February 25, 2000

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

SUBJECT: DRAFT SAFETY EVALUATION CONCERNING THE WESTINGHOUSE OWNERS GROUP LICENSE RENEWAL EVALUATION: AGING MANAGEMENT FOR REACTOR COOLANT SYSTEM SUPPORTS, WCAP-14422, REVISION 2, FEBRUARY 19997

Dear Mr. Newton:

The U.S. Nuclear Regulatory Commission staff has reviewed the Westinghouse Owners Group (WOG) License Renewal Evaluation: Aging Management for Reactor Coolant System Supports, WCAP-14422, Revision 2, February 1997. The staff thereby transmits its draft safety evaluation (DSE) to you as an enclosure to this letter. The staff will issue a final safety evaluation upon resolution of the open items identified in the DSE.

Resolution of the open items in the DSE and satisfactory completion of the identified application action items will allow the staff to find that a WOG-member utility that references the report in a license renewal application has satisfied the requirements of 10 CFR 54.21(a)(3) for the reactor coolant system supports within the scope of WCAP-14422.

Once you have reviewed the DSE, the staff would like to schedule a meeting with you to discuss the findings in the DSE and the schedule for resolving the open items.

Sincerely,

/RA/

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

Project No. 686

Enclosure: DSE

cc w/encl: See next page

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DRAFT 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

TABLE OF CONTENTS

<u>LIST OF ACRONYMS AND ABBREVIATIONS</u>

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
DSE	draft safety evaluation
EPRI	Electric Power Research Institute
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 draft safety evaluation (DSE) 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 and preparing this DSE.

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 return 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 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 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.

Until these matters are resolved by the WOG, this is Open Item 1.

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 effects are consistent with those listed in Table 3.9-3 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 3.9-3 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 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 should also provide a discussion of the applicability of the aging effects caused by strain aging embrittlement. This is Open Item 2.

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) and Tables 3.5-3 and 3.6-3 of the draft SRP-LR (Ref. 16), 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 I "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. WOG should provide information to the staff to confirm that the WOG member plants listed in Group I in Appendix C of NUREG-0577 have performed the recommended evaluations and demonstrated that low fracture toughness was not a concern for the RCS supports of their plants.
This is Open Item 3.

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 DSE. 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 Applicant Action Item 3.

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 as Applicant Action Item 4..

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 Applicant Action Item No. 5 (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 agrees with the staff's assessment in Table B9 of NUREG-1557 that "for class 1 structures that meet the basis requirements [see the SRP-LR statement below], aggressive chemical attack is [a] non-significant ARDM," and Section 3.9.II.C.3 of draft SRP-LR, which states that "increase of porosity and permeability, cracking, and spalling caused by aggressive chemical attack is non-significant if the Class 1 structure components listed in Table 3.9-1 . . . are not exposed to [an] aggressive environment (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." 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 Applicant Action Item 5.

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 WOG report should indicate that the aging effects associated with elevated temperature are applicable and should require the applicants for license renewal to demonstrate that existing design features are capable of preventing any unacceptable degradation during the period of extended operation. This is Open Item 4.

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 Section 3.0.1.C 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 has to provide plant-specific AMPs that address the missing review elements contained in Section 3.0.1.C of the SRP-LR. This is Applicant Action Item 6.

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) 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 surveillance techniques specified for detecting 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 Applicant Action Item 7.

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 Applicant Action Item 8.

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 where warranted (see Applicant Action Item 4).

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." However, this AMP does not provide an explanation on how the "as-needed" frequency is determined. The staff considers that the inspection frequency is acceptable because they are based on NRC accepted ASME Codes. However, the frequency of leakage monitoring should be addressed by the applicants in license renewal applications as part of the plant-specific programs. This will be addressed by Applicant Action Item 5 (see Section 3.4.2).

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. WOG should revise the inspection frequency of AMP-1.2 to that of ACI 349.3R-96. This is Open Item 6. 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 Applicant Action Item 7.).

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 applicant Action Item 9.

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 staff concludes that WOG should provide a description of the inspection acceptance criteria comparable to the level of ACI 349.3R-96.
This is Open Item 7.

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 report.

3.6 Plant-Specific Programs

The staff reviewed the recommended plant-specific programs in Section 5 of the WOG GTR and finds that all 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 DSE.

4 STAFF CONCLUSION AND OPEN 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) 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 should 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) 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) Applicant Action Item 3 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.

(4) Applicant Action Item 4 Recommendations from Section 5 of the WOG report (Section 3.3.1.8)

- 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.

(5) Applicant Action Item 5 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.

(6) Applicant Action Item 6 SRP-LR (Section 3.4)

The attributes of the AMPs provided in the WOG report do not address all elements as listed in Section 3.0.I.C 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.

(7) Applicant Action Item 7 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.

(8) Applicant Action Item 8 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 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.

Applicant Action Item 9 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.

4.2 Open Items

Any WOG members can reference WCAP-14422 in a license renewal application, provided WOG or the applicant resolves the open items listed below.

(1) Open Item 1 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.

The staff can only review materials provided in the report. WOG has to correct these discrepancies and omissions to enable the staff to complete its review.

(2) Open Item 2 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. WOG should discuss the applicability of the aging effects caused by strain aging embrittlement to the RCS support components.

(3) Open Item 3 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 3.9-3 of SRP-LR and 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. WOG should confirm that its member plants listed as Group I in Appendix C of NUREG-0577 have performed the recommended evaluations in accordance with NUREG-0577 to demonstrate that the steel components of their RCS supports have sufficient fracture toughness to perform their intended functions.

(4) Open Item 4 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 WOG report should indicate that the aging effects associated with elevated temperatures are applicable and requires that applicants for license renewal demonstrate that existing design features are capable of preventing any unacceptable degradation during the extended period of operation.

(5) Open Item 5 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 not to be adequate. 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. The

WOG should revise the inspection frequency of AMP-1.2 to that recommended by ACI 349.3R-96.

(6) Open Item 6 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. Therefore, the staff considers that WOG, as a minimum, should provide a description of the inspection acceptance criteria similar to that of ACI 349.3R-96.

4.3 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 resolution of the open items described in Section 4.2 of this DSE and upon completion of any action items in Section 4.1 of this DSE, 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 action items described in Section 4.1 and resolution of the open items of Section 4.2, 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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