unacceptable degradation of material properties (e.g., unacceptable reduction of fracture toughness) due to irradiation over the life of the plant." Provide additional information on the RCS supports design relating to fracture toughness, for example, specific radiation data, design criteria, and evaluation procedures.

Response

The referenced statements are being misinterpreted. Radiation data were not considered in the design. However, radiation data are considered in the resolution of industry issues associated with irradiation and fracture toughness. This has been clarified in the report.

Request for Additional Information

23. Pages 33 and 40 discuss neutron embrittlement in relation to high energy flux (>1 MeV). However, at low temperatures applicable to RCS supports, the low energy flux is more significant towards contributing to neutron embrittlement. Provide discussion considering the full energy spectrum evaluated in units of "displacements per atoms."

Response

An evaluation of the full energy spectrum evaluated in units of "displacements per atom" has been initiated. See also response to RAI#21 providing steps to be taken for resolution.

Request for Additional Information

24. Page 34 (Section 3.3) states, "There are three industry open issues however, the executive summary on page iii states, "... there are five issues ..." Resolve this inconsistency.

Response

To resolve the inconsistencies, a modification is made as follows (Section 3.1):

> The aging management issues in the nuclear industry have been compiled in an EPRI report [Ref. 1 of WCAP-14422]. Two aging effect issues relevant to the RCS supports were identified from this EPRI report: (1) aggressive chemical attack, and (2) corrosion. In addition, there are three issues that have been identified and addressed by utilities that pertain to the RCS primary component supports, which are addressed by utilities in the current licensing term: (1) low fracture toughness and lamellar tearing [USI A-12 and NUREG-0577]; (2) stress corrosion cracking (SCC) of high-strength bolting materials; and (3) neutron embrittlement of reactor vessel supports by low-temperature, low-fluence irradiation.

Request for Additional Information

25. Page 34 (Section 3.3) mentions the issue of "low fracture toughness and lamellar tearing." However, there is no further discussion relating to the subject in the report. This subject appears to be construction related and not aging related. The resolution of the associated Unresolved Safety Issue (USI) A-12 should continue, along with the rest of the CLB, into the renewal term. Clarify the WOG intent relating to this subject for renewal.

Response

This subject is construction-related; however, a utility may have new construction within the life of the plant. It is the WOG intent that the resolution of the associated Unresolved Safety Issue (USI) A-12 would continue, along with the rest of the CLB, into the renewal term. If a utility wants to deviate from the established A-12 resolution during the renewal term, they would be obligated to make the deviation part of their plant-specific license renewal application. This discussion has been included in Section 4.0.

Request for Additional Information

26. In Section 4.0, the report should describe and justify how the effects of aging will be managed so that the intended functions of the RCS supports are maintained for the extended period of operation, as required by 54.21(a) (3).

Response

Section 3.0 provides the results of the evaluation of potential significance of age-related degradation effects on particular RCS support structures. Section 4.0 provides the results of the evaluation of programs to manage the potentially significant effects of these age-related degradation mechanisms on particular structures. The program elements are described to a level of detail that includes the frequency and coverage of inspection, the type of inspection, the acceptance criteria for the inspections, and any other pertinent information necessary to establish the finding. Attribute tables giving this level of detail are provided in Section 4.0.

Request for Additional Information

27. Section 4.0 (beginning on page 35) evaluates aging mechanisms. Discuss the potential of bolting becoming loose during extended service, that is, loss of preload. Also, discuss whether Section XI Subsection IWF inspections can detect loose bolting if Subsection IWF is relied on to manage this aging effect for renewal.

Response

One of the unacceptable conditions for the visual (VT-3) examination of supports, in accordance with Examination Category F-A of Section XI (IWF-3410, "Acceptance Standards"), is "missing, detached, or loosened support items." This unacceptable condition must be corrected before continuation of service. This is not meant to imply that visual inspection can detect loss of preload, only that loss of preload sufficient to cause bolts or fasteners to become loose can be detected.

Request for Additional Information

28. Page 39 (Section 4.2.2) on boric acid corrosion states, "... no further evaluation is required beyond assuring the absence of reactor coolant leaks." Borated water leaks have been found in a number of pressurized water reactor containments and they have accumulated on containment basemats and in reactor cavities. Many of the RCS supports are at the lowest level of the containment where they may be exposed to wetted surfaces. Describe the program for providing assurance of absence of reactor coolant leaks, especially for reactor cavity seal leakage during refueling. Also, because it may be difficult to completely eliminate the presence of boric acid, discuss management activities for renewal should leakage occur.

Response to #28 (Cont.)

Programs implemented to provide assurance of absence of the reactor coolant leaks are outside the scope of this generic technical report. It is recognized that boric acid is an aggressive corrosive chemical whose potential presence must be managed during the current license renewal period as well as for the renewal period. Boric acid corrosion resulting from RCS leaks can affect the reactor coolant loop system supports. Utilities have developed and implemented plans to address corrosive effects of reactor coolant system leakage per Generic Letter 88-05 requirements. Periodic inservice inspections, in accordance with the requirements of the ASME Code Section XI, Subsection IWF, plus any utility commitments in their CLB in response to Generic Letter 88-05, are capable of managing these effects in the license renewal term.

Boric acid corrosion management programs have been developed in response to Generic Letter 88-05 requirements. Such a program has been included in the report and consists of the following:

- Inspect for presence of boric acid crystals via visual examination of the components
- If boric acid is found, gather information for engineering evaluation and then clean
- If a significant amount, as defined by the presence of crystal buildup, is present then:
  - Remove crystals and clean surface
  - Take measurements for evaluation of degradation
  - If functional integrity limits are exceeded due to corrosive deterioration, make necessary repairs
  - Identify potential path of leak, identify source (if possible) and fix the leak. If source cannot be found or quickly repaired, it may be necessary to place detectors in critical areas to monitor buildup and take appropriate actions in a timely manner.
- If there are only traces of boric acid, which are defined as light tracks, then only cleaning is required
- Perform evaluation of degradation and determine if refurbishment is necessary.

This discussion has been included in Subsection 4.2.1.

Request for Additional Information

29. Page 39 (Section 4.2) discusses boric acid corrosion of the RCS supports. Do aging management programs resulting from Generic Letter 88-05 on boric acid corrosion of the RCS involve activities applicable to the RCS supports? If not, what program is relied on to mitigate the potential of boric acid corrosion of the RCS supports, including associated bolting, for the period of extended operation?

Response

See response to RAI #28.

Request for Additional Information

30. Page 39 (Section 4.2) discusses corrosion. It addresses only boric acid corrosion. However, there is a potential for corrosion without boric acid. For example, water leakage from air handling units have caused corrosion. Corrosion may occur if the coating is not intact or if there is dissimilar materials (galvanic corrosion). Further evaluate loss of material due to corrosion.

Response

The discussion related to corrosion in the report is not limited to boric acid corrosion. All plausible types of possible corrosion are included. Dissimilar metal (galvanic) corrosion is not plausible for RCS support structures constructed from low-alloy or carbon steel. Corrosion from water leakage was added, in particular, for the RPV supporting cooling channels. The program elements described in Section 4.0 for managing the effects of corrosion are applicable, regardless of the agent(s) for the corrosion, and are addressed in Section 4.0.

Request for Additional Information

31. Page 41 (Section 4.4.2) discusses the significance of thermal aging of cast stainless steel to the RCS supports. Because there is no cast stainless steel used in the RCS supports as indicated on pages 16 and 47, this discussion appears unnecessary and potentially confusing.

    The discussion becomes more confusing when it continues to state: (i) "... such as for bolts, thermal aging is addressed by the NRC program [7]" and (ii) "... through the application of the leak-before-break methodology ..." (i) Reference [7] addresses USI A-12, "Steam Generator and Reactor Coolant Pump Supports," which does not include bolts or thermal aging. Are there cast stainless steel bolts on the RCS supports? (ii) If leak-before-break is to be applied to reduce dynamic loads and leak-before-break is not already part of the CLB, a CLB change request must be submitted for NRC approval.

    Revise thermal aging discussion.

Response

There are no austenitic cast stainless steel used in the supports that are within the scope of this report. Therefore, thermal aging embrittlement is not applicable. The report reflects this.

Request for Additional Information

32. Page 41 (Sections 4.4.1 and 4.4.2) indicates that thermal embrittlement is applicable at temperatures above 600EF. However, research at the Argonne National Laboratory shows that embrittlement can occur at service temperatures as low as 540EF. (See NUREG/CR-4513, Rev. 1) Justify the higher value in the report.

Response

See the response given for RAI #31.

Request for Additional Information

33. Page 43 states, "Fatigue crack growth is used for evaluating structural parts for which a crack is found or postulated." This statement implies that a fatigue crack growth analysis is associated with the design or inspection of structures. Identify specific RCS supports that have been evaluated using such a methodology.

Response

The following will be added to the GTR:

> Fatigue crack growth analysis is not specifically required as a part of the periodic inservice inspection program to manage the potentially significant effects of fatigue on RCS support structures. If supplemental inservice examinations are carried out in accordance with Paragraph IWF-3200 to characterize a defect or flaw found as the result of an unacceptable visual inspection condition, Paragraph IWF-3122.3 permits that defect or flaw to be accepted by analysis, which would include standard procedures for fatigue crack growth analysis contained elsewhere in ASME Code Section XI.
>
> Fatigue crack growth analysis is not performed for design or inspection of structures. This type of analysis would be performed only when warranted by special cases where fatigue crack growth would be suspected. For the supports within the scope of this report, this type analysis is not necessary. Situations where this could occur would be in regions of integral attachments, lugs, nozzles, or welds on a component shell. These areas are not in the scope of this report, but addressed in specific component reports.

Request for Additional Information

34. Page 44 shows fatigue "cumulative usage factors (CUFs)" for RCS supports. Clarify the fatigue design requirements, such as the code of record, for the RCS supports.

Response

The design fatigue requirements for the supports follow ASME Section III, Subsection NF, or for the older plants, with a contract date prior to 1974, American Institute of Steel Construction specification. A statement has been added stating the basis (ASME Section III, Subsection NF) of the usage factors given. Note that the fatigue usage factors are given to show that fatigue is nonsignificant (usage factor < 0.15 for 60 years of operation and much lower than 20,000 cycles).

Request for Additional Information

35. Table 4-1 on page 44 shows a fatigue CUF estimate. Provide additional information relating to the evaluation, such as specific support evaluated, severity of fatigue service compared with other RCS supports, and load pairings associated with each plant operating condition. Also, provide an evaluation of emergency and faulted operating conditions on CUF because they are not included in Table 4-1.

Response

It is not necessary to provide the information requested since fatigue is not significant. As noted in the report and in response to RAI #34, the fatigue usage factor is less than 0.15 for 60 years of operation and experiences much less than 20,000 cycles. The estimate of fatigue cumulative usage is provided to show that fatigue should not be considered an aging effect for these supports. Where fatigue is potentially significant are in the regions of the component boundaries (i.e., integral attachments, weld interface between support and component, lugs, etc). These areas are not within the scope of this general technical report (see response to RAI #4). Fatigue in these regions is discussed, if significant, in the component GTRs.

Request for Additional Information

36. Page 46 (Section 4.8.1.3) indicates that exposure to 300EF or higher temperature could cause concrete to deteriorate. However, during previous discussions with the Nuclear Management and Resources Council (NUMARC) on the license renewal Class 1 structures, pressurized water reactor containment, and boiling water reactor industry reports (IRs), the corresponding values are operating temperatures of 150EF and local area temperature of 200EF. For temperatures above these values, the effects of elevated temperature has to be evaluated, including potential synergistic effects of high temperature and irradiation. Justify the 300EF value in the WOG topical report. Also, describe the concrete temperature with cooling as discussed in the report.

Response

The 300EF concrete exposure temperature discussed in the topical report is the temperature at which the concrete begins to deteriorate with surface scaling and cracking becoming physically visible.

Response for #36 (Cont.)

It is agreed that concrete operating temperature should not exceed 150EF, and local area temperatures should be kept below 200EF. Reactor vessel supports could be subjected to high temperatures that could potentially result in local temperatures above 200EF if supplemental cooling is not provided. For those support configurations where the local temperature at the concrete surfaces could exceed 200EF, special design features are incorporated based on air or water cooling to keep the local temperature below 200EF. Subsection 3.2.8(c) has been modified to incorporate this added discussion so that there is no misunderstanding that a 300EF temperature is considered acceptable for concrete.

Request for Additional Information

37. Page 47 (Section 4.9.1) states only the following bolts are to be reviewed for stress corrosion cracking (SCC): "Steam generator supports and their anchor bolting, the anchor bolting of the reactor coolant pumps and of the pressurizer support skirt, and pressurizer support bolting." Are there other bolts on the RCS supports within the scope of the report that are not on this list? Is so, justify why these other bolts need not be managed for SCC.

Response

The current inservice inspection program, Examination Category F-A of the ASME Code Section XI, Subsection IWF, should be found to be adequate to manage the effects of stress corrosion cracking (SCC) for all support bolting, not just steam generator supports and their anchor bolting, reactor coolant pump support anchor bolting, pressurizer support skirt anchor bolting, and pressurizer support bolting.

The possibility of other bolts experiencing SCC is recognized in the report. A management process is given in the report (Section 4.0) that is based on the utilities following EPRI guidelines in EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants." However, a utility may use another practice that would be described in their application for renewal.

Request for Additional Information

38. Page 47 (Section 4.9.1) discusses a specific program for aging management of bolting for renewal. However, licensees have existing commitments in response to Bulletin 82-02 on managing bolting degradation that are part of the CLB. (Bulletin 82-02 forms part of the basis for resolving GSI-29.) Clarify whether aging management programs in licensee responses to Bulletin 82-02 are commitments for license renewal.

Response

The current utility commitments in response to Information Bulletin 82-02 are adequate to manage aging; therefore, a utility may extend their existing commitments in response to Bulletin

Response to #38 (Cont.)

82-02 into the extended plant term. If a utility decides to modify this commitment, a utility should address this in the plant-specific license renewal application. Note that Bulletin 82-02 does not apply to RCS support bolts. This is discussed in Section 4.1 of the report.

Request for Additional Information

39. Page 47 (Section 4.9.2) discusses the management of SCC for bolting. However, for those support configurations that are bolted to concrete, it is not apparent how these bolted connections are assured of meeting their intended function considering the potential for degradation from vibration and other environmental conditions. Discuss the applicability of programs for renewal that ensure anchor bolt integrity, such as torque check of bolts and stud nuts or ultrasonic examinations of bolts and studs.

Response

The fracture or looseness of support bolting are both unacceptable conditions for continued operation and require corrective action under Examination Category F-A of Subsection IWF of the ASME Code Section XI.

See also response to RAI #27.

Request for Additional Information

40. Page 48 (Section 4.9.2) describes "Type (1)" analysis as requiring loss-of-coolant accident (LOCA) loads. Clarify that the analysis should include all loads consistent with the CLB, such as LOCA and seismic loads.

Response

The Type (1) analysis has been eliminated. See Response to RAI #41.

The current inservice inspection programs are adequate to manage such effects, and no reanalysis program is needed. If an evaluation is needed as the result of an unacceptable condition found from the inspection, the ASME Code Section XI provides the necessary guidance.

Request for Additional Information

41. Page 48 (Section 4.9.2) describes "Type (2)" analysis for bolting based on low stresses to preclude SCC. The staff disagrees with the validity of the WOG argument because there is always high stresses at the root of some bolt threads that can be the site of SCC. WOG should remove "Type (2)" as an option.

Response

Both the Type (1) and Type (2) option will be removed from the generic technical report. The procedure given in EPRI NP-5769 may be followed. See response to RAI #37.

Request for Additional Information

42. Page 48 (Section 4.9.2) describes certain analyses and states, "This type of analysis is plant specific ..." WOG should indicate that specific analyses are subject to NRC review and approval.

Response

Type (1) and (2) analyses will be removed, and no specific analyses are required. See also response given for RAI #41.

Request for Additional Information

43. Page 48 (Section 4.9.3) states, "As part of the ASME Section XI inspection requirements, the concrete in contact with the primary component supports is visually inspected." Page 3 states, "... ASME Code Section XI procedures ... cover both the support and the adjacent concrete." However, the ASME Section XI, Subsection IWF code boundary is the surface of the building structure. The staff believes that this boundary does not include the surface of the building structure such as adjacent concrete. Provide additional information on the implementation of Section XI procedures at WOG member plants. Also, describe the specific inspection program for adjacent concrete for renewal.

Response

Concrete that is adjacent to the support embedments is not within the scope of this generic technical report. This portion is included in the generic report associated with seismic Class 1 structures (see response to RAI #4). See also the response to RAI #1 for additional information on the implementation of Section XI procedures.

Request for Additional Information

44. Page 48 (Section 4.9.3) discusses aging management of concrete. Are any RCS supports within the scope of the report supported by masonry walls? If yes, WOG should address the management of cracking of masonry walls. For example, during previous discussions with NUMARC on its Class 1 structures IR, NUMARC indicated that licensee responses to Bulletin 80-11 and licensee programs to address Information Notice 87-67 are applicable for managing cracking of masonry walls.

Response

The supports that are part of this GTR are not attached to any masonry walls.

45. Page 48 (Section 4.9.3) states, "Given in the sections below, for information purposes, are techniques used for concrete degradation monitoring." The referenced sections are presumably Sections 4.9.3.1 and 4.9.3.2. Clarify the intent of "for information purposes." The report needs to identify specific programs that will be relied on during the period of extended operation. Further, the report needs to provide details of the programs for staff review. Describe and justify specific programs to be committed to for renewal to manage aging of concrete adjacent to the RCS supports for staff review.

Response

The referenced section was given for information purposes for utility personal related to concrete monitoring and repair. This section has been removed. In Section 4.0, information is given with direct references to management programs that are needed to monitor concrete degradation associated with the local areas around support embedments that follow American Concrete Institute (ACI) documents.

Request for Additional Information

46. A paragraph on page 49 (Section 4.9.3,1) states: "Deterioration of the pre-stressing system includes loss of prestress in tendons and cracks in wires and anchor blocks ..." This paragraph addresses prestressing concrete containment tendons and appears not to be applicable to RCS supports within the scope of the topical report. However, there are a few prestressed concrete reactor cavity walls, such as in ice condenser plants where the prestressing tendons are grouted and Regulatory Guide 1.35 referenced in this paragraph is not applicable. Clarify intent of this paragraph. If these prestressed concrete reactor cavity walls are within the scope of the topical report, discuss aging management programs for renewal.

Response

No supports within the scope of this report have prestress structures. Therefore, the discussion related to the deterioration of the prestressing system has been removed from the report.

Request for Additional Information

47. Page 49 (Section 4.9.3.1) indicates that a part of the water test criteria to assure non-corrosive environment is a pH value of greater than 5.5. However, page 46 states, "Rebars will be significantly corroded when the pH value of the concrete is reduced to 11 or less." Also, during previous discussion with NUMARC on the Class 1 structures IR, aggressive chemical attack of concrete is non-significant if the pH is greater than 5.5, but corrosion of embedded steel is non-significant if the pH is greater than 11.5. Thus, the pH should be greater than 11.5 to mitigate aggressive chemical attack and corrosion of

Additional Information for #47 (Cont.)

embedded steel. Discuss how corrosion of embedded steel is managed for the period of extended operation if the pH value of only 5.5 is used.

Response

Based on the NUMARC Class 1 structures industry report:

- Aggressive chemicals (<5.5 pH) or chloride or sulfate solutions beyond defined limits (500 ppm chlorides and 1500 ppm sulfates) can result in significant chemical attack if concrete is exposed for extended periods.

- Concrete exposed to above environment for intermittent periods only will not have significant degradation.

For the potential of corrosion to be significant, concrete must be exposed to the corrosive environment for extended periods of time. Cracks must occur to allow aggressive chemicals to reach the reinforcement. The use of ACI 318 and 349 design standards result in dense, well-cured concrete with low permeability with proper reinforcement. An acceptable management program is given in Subsection 4.2.1 of the report based on: removing collected water that contains aggressive chemicals; repair cracks.

Request for Additional Information

48. Page 50 (Section 4.10) evaluates time-limited aging analyses (TLAAs) and states, "No time-limited analyses have been identified as part of the original design process for the RCS supports. However, since aging progresses with time, all structural components that are subjected to age-related degradation can be considered time-limited." Based on the discussion in the report, it appears that WOG's evaluation is not consistent with the definition of TLAA in 10 CFR 54.3. For example, the staff found a potential TLAA for RCS supports: RCS supports designed in accordance with ASME Section III Subsection NF or the AISC specification for the Design, Fabrication and Erection of Structural Steel for Buildings contain a fatigue design requirement described in Subarticle NF-3330 Subsection NF and Appendix B of AISC. Both Subsection NF and AISC provide for fatigue design based on allowable stress range reductions for increasing cycles exceeding 20,000 cycles. Reevaluate applicable TLMs for RCS supports in accordance with 10 CFR 54.3.

Response

It is agreed that both Subsection NF and AISC provide for fatigue design based on allowable stress range reductions for increasing cycles exceeding 20,000 cycles. However, fatigue is not part of design qualification analyses for the component supports within the scope of this report since they are not subject to high fatigue usage factors or a significant number of stress cycles. High fatigue usage factors may occur in the related component scope, for example, reactor

Response to #48 (Cont.)

pressure vessel, steam generator, reactor coolant pump, and pressurizer (see response to RAI #4). The referenced statement made will be revised as follows and placed in Subsection 2.5 of the report.

> TLAAs have been identified as part of the original design process for the supports within the scope of this report. Both ASME Section III, Subsection NF and AISC provide for fatigue design based on allowable stress range reductions when exceeding 20,000 cycles. However, fatigue is not part of design qualification analyses for the component supports within the scope of this report since they are not subject to high fatigue usage factors and significant stress cycles in excess of 20,000. In Subsection 3.2.6 and Section 3.3 of the report fatigue and TLAAs are further discussed. It is concluded that no additional analyses are required to be performed by the utility for demonstration that TLAAs are acceptable for the extended period of operation since all required demonstration analyses are contained in the report (Subsection 3.2.6 and Section 3.3).

See also the responses given for RAIs 34 and 35.

Request for Additional Information

49. Page 50 (Section 4.10) discusses TLAAs. There may be cases where the plant owner had elected to design a RCS support, such as for some reactor vessel skirt supports, to ASME Section III Subsection NB requirements during construction. In those situations, the fatigue CUF in Subsection NB is a TLAA. Identify WOG member plant RCS supports that have been evaluated to Subsection NB requirements as part of the CLB.

Response

No cases for the RCS supports have been designed to ASME Section III, Subsection NB requirements during construction.

Request for Additional Information

50. Reference 7 on Page 51 (Section 5.0) is NUREG-0577 which was issued for public comment on the resolution of USI A-12. Subsequently, NUREG-0577, Rev. 1, was issued on October 1983 to resolve USI A-12. WOG should also reference the latter document.

    Further, Reference 13 addresses the Shippingport shield tank. WOG may wish to also reference NUREG/CR-5748, "Radiation Embrittlement of the Neutron Shield Tank from the Shippingport Reactor," dated October 1991, on the same subject.

Response

NUREG-0577, Rev. 1 is referenced. NUREG/CR-5748 has not been used since radiation is treated as a plant-specific evaluation to be submitted during the utility license application.

Request for Additional Information

51. Consider the following editorial comments:

    (a) Page iii uses the term "age related deterioration." To be consistent with the rest of the report, it should be "age-related degradation." (See heading of Section 4.0)

    (b) Page 15 (Section 3.1.2.6) uses the term "shock arresters." Because the rest of the report uses the term "snubbers," reword to be consistent, as applicable.

    (c) Page 2 states, "... can eliminate stress corrosion cracking. "Preferred wording is "... can essentially eliminate ..."

    (d) Page 34 states, "... brittle fracture ... there is little or no plastic deformation at the crack tip ..." Preferred wording is "... the plasticity at the crack tip is negligible ..." because there must be at least some plasticity in the immediate vicinity of the crack tip.

    (e) Page 49, "Windsor Prob" should be "Windsor Probe."

Response

The suggested editorial comments have been incorporated into the report as appropriate.

8.0 USE AND APPLICATION OF A GENERIC TECHNICAL REPORT IN A LICENSE RENEWAL APPLICATION

The primary intended purpose of the generic technical report (GTR) is to be used as a reference in the preparation of a license renewal application. This section describes that process, illustrated in Figure 8-1, with appropriate references to the Nuclear Energy Institute (NEI) 95-10, *Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule*, Revision 0, which provides the process and documentation format for a utility license renewal application.

Although a utility can perform an aging management review (AMR) that is plant-specific, using generically approved AMRs (e.g., aging management guidelines [AMGs], industry reports [IRs], and GTRs) can significantly reduce the efforts of the licensee and NRC in the preparation and approval of a renewed license. To achieve the most benefit of a GTR, it should be reviewed to identify how much of the plant-specific structure or component (SC) is bounded by the GTR, and which generic evaluations apply to the SC for that plant. This review would limit the plant-specific AMR to only those SCs at the plant that are not bounded by the GTR, or have not been evaluated generically.

The primary elements of using a generic AMR as a license renewal application reference are to identify those AMRs that apply and demonstrate how the generic AMR is applicable to the plant, as well as demonstrate that the aging effects will be managed.

8.1 IDENTIFY AND DEMONSTRATE APPLICABILITY

8.1.1 Step 1 - Determine if Report Has Been Reviewed and Approved

The first step in using a GTR in a license renewal application (see Figure 8-1) is to identify the GTR to be referenced. A list of reports that are used by applicant utilities for license renewal, which will be found in the Public Document Room at the NRC, is in Exhibit A of the license renewal application, Section 1-1, "Scope." This section of Exhibit A will identify which reports filed by utilities have and have not been approved by the NRC.

8.1.2 Step 2 - Identify and Compare Report Characteristics

Demonstrate how a GTR is applicable to the plant that is applying for a renewed license. This can be accomplished by completing four activities.

- Identify those characteristics that affect the conclusions of the GTR, such as:
  - Scope
  - Assumptions
  - Limitations
  - Configuration
  - Functions
  - Engineering and design parameters

- Protective measures
- Materials
- Fabrication
- Service conditions

- Compare the approved characteristics in the GTR to plant-specific characteristics.

- Identify plant characteristics that are bounded by approved characteristics in the GTR.

- Identify plant characteristics that are not bounded by approved characteristics in the GTR.

Comparing the approved characteristics to the plant-specific characteristics helps determine which plant characteristics are equivalent to, or bounded by the GTR. Those plant characteristics that are not equivalent or bounded by the GTR should be identified and evaluated in the plant-specific license renewal application. This would be documented in Sections 2.2 and 2.3, "SC Selection Process" and "Scoping Results," of Exhibit A of a license renewal application.

8.2 DEMONSTRATE THAT AGING EFFECTS WILL BE MANAGED

This demonstration requires six activities:

- Compare the approved time-limited aging analyses (TLAAs) in the generic AMRs with those identified from a review of the current licensing basis (CLB) in effect at the plant.

- Verify that plant TLAA characteristics are bounded by the generic AMRs.

- Compare the list of approved aging effects in the generic AMRs with those from a review of commitments that have changed the original CLB, and are based on the effects of aging, such as:

  - Responses to NRC communications: bulletins, generic letters, or enforcement actions

  - License event reports (LERs) and safety evaluation reports (SERs)

  - Safety analysis report (SAR) amendments and technical specification changes

- Compare approved program features in the AMRs with similar plant program features.

- Identify similar program features.

- Identify program features that are different from those approved in the generic AMRs.

8.2.1 Step 3 - Review Aging Effects Based on Plant Operating and Maintenance History

Compare TLAAs and aging effects from a GTR with those from a review of CLB changes in effect at a plant.

The TLAA comparison identifies:

- TLAAs that are applicable to the plant
- TLAAs that are not applicable to the plant
- Additional plant-specific TLAAs

The above three categories of TLAAs should be identified in Section 1.3, "TLAA Evaluation," in Exhibit A of the license renewal application. For TLAAs that are not applicable to the plant, a justification should be provided explaining why that TLAA does not apply. Additional plant-specific TLAAs that are identified require an evaluation (10 CFR 54.21 [c][1][i - iii]) and aging effect evaluation, as necessary.

The aging effect comparison identifies:

- Aging effects that are applicable to the plant
- Aging effects that are not applicable to the plant
- Additional plant-specific aging effects

Exhibit A of the license renewal application, Section 3.2, "Aging Management Review Process," should identify these three categories of aging effects. As with the TLAAs discussed above, a justification should be provided for those aging effects that are not applicable to the plant. Additional plant-specific aging effects that are identified require an evaluation and aging management program, as necessary, in the plant-specific license renewal application.

8.2.2 Step 4 - Compare Referenced Program Features

The comparison of program features from generic AMRs with plant programs identifies equivalent program features and those plant program features that are different. This comparison should be documented in Exhibit A of the license renewal application, Section 3.2, "Aging Management Review Process." For plant programs that differ from approved program features in generic AMRs, two options are available:

- Provide a justification explaining why the plant program is adequate for managing the aging effect, or

- Describe an enhancement to a plant program or a new plant program that is consistent with the program features approved in the GTR.

8.2.3 Step 5 - Identify Enhancements or New Programs

A description of a new program or program enhancement should include a demonstration of the enhanced or additional features. This demonstration should explain how the program features manage the aging effect to maintain an intended function consistent with the CLB, and why these features will be adequate for an extended period of operation.

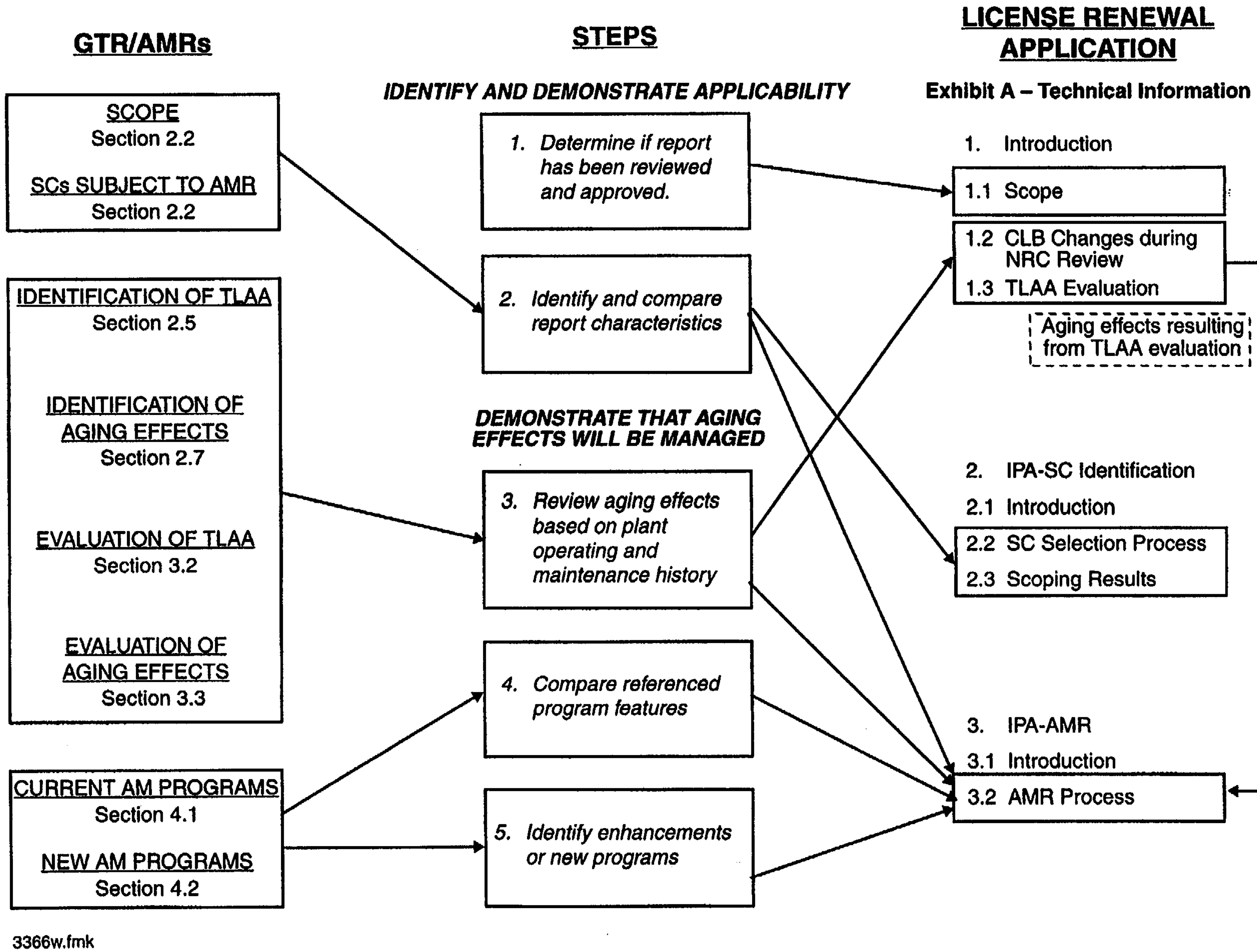


Figure 8-1 How to Use a GTR in a License Renewal Application

9.0 GENERIC SAFETY ISSUE-15: RADIATION EFFECTS ON REACTOR PRESSURE VESSEL SUPPORTS

9.1 OBJECTIVE

To address the concern that low-temperature, low-energy neutron irradiation may embrittle reactor pressure vessel (RPV) supports more rapidly than expected.

9.2 BACKGROUND

GSI-15 started out as a low-priority safety issue at the beginning of the study. The priority of the issue was changed after Oak Ridge National Laboratory (ORNL) reported (NUREG/CR-5320) unexpectedly high embrittlement rates in surveillance specimens from the high-flux isotope reactor (HFIR). A task action plan was prepared to evaluate the possibility that RPV supports may be degraded and subject to failure in the event of a design basis accident. In the course of completing the task action plan, several findings emerged that contributed to the technical resolution of the issue. Limited surveys of RPV supports conducted in response to the unexpectedly high HFIR embrittlement data noted that data often were too sketchy to be definitive. The United States Nuclear Regulatory Commission (U.S. NRC) revisited and re-evaluated the data to establish the changes in transition temperatures as a function of total radiation (neutron plus gammas), and displacement per atom (dpa) as shown in Figure 9-1 [Ref. 9-1].

In May 1996, the U.S. NRC issued the report NUREG-1509, which summarizes its findings during the re-evaluation. The report also provides an engineering approach, including screening criteria and technical evaluation procedures that may be used as guidelines acceptable to the U.S. NRC. This appendix is intended to summarize the guidelines given in NUREG-1509.

9.3 SCREENING CRITERIA

The NUREG suggests that RPV supports should be screened sequentially for evaluation, as illustrated in Figure 9-2. The procedure is designed to assess support vulnerability by eliminating supports that are not affected by embrittlement because of their configuration or state of stress. The most vulnerable supports are considered to be those that are exposed to a relatively high fluence, have high initial nil-ductility transition temperature (NDTT), and have high tensile stress. Figure 9-2 from the NUREG illustrates that these elements are the essential criteria for screening the RPV supports.

NUREG-1509 states that by satisfying the following criteria, the supports should be free from radiation embrittlement, the integrity may be reasonably assured, and no further investigation should be required.

- The initial NDTT of the RPV supports is well below the minimum operating temperature
- The radiation exposure at the support is low
- The peak tensile stresses are 6 ksi or less

9.4 CRITERIA FOR RE-EVALUATION

The NUREG re-evaluation process for the RPV support is divided into several distinct steps, as illustrated by the flow chart, Figure 9-3. Figure 9-3 in turn leads to either Figure 9-4 or 9-5 if the preliminary evaluation requires further evaluation. The figures are self-explanatory. However, the explanatory notes for the Figures 9-2 through 9-5 are duplicated from the NUREG in this appendix.

9.5 REFERENCES

9.1 NUREG-1509, "Radiation Effects on Reactor Pressure Vessel Supports," U.S. NRC (May 1996).

9.2 Y. Gohar and M. A. Abdou, *MACKLIB-IV, a Library of Nuclear Response Functions Generated with the MACK-IV Computer Program from ENDF/B-IV,* ANL/FPP/TM-106.

9.3 Samuel Glasstone and Alexander Sesonske, *Nuclear Reactor Engineering,* Princeton, New Jersey: D. Van Nostrand Company, Inc., 1963.

9.4 P. L. Andersen et al., "State of Knowledge of Radiation Effects on Environmental Cracking in Light Water Reactor Core Materials," *1989 Workshop on LWR Radiation Water Chemistry and its Influence on In-Core Structural Materials, Palo Alto, CA, 14-15 Nov. 1989,* EPRI-NP-7033, Mar 1991, p. 5.29-5.66.

9.5 J. O. Stiegler and L. K. Mansur, "Radiation Effects in Structural Materials," *Annual Reviews of Materials Science, 1979, 9,* pp. 405-454.

9.6 S. Glasstone and A. Sesonske, *op. cit.,* p. 602.

9.7 G. J. Dienes and G. H. Vineyard, *Radiation Effects in Solids,* New York: Interscience Publishers, 1957.

9.8 Samuel Glasstone and Alexander Sesonske, *Nuclear Reactor Engineering,* Princeton, New Jersey: D. Van Nostrand Company, Inc., 1963, p. 93.

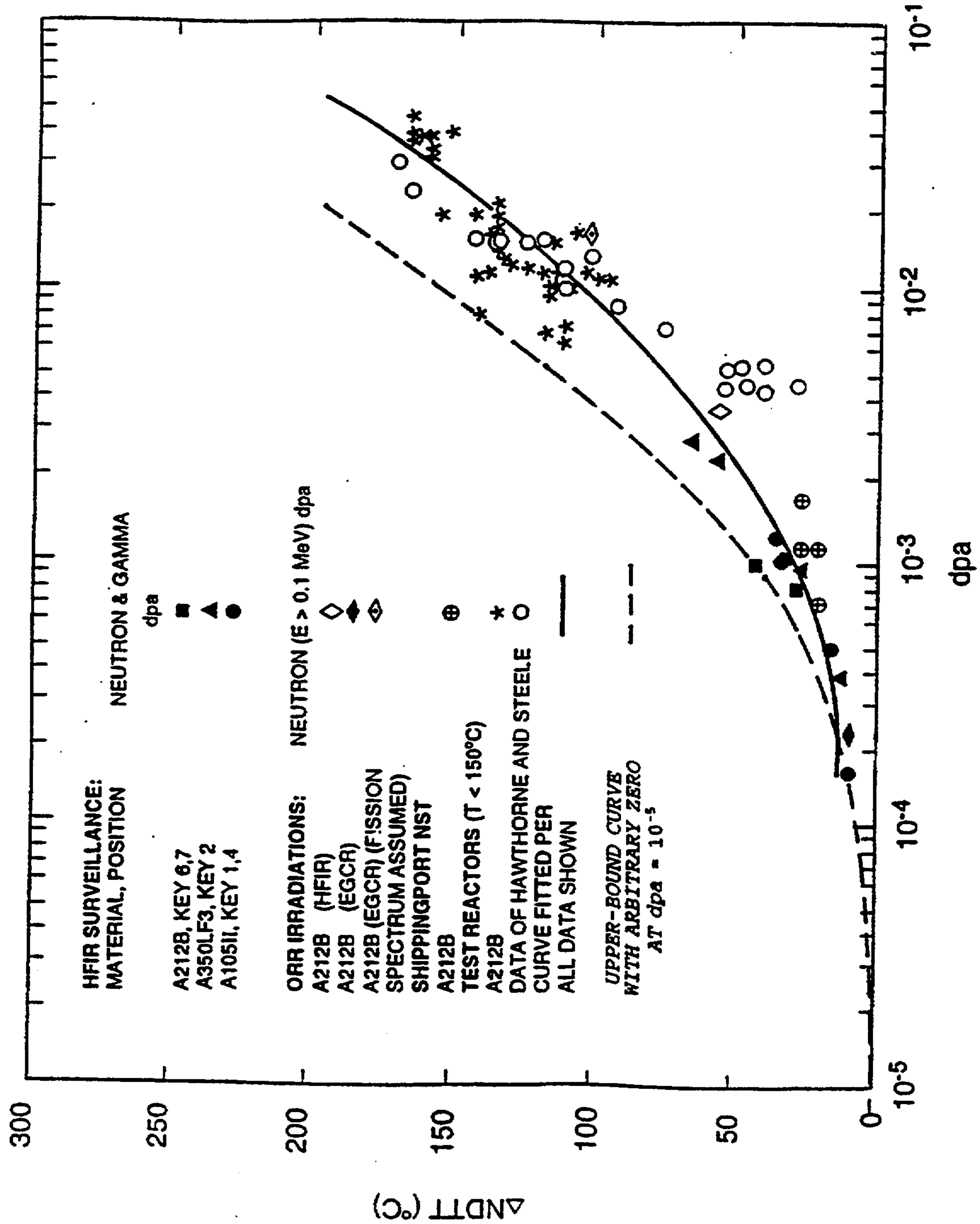
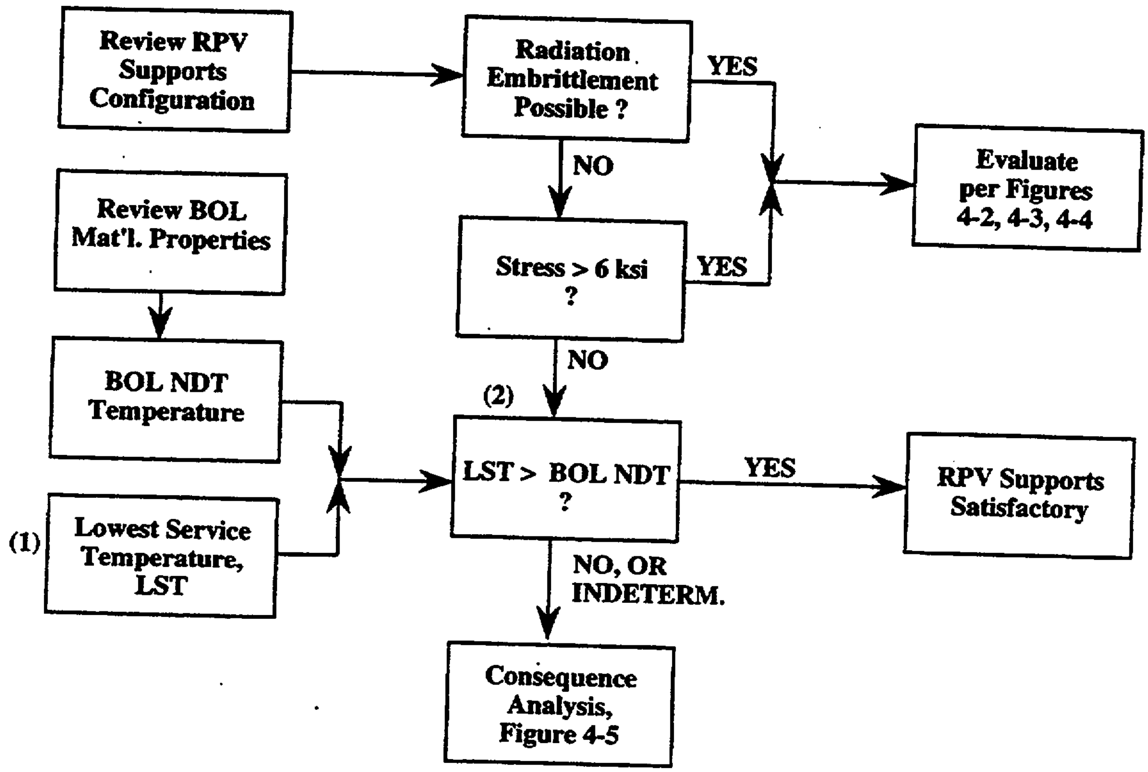


Figure 9-1 **The Change In Transition Temperature as a Function of Total Radiation (neutrons plus gammas), dpa [Ref. 9-1]**



Note: The numbers next to the blocks above refer to the corresponding paragraphs of "Explanatory Notes" at the end of this section.

Figure 9-2 Screening Criteria

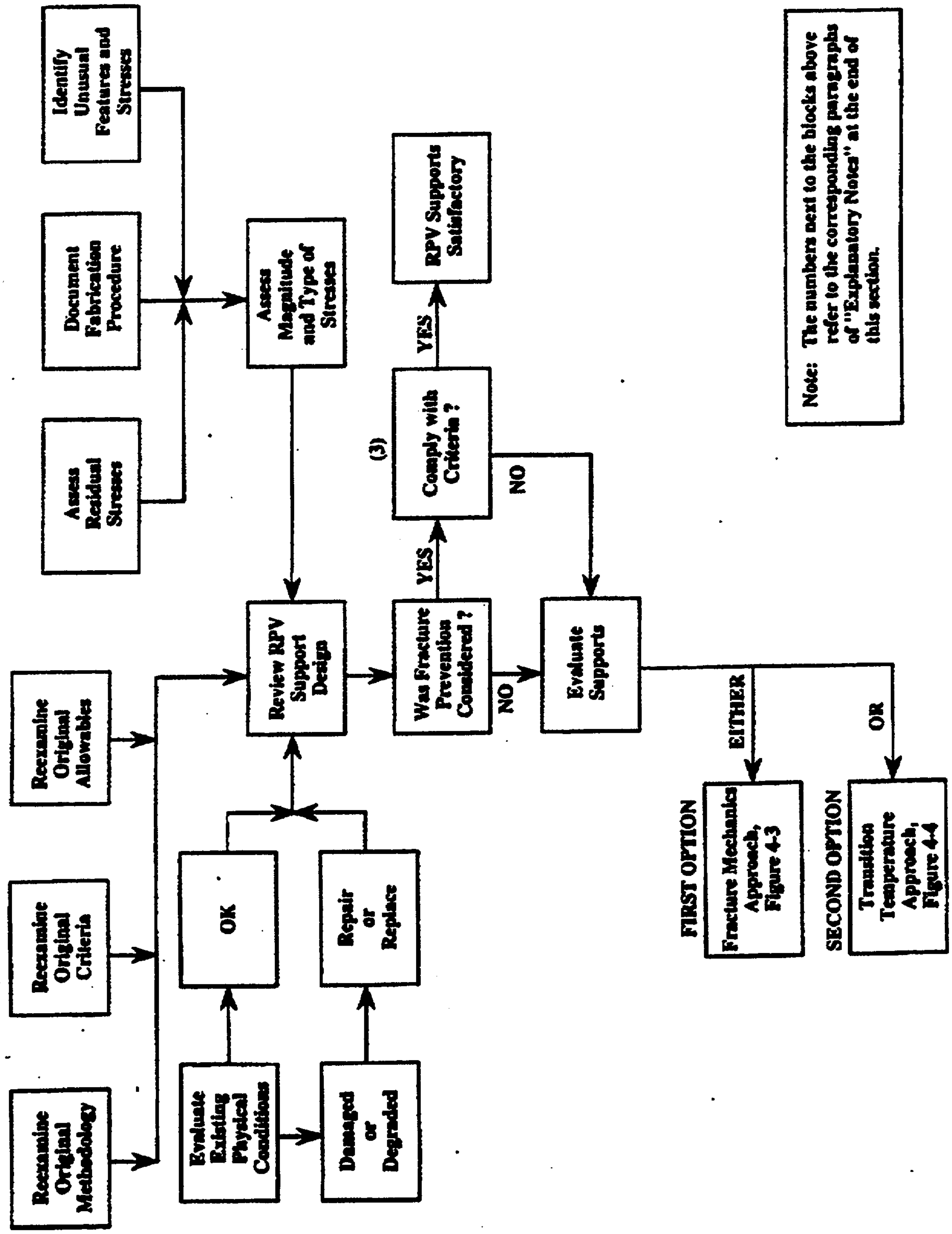


Figure 9-3 Preliminary Evaluation

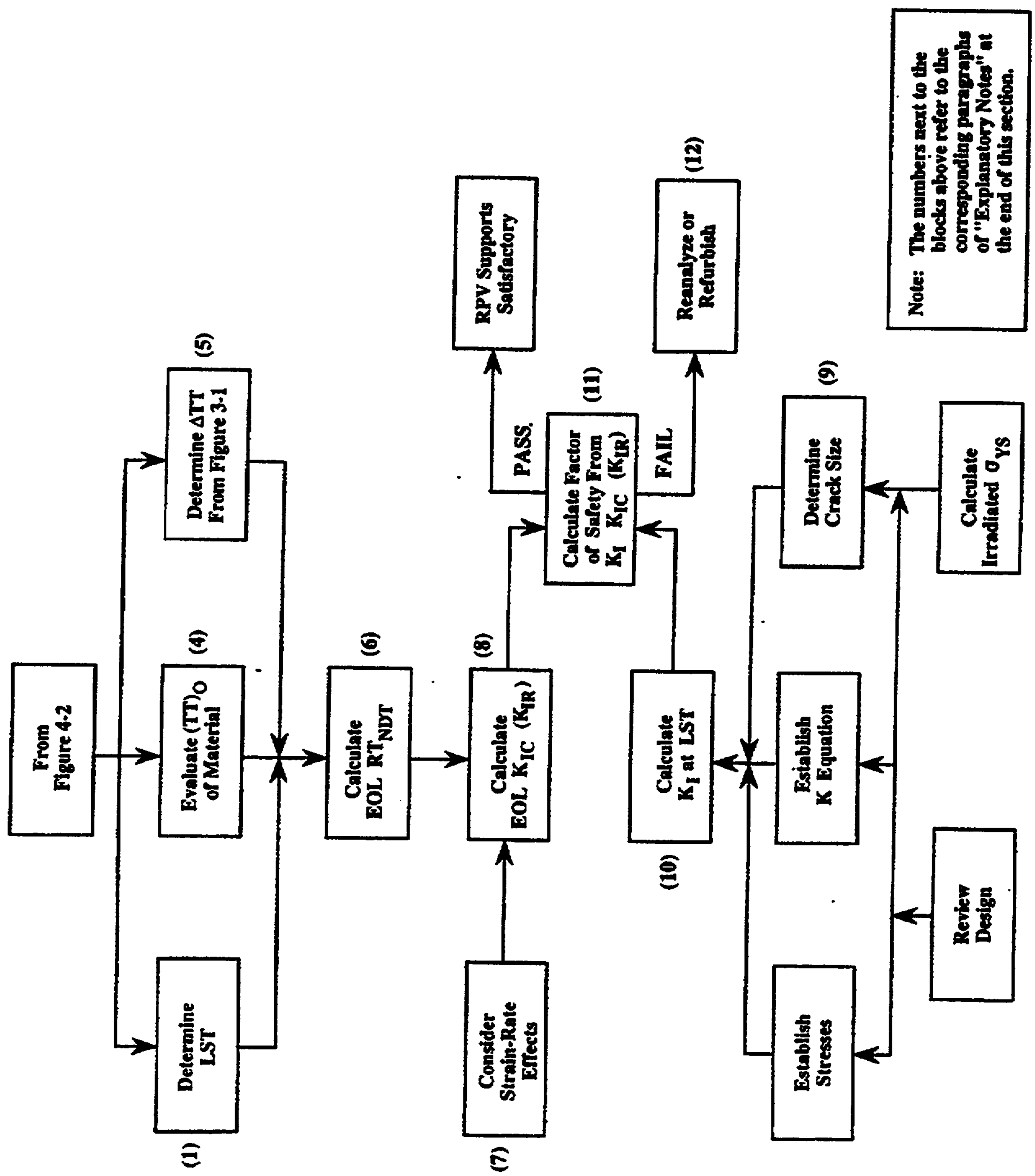


Figure 9-4 Fracture Mechanics Approach

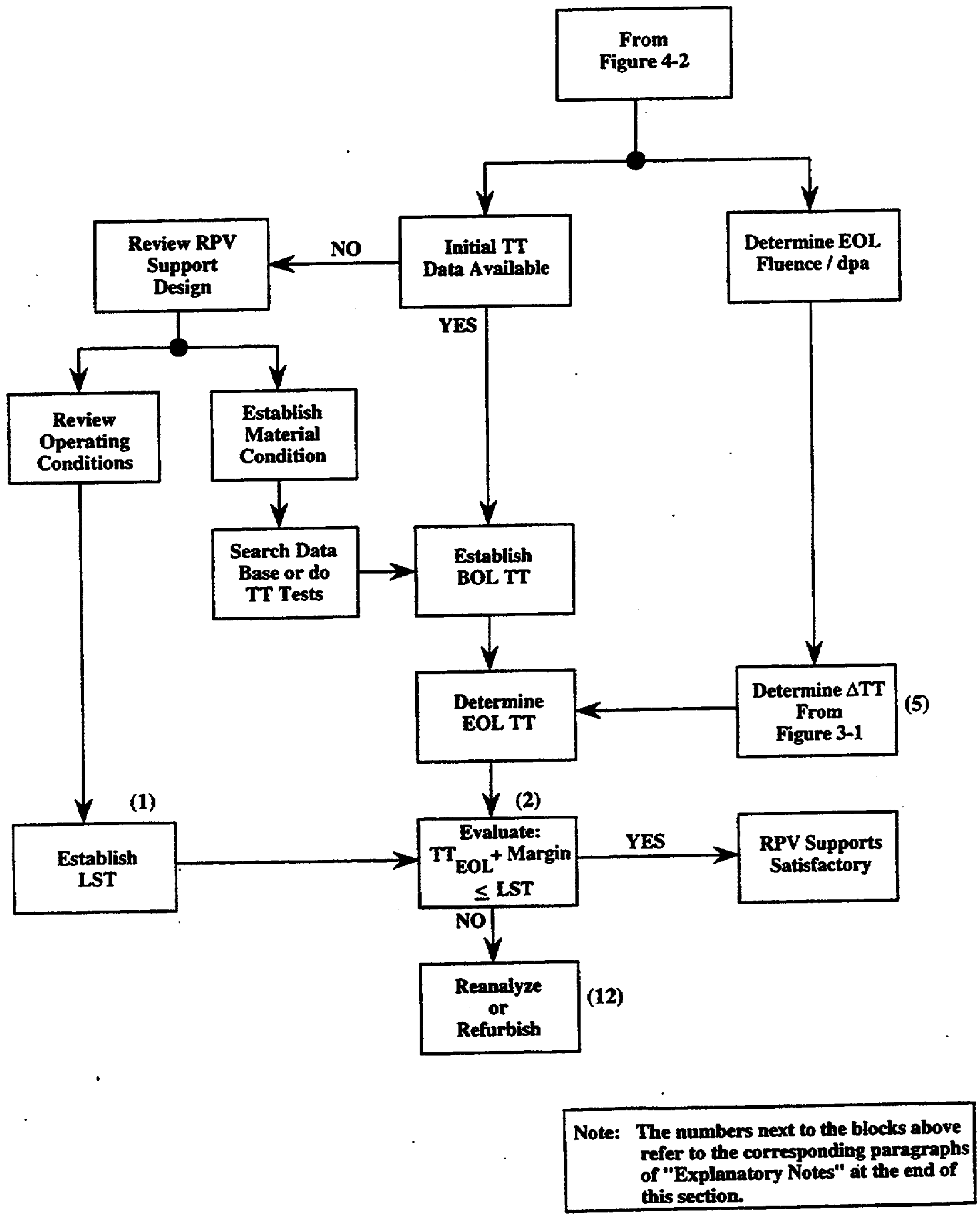


Figure 9-5 Transition Temperature Approach

Explanatory Notes for Figures 9-1 through 9-3

(See Figure 9-1)

1. The safety margin between LST-BOL$_{NDT}$ is established using Figure R-1200-1 of Appendix R to ASME Code Section XI [Ref. 9-2].

(See Figure 9-2)

1. <u>LST</u>

The lowest service temperature (LST) is defined as the minimum temperature of the most vulnerable part of the fracture-critical member when design basis accident loads occur. RPV support temperatures can be established either from measurements or theoretical calculations.

2. <u>Adjustments</u>

a. Irradiation

The radiation-induced temperature shift should be based on reliable and relevant dosimetry information.

b. Strain rate

Consideration for strain rate effects must be appropriate to the subject material. The loading rate should be estimated and its effect documented.

3. <u>NDTT Evaluation Procedure</u>

List all support materials and available NDTT data. State the authority for material tests (e.g., Subsection NF, ASME Code Section III).

a. Material having minimum specified yield strength of 180 ksi or less

For materials in new RPV supports, the NDTT should be determined in accordance with the provisions of ASTM E-208 [Ref. 9-3]. If Charpy V-notch testing is performed it should satisfy the requirements of Subsection NF, "Component Supports," Paragraphs NF 2320 and 2330 of ASME Code Section III [Ref. 9-4].

b. Estimated NDTT

For existing RPV supports, in case the NDTT cannot be determined experimentally, an estimated NDTT can be obtained from Table 9-1. The value of the NDTT, used for this purpose, should be the NDTT mean plus 1.3 standard deviation.

TABLE 9-1

COMPILATION OF NIL DUCTILITY TRANSITION TEMPERATURE RESULTS

Material	NDTT	σ	NDTT + 1.3σ	NDTT + 2σ
Cast Steels				
A-27, A-216	1" -6°F	12°F	10°F	18°F
(heat treated condition)	>1" 35	17	57	69
A-352				max. -20
Wrought Steels				
all "mild" steels*	27	31	67	89
all "mild" steels except A-201	40	28	77	96
C-Mn * (as-hot rolled)	22	13	39	48
(normalized)	-28	18	-5	8
HSLA* (as-hot rolled)	25**	12**	41**	49**
(normalized)	-50**	18**	-27**	-14**
Low Alloy, Non-Q&T				
A-302	8	28	45	64 max. -320
A-353				65**
A-387				
Quenched & Tempered				
A-508 C12				max. 40°F
A-514				max. -10°F
A-517				max. -20°F
A-533B C11				max. 20°F
A-537 C12				max. -60°F
A-543				max -60°F

* See Table 9-2 for ASTM specs included in this category.

** HSLA steels, ' high-strength" means yield strength >40 ksi. For further discussion on HSLA steels, see NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plant Stations" [Ref 9-5]

(Source: Table 4.4 of NUREG/CR-3009 [Ref. 9-6]

TABLE 9-2
CLASSIFICATION OF WROUGHT GRADES INTO GROUPS

Plain Carbon:	A-7, A-53, A-106, A-201, A-212, A-283, A-284, A-285, A-306, A-307, A-501, A-515
Carbon-Manganese:	A-36, A-105, A-516, A-537
High-strength low alloy:	A-441, A-572, A-588, A-618
Low alloy (not quenched & tempered):	A-302, A-322, A-353, A-387
Quenched & tempered:	A-193, A-194, A-325, A-354, A-461, A-490
	A-508, A514, A517, A-533, A-537, A-540
	A-543, A-563, A-574

Source: Table 3.2 of NUREG/R-3009 [Ref. 9-6]

c. Bolting Materials

Code bolting materials shall meet the fracture toughness requirements of Subsection NC, Paragraph NC-2332.3, and Appendix G, Article G-4000, "Bolting," ASME Code Section III [Ref. 9-4]. Those materials not specified in the Code must be analyzed in accordance with, and meet the criteria of notes 3a or 3b above.

d. Steels having minimum specified yield strength greater than 180 ksi

Resistance to fracture under tensile loads for materials with minimum yield strength greater than 180 ksi is considered unreliable, unless it can be justified by LEFM analysis. If such a justification cannot be provided, high-strength materials should be assumed to have inadequate fracture toughness, and the fracture mechanics or transition temperature options (Figures 9-3 and 9-4) should be deemed inapplicable. Structural adequacy of RPV supports should be demonstrated by means of the structural consequence analysis (Figure 9-5).

4. The "Criteria" are those contained in Article IWB-3000 of ASME Code Section XI [Ref. 9-2].

(See Figure 9-3)

(1) a. Estimated fracture toughness by conversion of an hyperbolic tangent function fitted to Charpy data is not acceptable.

b. Confirmation of correlations between CVN and K_{Ic} data is required.

c. The proper determination of fracture toughness curves is based on multiple K_{Ic} tests at each of several temperatures for each class of material.

d. Minimum fracture toughness values contained in Table 9-3 may be used if sufficient evidence is available to demonstrate that the material use din the RPV supports is the same as the listed in Group III in Table 9.6 of NUREG/CR-3009 [Ref. 9-6].

e. Where applicable, fracture toughness (K_{Ic} and/or K_{Ia}) can be obtained from the information contained in Appendix A, Figure A-4200-1, of ASME Code Section XI [Ref. 9-2].

(2) a. Acceptance criteria for the flaw size can be based on Subarticle IWB-3611, of ASME Code Section XI [Ref. 9-2].

b. The analysis of flaw indications should be in accordance with the provisions of Appendix A of ASME Code Section XI [Ref. 9-2].

(3) a. The maximum stress intensity factor, K_I, must be related to the flaw size a_f, as defined in Subarticle IWB-3600 of ASME Code Section XI [Ref. 9-2].

b. If the supports are subjected to combined loading, which necessitates consideration of Mode II, an appropriate fracture toughness shall be established based on the present state of the art.

c. If applicable, the reference temperature for the nil ductility transition (RT_{NDT}) may be used in conjunction with the provisions of Appendix G, Article G-2000, of ASME Code Section III, Division 1 [Ref. 9-4].

d. Calculated K_I using Eq. 1 in Appendix A, Article A-300, of ASME Code Section XI [Ref. 9-2].

(4) Safety factors shall satisfy the criteria of Article IWB-3600 of ASME Code Section XI [Ref. 9-2].

(5) a. The analysis may be performed using elastic-plastic properties of the material. The load combinations, allowable stresses, and design criteria for linear supports (consisting of shapes, beams, and columns) should conform with the provisions of Subsection 3.8.3, "Concrete and Steel Internal Structures of Steel and or Concrete Containments," and for non-linear supports with Subsection 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures" of NUREG-0800 [Ref. 9-7], respectively.

b. The thermo-hydraulic loads may be based on ANSI/ANS 58.2, Appendix B [Ref. 9-8].

Table 9-3
MINIMUM FRACTURE TOUGHNESS DATA AT 75°F

	ksi in$^{1/2}$
Plain Carbon	32
C/Mn	36
HSLA	36
Low Alloy (non Quenched and Tempered)	
A-302	30
A-353	150
A-387	65
Quenched and Tempered	
A-508	35
A-514/A-517	65
A-533	35
A-537	55
A-543	95
Other	
A-461, Gr. 630	100

Source: Table 4.5 of NUREG/CR-3009 [Ref. 9-6]