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

October 18, 1995

Mr. Don Croneberger
Program Director
Generic License Renewal Program
B&W Nuclear Service Company
P.O. Box 10935
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SUBJECT: DRAFT SAFETY EVALUATION CONCERNING THE B&WOG GENERIC LICENSE RENEWAL PROGRAM TOPICAL REPORT ENTITLED, "DEMONSTRATION OF THE MANAGEMENT OF AGING EFFECTS FOR THE REACTOR COOLANT SYSTEM PIPING," BAW-2243, MARCH 1995

Dear Mr. Croneberger:

The U.S. Nuclear Regulatory Commission staff has reviewed your, "Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping," applicable to the B&WOG Generic License Renewal Program (GLRP) member units, and is transmitting the draft safety evaluation (DSE) to you as an enclosure to this letter. The staff will issue a final safety evaluation report upon resolution of the open items identified in the DSE.

Resolution of the open items noted in the DSE, will allow a B&WOG GLRP member plant that references the report in a license renewal application to satisfy the requirements of 10 CFR 54.21(a)(3). Specifically, a member plant that references the report will have adequately provided a demonstration, for the reactor coolant system (RCS) piping components within the scope of this report, that the effects of aging will be adequately managed so that the intended function will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. Additionally, after resolution of the open items, referencing this report in a license renewal application and summarizing the aging management programs contained in this report in a final safety analysis report (FSAR) supplement, will provide the staff with sufficient information to find, in accordance with 10 CFR 54.29(a)(1) that actions have been identified and have been or will be taken with respect to managing the effects of aging during the period of extended operation on the functionality of the RCS piping components covered by this report, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB.

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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,

Original signed by

Dennis M. Crutchfield, Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Enclosure: DSE

Project No. 683

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DRAFT SAFETY EVALUATION (DSE)

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CONCERNING

"DEMONSTRATION OF THE MANAGEMENT OF AGING EFFECTS FOR THE REACTOR COOLANT SYSTEM PIPING"

BABCOCK & WILCOX OWNERS GROUP REPORT NUMBER BAW-2243,

PROJECT NUMBER 683

TABLE OF CONTENTS

Appendix A. List of Correspondence

Appendix B. Summary of On-Site Review

DRAFT SAFETY EVALUATION (DSE)

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CONCERNING

"DEMONSTRATION OF THE MANAGEMENT OF AGING EFFECTS FOR THE REACTOR COOLANT SYSTEM PIPING"

BABCOCK & WILCOX OWNERS GROUP REPORT NUMBER BAW-2243,

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1.0 INTRODUCTION

Pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.51, licenses to operate nuclear power plants are issued by the U.S. Nuclear Regulatory Commission (NRC) for a fixed period of time not to exceed 40 years; however, these licenses may be renewed by the NRC for an additional period not to exceed 20 years before expiration of the current operating term. The revised license renewal rule, 10 CFR Part 54, published on May 8, 1995, sets forth the requirements for the renewal of operating licenses for commercial nuclear power plants (Reference 1).

Applicants for license renewal are required by the license renewal rule to perform an integrated plant assessment (IPA). The first step of the IPA, 10 CFR 54.21(a)(1), requires the applicant to identify and list structures and components that are subject to an aging management review 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). Then, 10 CFR 54.21(a)(3) requires that for each structure and component identified in 10 CFR 54.21(a)(1), the applicant demonstrates that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. This demonstration may be contained in a plant-specific application or in the form of topical reports on specific structures and components applicable to a group of plants identified in the reports.

1.1 Babcock & Wilcox Owners Group Topical Report

By letter dated March 9, 1995, the Babcock & Wilcox Owners Group (B&WOG) Generic License Renewal Program (GLRP) submitted topical report BAW-2243, "Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping" (Reference 2) for staff review and approval. The focus of the report is on the management of the effects of aging of the reactor coolant system (RCS) piping during any period of extended operation.

The B&WOG report evaluated the aging management of the RCS piping for license renewal for GLRP member plants. The purpose of the report is to provide a technical evaluation of the effects of aging on the RCS piping and demonstrate that the aging effects for the RCS piping within the scope of the report are adequately managed for the period of extended operation associated with license renewal. The report is intended to support an individual Babcock & Wilcox (B&W) nuclear power plant utility owner in the GLRP with the technical details necessary for submitting an application for license renewal.

1.2 Conduct of Staff Review

The staff reviewed the report to determine whether the requirements set forth in 10 CFR 54.21(a)(3) can be met. The staff also obtained the assistance of Argonne National Laboratory to review potential effects of aging and aging management programs. The staff issued requests for additional information (RAIs) after completing the initial review. B&WOG responded to the staff's RAIs. The B&WOG representatives provided further clarification of their responses to the RAI questions in a number of meetings and teleconferences held with the staff. Requests for additional information, meeting summaries, and other correspondence are listed in Appendix A.

The staff's review also included an on-site examination of reference documentation at Oconee Nuclear Station Unit 1, hereinafter referred to as Oconee. Oconee is a member of the B&WOG GLRP. The purpose of the site visit was to:

(1) review and verify site specific implementation details of existing programs for managing the effects of aging described in the report as necessary during the period of extended operation to maintain the functionality of the RCS piping components, and

(2) determine from a plant among the referenced B&WOG units if the information contained in the report can reasonably be expected to be bounding for the referenced B&WOG units.

The on-site review results were discussed with representatives of the B&WOG in a meeting held on May 3 and 4, 1995, at Oconee. The results of the staff on-site review conducted at Oconee are summarized in Appendix B. Subsequent to the site visit, the B&WOG provided additional information in response to the staff's RAIs to adequately address the staff's comments from the on-site review. As a result, there are no open issues remaining from the Oconee on-site review.

2.0 SUMMARY OF TOPICAL REPORT

The report contains a generic evaluation of the management of effects of aging of the B&W RCS piping components so that the intended functions will be

maintained for any period of extended operation. The evaluation applies to the following B&WOG GLRP member plants:

Arkansas Nuclear One Unit 1
Crystal River Unit 3
Oconee Nuclear Station Units 1, 2, and 3
Three Mile Island Unit 1

Time-limited aging analyses (TLAAs), as defined in 10 CFR 54.3, for the RCS piping are outside the scope of the report.

2.1 Components and Intended Functions

The report addresses the plant-specific piping components of the reactor coolant system within the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI inservice inspection (ISI) program for Class 1 components. They are:

Piping
Valve bodies
Bolting

Section 2 of the report contains a detailed description of the plant-specific RCS piping components and their materials of construction.

In its August 11, 1995, RAI response, the B&WOG states, "The RCS piping has one function that is within the scope of license renewal: to maintain the RCS pressure boundary integrity so that the RCS may continue to perform its intended function(s) in the period of extended operation." The B&WOG clarified that the RCS piping components are to maintain their structural integrity under normal, upset, emergency, and faulted conditions, that is, ASME "Service Levels A, B, C, and D," respectively, in accordance with the CLB. Nonetheless, the B&WOG indicates that it is consistent with the industry's approach in that the "system" level intended function and not the "component" level intended function is to be maintained for renewal.

2.2 Effects of Aging

The report evaluates the applicability of the following effects of aging on the RCS piping components:

Cracking (initiation and growth)
Loss of fracture toughness
Loss of material
Loss of bolting preload (mechanical closure integrity)

The B&WOG also reviewed the operating experience of the RCS piping relating to the effects of aging. A summary of the identified potential aging effects is provided in Table 3.1 of the report. Briefly, the report describes the following as potential effects of aging for the specific RCS piping components:

Component	Potential Effects of Aging
Piping	Cracking Loss of material (carbon steel external surface)
Valve bodies	Cracking Loss of fracture toughness (cast stainless steel) Loss of material (carbon steel external surface)
Bolting	Cracking Loss of bolting preload Loss of material (low-alloy steel)

2.3 Aging Management Programs

The B&WOG report evaluates the following existing programs and concludes that they are adequate for managing the effects of aging of the RCS piping components to maintain their intended function for any period of extended operation:

ASME Section XI Class 1 ISI (Reference 3)
Response to Generic Letter 88-05 (Reference 4) on boric acid corrosion
Technical specification RCS leakage limits

In addition, the report proposes to establish a new procedure to manage the potential for loss of fracture toughness of cast stainless steel valve bodies during the period of extended operation.

A summary of specific programs for managing the effects of aging during the period of extended operation to maintain the functionality of the RCS piping components is provided in Table 4.1 of the report.

3.0 STAFF EVALUATION

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

Besides the IPA, Part 54 requires an evaluation of TLAAs in accordance with 10 CFR 54.21(c). Because the B&WOG has elected to exclude TLAAs applicable to the RCS piping components from the scope of the report, the staff did not review applicable TLAAs.

3.1 Components and Intended Functions

3.1.1 Components

The B&WOG indicated in its July 7, 1995, RAI response that the report addresses the plant-specific piping components of the reactor coolant system within the ASME Section XI ISI program for Class 1 components, including the associated valve bodies and bolting. As part of the individual plant's ISI programs, each licensee has used Regulatory Guide 1.26 (Reference 5) as the reference for classifying which components are Class 1.

The RCS piping components addressed in the report are piping, valve bodies, and bolting. Piping includes fittings, branch connections, safe-ends, and thermal sleeves. The RCS piping boundary extends to and includes the welds to nozzles or safe-ends of major RCS components. The B&WOG has addressed valves in the report as required by 10 CFR 54.21(a)(1)(i). Valve internals are not addressed because they perform the intended function, including pressure boundary, with moving parts and are excluded in accordance with 10 CFR 54.21(a)(1)(i). Bolting is associated bolting for valve body-to-bonnet and flanged connections.

Section 2 of the report contains a detailed description of the RCS piping components within the scope of the report. Some examples of piping are: 36-inch hot leg, 28-inch cold leg, 12-inch decay heat drop line, 10-inch surge line, 2-1/2-inch high pressure injection, 3/4-inch incore monitoring system line, and 1/2 to 1-inch instrumentation, vent, drain, and sampling. Some examples of valves are: 14-inch core flood check valves, 12-inch decay heat drop line isolation valves, 2-1/2-inch high pressure injection check valves, 2-1/2-inch pressurizer code safety valves, 1-1/2-inch auxiliary spray line check valves, and less than 2-inch instrumentation, vent, drain, and sampling line valves. All bolting within the scope of the report is less than 2 inches in diameter.

The staff finds that the use of the ASME Section XI Class 1 ISI boundary to define the scope of the report provides a clear and convenient boundary for GLRP member plants to use when determining what components at their specific plants are covered by this report. Additionally, the Class 1 boundary is convenient because the principal aging management program described in the report as necessary to manage the effects of aging during the period of extended operation to maintain the functionality of the RCS piping components is the ASME Section XI Class 1 ISI. This would provide uniformity in aging management for all Class 1 ISI components during the period of extended operation and would minimize administrative burden in identifying whether individual components within the report scope are covered by the Section XI Class 1 ISI program. The staff notes that the report does not constitute a complete listing of the structures and components subject to an aging management review for the B&WOG GLRP member plants as required by 10 CFR 54.21(a)(1) nor does it describe and justify any methodology for the generation of such a list as required by 10 CFR 54.21(a)(2). Therefore the staff will not make any finding relative to whether the report constitutes the complete list of RCS piping components subject to an aging management review or a scoping methodology. Individual plant applicants will need to provide a

comprehensive list of structures and components subject to an aging management review and a methodology for developing this list as part of their license renewal applications.

During the staff's on-site review at Oconee, as discussed in Appendix B, the staff found that some attachment piping to the primary loop piping was not constructed to the American National Standards Institute B31.7 Class 1 standards as indicated in the final safety analysis report for Oconee. In a letter to the NRC dated June 26, 1995, Duke Power Company, the licensee for Oconee, committed to performing the necessary Class 1 analysis on its attachment piping by August 31, 1999, in order to resolve this discrepancy. A subsequent review of other B&WOG member plants revealed similar discrepancies at Crystal River and Three Mile Island Unit 1. Crystal River and Three Mile Island Unit 1 also do not have a B31.7 Class 1 analysis for the attachment piping. In accordance with 10 CFR 54.30, the staff has determined that this matter is not within the scope of the license renewal review because it questions whether a licensee is currently meeting its CLB. This matter has been referred to the Project Directorates for Crystal River and Three Mile Island Unit 1 in the Office of Nuclear Reactor Regulation for disposition under the plant's current license.

3.1.2 Intended Functions

In its August 11, 1995, RAI response, the B&WOG states, "The RCS piping has one function that is within the scope of license renewal: to maintain the RCS pressure boundary integrity so that the RCS may continue to perform its intended function(s) in the period of extended operation." The B&WOG provided further clarification on the structural integrity of the RCS by stating: "The RCS piping components have been designed to accommodate all service loadings (i.e., Levels A-D) ... Aging management for component aging effects ... will ensure that the RCS piping components can sustain a Level C or D event during the period of extended operation." A major aging management program for the RCS piping components proposed by the B&WOG for the period of extended operation is the ASME Section XI ISI program. Further, the B&WOG states, "For the components in the scope of the RCS piping report, ASME Section XI ensures that all Service Conditions (A-D) are protected through the establishment of acceptance standards to ensure that safety margins are maintained." Finally, the B&WOG states, "When evaluating an operating component ... Section XI requires the use of the original safety margins for all operating conditions, i.e., normal, upset, emergency and faulted conditions."

The staff agrees with the B&WOG that the intended function applicable to the RCS piping components is the maintenance of the structural integrity of the reactor coolant pressure boundary under normal, upset, emergency, and faulted conditions, that is, ASME "Service Levels A, B, C, and D," in accordance with the CLB. An RCS piping component should not fail under a design loading condition such as a seismic event or other transients evaluated in the plant's CLB.

Although B&WOG agrees that the reactor coolant pressure boundary integrity is to be maintained for renewal, the phrase "so that the RCS may continue to perform its intended function(s)" is not the appropriate demonstration to

support the necessary finding for renewal because this suggests that the only concern is maintenance of a system level function rather than a structure or component function. Part 54.21(a)(3) of Title 10 of the Code of Federal Regulations requires a demonstration that the effects of aging on a structure and component will be adequately managed so that the intended functions will be maintained for the period of extended operation. The rule requires at 54.21(a)(3) that a renewal applicant demonstrate that the intended functions are maintained at the basic structure or component level rather than at a system level.

The staff's concern with the system level approach suggested by the phrase "so that the RCS may continue to perform its intended function(s)" is that a renewal applicant could potentially rely on redundancy, diversity, and defense-in-depth, which are part of the CLB, in lieu of the aging management necessary for renewal of certain structures and components subject to aging management review as required by 10 CFR 54.21(a)(1). The industry has previously proposed a similar approach to exclude these structures and components from renewal review during rulemaking to revise Part 54. However, the industry's proposal was considered by the Commission and explicitly rejected (60 FR 22487). The staff believes that the rule requires redundancy, diversity, and defense-in-depth be maintained during the period of extended operation consistent with the CLB. Currently, the staff is developing a regulatory guide to implement Part 54. The industry, through the Nuclear Energy Institute (NEI), is also developing an implementation guide for Part 54. This issue is also a continuing concern generically in the staff's review of the NEI guideline (Reference 6).

Although the staff has difficulty with the B&WOG's wording that implies that the intended functions are to be maintained at a system level as discussed above, the staff finds, subject to resolution of all open items identified in this safety evaluation, that the B&WOG report provides a satisfactory demonstration that the aging effects on all RCS components within the scope of this report will be adequately managed such that there is reasonable assurance that these RCS components will perform their intended function(s) in accordance with the CLB. Nonetheless, the characterization in the B&WOG report and RAI responses that only system level functions should be maintained remains a concern with the staff and given its importance in this first report, is considered an open item until properly clarified by the B&WOG. This is Open Item No. 1.

3.2 Effects of Aging

The effects of aging evaluated in the report are: cracking (initiation and growth), loss of fracture toughness, loss of material, and loss of bolting preload (mechanical closure integrity). The B&WOG reviewed these effects of aging for their specific applicability to the RCS piping, valve bodies, and bolting. After reviewing the report and published aging research results, the staff agrees that the B&WOG has properly identified the potential aging effects to be evaluated for the RCS piping components. A discussion of the specific aging effects on the various RCS components is provided below.

3.2.1 Piping

The B&W hot leg and cold leg piping is fabricated from carbon steel internally clad with stainless steel. A 9-1/2-inch flow meter section of the hot leg is internally clad with Alloy 82/182. The other piping within the scope of the report is fabricated from stainless steel. Safe-ends are fabricated from either stainless steel or Alloy 600. Alloy 82/182 weld buildup was also used. Branch connections are fabricated from either stainless steel or Alloy 600.

The B&WOG report states that the potential effects of aging on piping are cracking and loss of material. The latter is applicable only to the external surfaces of carbon steel piping.

The staff concurs with the B&WOG that cracking is a potential effect of aging on the piping. Cracked piping may not have the structural integrity to withstand design event loads, such as seismic, prescribed in the plant's CLB. Stainless steel piping, and Alloy 600 penetrations and steam generator tubes have cracked in service. B&W plant high pressure injection branch connections to the cold leg contain stainless steel thermal sleeves that have experienced cracking. Further, the staff agrees with the B&WOG that the regions of the RCS piping potentially susceptible to cracking are the welds because welds contain welding residual stresses and the associated heat affected zones contain a microstructure affected by welding temperatures. The RCS piping base metal is not susceptible to cracking because it has not been affected by the welding process.

The hot leg and cold leg piping is clad on the inside with stainless steel. Although the cladding is not credited in the plant's CLB as a load carrying element of the RCS piping, cracking of the cladding could expose the underlying piping base metal to the reactor coolant environment resulting in potential loss of material or cracking of the base metal, thus challenging the structural integrity of the RCS piping to withstand design event loads. The staff concurs with the B&WOG that cracking of the stainless steel cladding is unlikely because the stainless steel cladding is of high quality based on controls during fabrication, such as ferrite level, and liquid penetrant inspection prior to service. The stainless steel cladding is also unlikely to crack in service because of primary water chemistry control and the RCS main loop flow conditions are unlikely to result in thermal stratification that could subject the cladding to thermal striping or thermal fatigue. Cracking of stainless steel cladding has not been observed in RCS piping. However, it has been observed inside the pressurizer which could be subject to thermal stresses or fatigue from water level fluctuations.

The staff also agrees with the B&WOG's assessment that the Alloy 82/182 cladding, being a similar material to Alloy 600, may be susceptible to cracking. However, the staff cannot conclude that cracking of the Alloy 82/182 cladding would not cause degradation in the underlying base metal as contended by the B&WOG. Operating experience has shown that cracked cladding could lead to underlying base metal degradation (Reference 7). As one way to provide a reasonable demonstration that cracking of Alloy 82/182 cladded piping at the B&WOG GLRP member plants is not occurring or is not leading to degradation of the base metal, the staff would consider a one-time inspection

for license renewal of a sample of the Alloy 82/182 cladding or underlying base metal which confirms that the Alloy 82/182 cladding is not cracked or the underlying base metal is not degraded due to clad cracks as suggested by the B&WOG. This one-time inspection would be performed near, but prior to, the end of the current license term. This one-time inspection could be performed by the B&WOG at only one selected site if the B&WOG justifies that the inspection results bound all B&WOG GLRP member plants. This is Open Item No. 2.

The staff concurs with the B&WOG's assessment of loss of material for the RCS piping. Primary coolant leaked onto external surfaces of carbon steel components has caused boric acid corrosion resulting in loss of material from the external surfaces of the carbon steel components. However, internal pipe wall thinning due to primary coolant erosion or erosion/corrosion is not a concern for cladded carbon steel or stainless steel piping because of its resistance to erosion and erosion/corrosion.

3.2.2 Valve Bodies

The valve bodies and bonnets are fabricated from either forged or cast stainless steel, except that the pressurizer code safety valves have carbon steel bonnets. The B&WOG report states that the potential effects of aging on valve bodies are cracking, loss of fracture toughness, and loss of material. Loss of toughness is applicable only to the cast stainless steel materials. Loss of material is applicable to external surfaces of the carbon steel safety valve bonnets which have internal stainless steel inserts.

The staff concurs with the B&WOG's assessment of the potential effects of aging on the valve bodies. Stainless steel materials have cracked in service. Also, there is a reduction in fracture toughness of cast stainless steel materials due to thermal aging (References 8 and 9). Loss of material from the external surfaces of carbon steel safety valve bonnets from boric acid wastage due to coolant leakage is also an applicable aging effect.

3.2.3 Bolting

The bolting is fabricated from low-alloy steel and stainless steel. The B&WOG report states that the potential effects of aging on bolting are cracking, loss of bolting preload, and loss of material. Loss of material is applicable only to low-alloy steel materials.

The staff concurs with the B&WOG's assessment of the potential effects of aging on the bolting. Bolting has cracked and preloads have been reduced in service. Also, low-alloy bolting has corroded due to primary coolant leakage resulting in loss of material.

3.3 Aging Management Programs

3.3.1 Operating Experience Review

The staff has a concern regarding the B&WOG's consideration of operating experience. Operating experience contains information relating to plant aging

relevant to the RCS piping components. In particular, the staff review identified operating experience discussed in several generic communications concerning the effects of aging of the RCS piping components that the report did not address relative to programs necessary to manage these aging effects. Except for Generic Letter 88-05 on boric acid corrosion (Reference 4), the B&WOG has not proposed programs for managing the effects of aging derived from operating experience contained in the following generic communications:

Bulletin 79-17 on pipe cracking in stagnant borated water (Reference 10)

Bulletin 82-02 and Generic Letter 91-17 (and GSI-29) on bolting degradation (References 11 and 12)

Generic Letter 85-20 (and GSI-69) on thermal sleeve cracking (Reference 13)

Bulletin 88-08 on fatigue of attachment piping (Reference 14)

Bulletin 88-11 on fatigue of surge line (Reference 15)

Information Notice 90-10 on Alloy 600 cracking (Reference 16)

The staff has determined that operating experience described in the above generic communications represents relevant aging effects on the RCS piping components for which the B&WOG must provide aging management programs for the period of extended operation. In its August 11, 1995, RAI response, the B&WOG asserted that any existing programs that respond to these generic communications are "design validation" programs rather than programs to be credited for managing the effects of aging in the period of extended operation. Therefore, B&WOG concludes that no specific commitment regarding these programs is necessary for renewal. The B&WOG considers any existing programs in response to these generic communications as part of the CLB, and as such, will be carried forward into the period of extended operation unless modified in accordance with 10 CFR 50.59. A similar approach to rely on the existing regulatory process for license renewal was an option discussed at the NRC license renewal public workshop on September 30, 1993, during rulemaking to revise Part 54 and was not adopted in the revised rule.

Part 54 requires an explicit evaluation of programs, either existing or additional, for managing the effects of aging during the period of extended operation to maintain the functionality of the RCS piping components. Thus, the B&WOG's proposal to rely on the CLB, the so-called "design validation," to address the effects of aging for renewal is unacceptable. Further, a summary description of the programs and activities for managing the effects of aging must be provided by the renewal applicant in the FSAR supplement for license renewal in accordance with 10 CFR 54.21(d).

The B&WOG must discuss existing programs or propose new programs for managing the effects of aging, including any aging identified by information from operating experience, during the period of extended operation to maintain the functionality of the RCS piping components. These programs for renewal could be inspections or other activities and the B&WOG could also justify the use of

existing programs, including responses to generic communications. This is Open Item No. 3.

3.3.2 Evaluation of Aging Management Programs

The B&WOG evaluated existing programs and found them adequate in managing the effects of aging so that the intended function of the RCS piping components will be maintained consistent with the CLB for any period of extended operation. The only exception is that the B&WOG proposes to establish a new procedure to manage the potential for loss of fracture toughness of cast stainless steel valve bodies during the period of extended operation.

The B&WOG describes the following programs as programs necessary for managing the effects of aging during any period of extended operation to maintain the functionality of the RCS piping components:

Component	Programs
Piping	ASME Section XI "Examination Categories B-F, B-J, and B-P" Response to Generic Letter 88-05 Technical specification RCS leakage limits
Valve bodies	ASME Section XI "Examination Categories B-M-1, B-M-2, and B-P" Response to Generic Letter 88-05 Technical specification RCS leakage limits New program to manage loss of toughness of cast stainless steel
Bolting	ASME Section XI "Examination Categories B-G-2 and B-P" Response to Generic Letter 88-05 Technical specification RCS leakage limits

A description of the background of the existing programs is as follows:

ASME Section XI Class 1 ISI - All plants have ISI programs on 10-year intervals based on ASME Section XI as required by 10 CFR 50.55a(g). Plant-specific ISI programs are reviewed and approved by the staff. The Class 1 ISI program is described in Subsection IWB of Section XI and is divided into "Examination Categories." When indications or flaws are detected, Section XI also provides evaluation criteria or procedures. If the flawed component is found unacceptable for continued service, Section XI provides repair and replacement procedures.

Response to Generic Letter 88-05 - All licensees have responded to the generic letter describing their programs for mitigating the effects of boric acid corrosion of external surfaces of carbon steel reactor coolant pressure boundary components. The staff has reviewed the responses and has audited some licensee programs as part of operating plant activities.

Technical specification RCS leakage limits - Technical specifications contain surveillance requirements to monitor and trend RCS leakage, specific limits for identified and unidentified RCS leakage, and no leakage from the reactor coolant pressure boundary. Exceeding any of the RCS leakage limits results in entering a limiting condition of operation and may result in plant shutdown and NRC notification (10 CFR 50.72) and reporting (10 CFR 50.73), as appropriate. The 10 CFR 50.73 licensee event report also requires a description of corrective action to prevent recurrence.

The staff has a concern about the lack of specificity of the edition of the ASME Section XI code in the B&WOG report. The report does not explicitly identify an edition of Section XI programs for managing the effects of aging during the period of extended operation to maintain the functionality of the RCS piping components. The B&WOG intends to maintain the plant-specific edition of Section XI in effect for each plant's current 10-year ISI program, which are the 1983, 1986, and 1989 editions. Without identifying a specific edition of Section XI, the staff can not fully evaluate the specific elements of the programs for managing the effects of aging during the period of extended operation to maintain the functionality of the RCS piping components because there are variations between the various code editions. For example: Subsubarticle IWB-3640 on evaluating austenitic piping considering weld properties is not in the 1983 edition; Paragraphs IWB-3517 and IWB-3519 on the acceptance standards for bolting and valve bodies, respectively, are not in the 1983 edition; Subsubarticle IWB-3650 on evaluating ferritic piping is not in the 1983 and 1986 editions; and Appendix VII on examination personnel qualification is not in the 1983 and 1986 editions.

The B&WOG should clearly identify the edition of Section XI from which programs are relied on for managing the effects of aging during the period of extended operation to maintain the functionality of the RCS piping components. Because the latest edition of Section XI which has been reviewed and endorsed by the staff in 10 CFR 50.55a is the 1989 edition, the B&WOG should reference the 1989 edition. Or, the B&WOG must identify differences between the various code editions as they affect the RCS piping components and justify them, using the 1989 edition as the baseline.

Further, the staff has a concern about the reliability of ultrasonic examinations. The ASME addressed this concern by introducing Appendix VIII on performance demonstration in the 1989 Addenda to the 1989 edition of Section XI. The B&WOG should commit to Appendix VIII for renewal to provide assurance on the reliability of ultrasonic examinations for managing aging of the RCS piping components during the period of extended operation.

The use of a specific Section XI edition and Appendix VIII of Section XI for renewal is Open Item No. 4.

Specific programs for managing the effects of aging during the period of extended operation to maintain the functionality of the RCS piping, valve bodies, and bolting are evaluated in Sections 3.3.2.1, 3.3.2.2, and 3.3.2.3, respectively, of this safety evaluation.

3.3.2.1 Piping

The report describes the ASME Section XI Class 1 ISI program "Examination Categories B-F, B-J, and B-P," response to Generic Letter 88-05, and technical specification RCS leakage limits as programs necessary to manage the effects of aging of the RCS piping during the period of extended operation to maintain the reactor coolant pressure boundary. For ASME Section XI, "Examination Category B-P" consists of system leakage and hydrostatic tests. Visual "VT-2" examination is conducted to locate evidence of leakage during the tests. The leakage test is conducted every refueling outage prior to plant startup and the hydrostatic test is conducted every 10 years. Technical specification RCS leakage limits are described in Section 3.3.2 of this safety evaluation. "Examination Categories B-F and B-J" are as follows:

"Examination Category"	Component Description	Size (inches)	Examination
B-F	Pressure retaining dissimilar welds in vessel nozzles	≥4 >1 but <4	Volumetric and surface Surface
B-J	Pressure retaining welds in piping	≥4 >1 but <4	Volumetric and surface Surface

Volumetric examination indicates the presence of discontinuities throughout the volume of material and uses techniques such as ultrasonics or radiography. Surface examination indicates the presence of surface discontinuities and uses techniques such as liquid penetrant or magnetic particles. "Examination Category B-F" examination is conducted for all applicable welds every 10 years. "Examination Category B-J" examination is conducted for 25 percent of the applicable welds every 10 years.

When an indication or flaw is detected, the component is evaluated according to Article IWB-3000 of Section XI to determine if the component is acceptable for continued service, that is, if the component can maintain its structural integrity under ASME "Service Levels A, B, C, and D." Analytical flaw evaluations are subject to review and approval by the NRC. If the component is determined unacceptable for service, Articles IWB-4000 and IWB-7000 of Section XI provide repair and replacement procedures, respectively.

The first aging effect applicable to piping is cracking. The B&WOG indicates and the staff agrees that the region of the RCS piping potentially susceptible to cracking is at the welds as discussed in Section 3.2.1 of this safety evaluation. Thus, the aging management program should focus on welds. Should cracking be detected in the welds, Section XVI, "Corrective Action," in Appendix B to 10 CFR Part 50 requires a root cause determination and corrective measures. The corrective action may involve activities extending beyond welds as appropriate. The exception is Alloy 82/182 cladding which may result in degradation of underlying base metal and is Open Item No. 2.

Operating experience shows that cracking, if it occurs, originates from the inside surface of the piping. Some factors contributing to this are: water

environment, geometric discontinuities such as crevices, and welding residual stresses. A volumetric inspection can usually detect significant cracking originating from the inside surface. However, a crack originating from the inside surface of the piping but has not penetrated through the pipe wall can not be detected by either surface examination of the outside surface or leakage test. "Examination Categories B-F and B-J" contain volumetric examinations for 4 inches or larger piping. When a flaw is detected, ASME Section XI provides structural integrity evaluation criteria in Article IWB-3000 considering all loading conditions. Thus, the staff finds ASME Section XI Class 1 ISI adequate to manage cracking of 4 inches or larger RCS piping to ensure the reactor coolant pressure boundary function of the piping components during the period of extended operation.

For RCS piping less than 4 inches but greater than 1 inch, the report relies on a surface examination of the piping outside surface and leakage detection under "Examination Category B-P" and technical specifications. For RCS piping less than or equal to 1 inch, the report relies solely on leakage detection. As discussed above, a crack originating from the inside surface of the piping but has not penetrated through the pipe wall can not be detected by either surface examination of the outside surface or leakage test. Part-through wall cracks are not self-revealing. Although the staff agrees that plants have extensive leakage monitoring requirements and that corrective actions would be taken to prevent recurrence, the staff's concern is that piping with a part-through wall crack and therefore, not leaking, may not have the structural integrity to ensure the reactor coolant pressure boundary function of the piping components for all design loads. Cracked, but not leaking, piping could fail during a design loading condition such as a seismic event. Further, aging could be a common cause of degradation of piping in a similar service environment. Thus, the staff finds the B&WOG proposal for managing cracking of RCS piping less than 4 inches unacceptable.

One way to provide a reasonable demonstration of the integrity of the RCS piping between 1 and 4 inches, would be a one-time inspection for license renewal of a small sample of RCS piping between 1 and 4 inches in size using volumetric examination to provide evidence that the piping is not experiencing cracking and therefore would be capable of maintaining its pressure boundary integrity under CLB design loads. This one-time inspection would be performed near, but prior to, the end of the current license term on a small sample of piping welds which are already being inspected with surface examination techniques. The B&WOG should propose a small sample distributed among subsystems, pipe sizes, and flow conditions. If a flaw is detected in this sample, the successive inspections described in Subsubarticle IWB-2420 of Section XI and additional examinations described in Subsubarticle IWB-2430 of Section XI would apply, as appropriate. Appendix B to 10 CFR Part 50 requires a root cause determination and corrective measures.

For piping 1 inch or less in size, the B&WOG should propose a program to manage cracking so that the reactor coolant pressure boundary function of this piping will be maintained consistent with the CLB for the period of extended operation. This could be a one-time inspection for renewal of a small sample of 1 inch or less piping using volumetric methods similar to the inspection discussed for piping between 1 and 4 inches. Alternatively, the B&WOG may

propose other options with justification for staff review and approval. However, leakage detection alone is not an acceptable aging management program to ensure the reactor coolant pressure boundary function of these piping components. The potential concern of common mode failure due to aging which may cause the failure of multiple degraded pipes under design loading, such as seismic, should be addressed. The staff believes that a risk based approach could be beneficial in considering the breadth and scope of an aging management approach. However, the statements of consideration accompanying Part 54 states, "Probabilistic arguments may assist in developing an approach for aging management adequacy. However, probabilistic arguments alone will not be an acceptable basis for concluding that, for those structures and components subject to an aging management review, the effects of aging will be adequately managed in the period of extended operation." (60 FR 22468)

The program to manage cracking of less than 4-inch piping is Open Item No. 5.

The staff finds that operating experience for thermal sleeves has not been adequately addressed in the B&WOG report for the period of extended operation. The high pressure injection connection thermal sleeves have been examined and those susceptible to cracking have been replaced with a new design. Regardless of whether the thermal sleeve has been replaced or not, the B&W plant owners are performing periodic augmented inspections on the thermal sleeves and associated safe ends using volumetric methods. The staff believes that continued inspection is necessary because of the potential for cracking. However, the B&WOG has neither committed to continue the augmented inspections nor proposed alternative programs to manage cracking of the thermal sleeves during the period of extended operation. This is part of Open Item No. 3.

Additionally, the B&WOG has not proposed programs to manage attachment piping, surge line, Alloy 600 material, and piping in stagnant borated water during the period of extended operation to address information from operating experience. This is also part of Open Item No. 3. The staff recognizes, however, that fatigue analysis is a TLAA and is not within the scope of this B&WOG report. Should the B&WOG evaluate TLAAs of the RCS piping components before the staff resolution of GSI-166, "Adequacy of Fatigue Life of Metal Components," the B&WOG could follow the staff guidance in SECY-95-245 (Reference 17) to evaluate a sample of components with high fatigue usage, using the latest available environmental fatigue data.

The second aging effect applicable to piping is loss of material. The report describes the boric acid wastage surveillance programs implemented by licensees in response to NRC Generic Letter 88-05 as necessary in managing the potential loss of material on the external surfaces of the carbon steel RCS piping components. Generic Letter 88-05 requested licensees to provide assurance that a program has been implemented to address the corrosive effects of RCS leakage at less than technical specification limits. Briefly, the program described in Generic Letter 88-05 includes: determination of potential leakage locations, procedures for locating small leaks, methods for conducting examinations and performing evaluations, and corrective actions to prevent recurrences. Although small leaks will likely continue to occur, the staff finds the programs committed to by licensees in response to Generic Letter 88-05, in conjunction with leakage detection conducted under

"Examination Category B-P" and technical specification RCS leakage limits, adequate in managing the loss of material on the external surfaces of carbon steel reactor coolant pressure boundary components for the period of extended operation.

3.3.2.2 <u>Valve Bodies</u>

The report describes the ASME Section XI Class 1 ISI program, "Examination Categories B-M-1, B-M-2, and B-P," response to Generic Letter 88-05, technical specification RCS leakage limits, and a new program to manage loss of fracture toughness of cast stainless steel as programs necessary to manage the effects of aging of the RCS valve bodies during the period of extended operation to maintain the reactor coolant pressure boundary. ASME Section XI "Examination Category B-P" consists of system leakage and hydrostatic tests and is described in Section 3.3.2.1 of this safety evaluation. Technical specification RCS leakage limits are described in Section 3.3.2 of this safety evaluation. "Examination Categories B-M-1 and B-M-2" are as follows:

"Examination Category"	Component Description	Size (inches)	Examination
B-M-1	Pressure retaining welds in valve bodies	≥4	Volumetric
B-M-2	Valve bodies	>4	Visual "VT-3" of internal surfaces

Examination is limited to at least one valve within each group of similar valves every 10 years. Visual "VT-3" examination is conducted to determine the general mechanical and structural condition of components. Flaws detected in "Examination Categories B-M-1 and B-M-2" may be acceptable for continued service if they meet the acceptance standards in IWB-3518 and IWB-3519, respectively.

"Examination Category B-M-1" also contains a surface examination for welds in valve bodies of less than 4 inches. However, the report states that valves of 4 inches and less within the scope of the report do not contain welded joints within the valve bodies. Thus, this portion of "Examination Category B-M-1" is not applicable to valve bodies within the scope of the report.

The first aging effect applicable to valve bodies is cracking. The staff finds the ASME Section XI valve body examinations, that is, "Examination Categories B-M-1 and B-M-2," adequate in managing potential cracking. This is because these examinations are based on volumetric examinations and visual examination of the internal surfaces.

Although valves less than 4 inches in size within the scope of the report would not be inspected under "Examination Categories B-M-1 and B-M-2," the staff finds cracking of these valves would be adequately managed during the period of extended operation because (1) valves have a thicker wall and a lower stress than adjacent piping and degradations would likely be detected in piping first through piping inspections and (2) degradation detected in valves

4 inches and larger through inspections would reasonably result in corrective action for valves less than 4 inches, as appropriate.

The second aging effect applicable to the valve bodies is loss of fracture toughness for valve bodies fabricated from cast stainless steel. The B&WOG is proposing a new program for managing the potential for loss of toughness for cast stainless steel during the period of extended operation.

The B&WOG evaluated the loss of fracture toughness of cast stainless steel due to thermal aging and concludes that the toughness of aged cast stainless steel is similar to that of submerged arc welds (SAWs). The staff reviewed the recently developed lower-bound toughness property for aged cast stainless steel (References 8 and 9) and agrees that it is similar to that used in Subsubarticle IWB-3640 of ASME Section XI in evaluating SAWs. This suggests that aged cast stainless steel and SAWs could be treated similarly regarding their toughness behavior.

Section XI contains procedures to evaluate flaws in SAWs, that is, IWB-3640. However, there is currently no procedure in ASME Section XI to evaluate flaws in cast stainless steel materials. Because the lower-bound toughness of aged cast stainless steel is similar to the toughness used in evaluating flaws in SAWs in Section XI, the staff agrees that it is appropriate to use IWB-3640, based on the toughness of SAWs, to evaluate flaws in aged cast stainless steel when a flaw evaluation is performed per ASME Section XI during the period of extended operation. Alternatively, the staff also agrees that the toughness of aged cast stainless steel components used for flaw evaluations during the period of extended operation can be justified on a case-by-case basis using actual material data and the procedures in NUREG/CR-6177 and NUREG/CR-4513, Revision 1 (References 8 and 9). The B&WOG also proposes a third option based on IWB-3640 flaw evaluation procedures for wrought stainless steel if IWB-3641(c) of Section XI is met. Subparagraph IWB-3641(c) states: "For cast stainless steel materials, adequate toughness for the pipe to reach limit load after aging shall be demonstrated." Although the staff has accepted this provision when endorsing ASME Section XI, this provision is of limited use because ASME Section XI does not contain specific procedures to demonstrate reaching limit load. In any case, all flaw evaluations are subject to NRC review and approval as specified in IWB-3640. Thus, the B&WOG's program for managing the loss of fracture toughness of cast stainless steel during the period of extended operation is acceptable.

The third aging effect applicable to valve bodies is loss of material from the external surfaces of carbon steel pressurizer code safety valve bonnets. The report describes the boric acid wastage surveillance programs implemented by licensees in response to NRC Generic Letter 88-05 as necessary in managing the potential loss of material on the external surfaces of the carbon steel valve bonnets. Similar to the discussion in Section 3.3.2.1 of this safety evaluation, the staff finds the programs committed to by licensees in response to Generic Letter 88-05, in conjunction with leakage detection under "Examination Category B-P", and technical specification RCS leakage limits adequate in managing the aging effect of loss of material on the external surfaces of carbon steel valves within the scope of this report during the period of extended operation.

3.3.2.3 Bolting

The report describes the ASME Section XI Class 1 ISI program, "Examination Categories B-G-2 and B-P," response to Generic Letter 88-05, and technical specification RCS leakage limits as programs necessary to manage the effects of aging of the RCS bolting during the period of extended operation to maintain the reactor coolant pressure boundary. ASME Section XI, "Examination Category B-P," consists of system leakage and hydrostatic tests and is described in Section 3.3.2.1 of this safety evaluation. Technical specification leakage limits are described in Section 3.3.2 of this safety evaluation. "Examination Category B-G-2" is as follows:

"Examination Category"	Component Description	Size (inches)	Examination
B-G-2	Pressure retaining bolting	≤2	Visual "VT-1" of all bolts, studs, and nuts

"Examination Category B-G-2" is conducted every 10 years. Valve bolting examination is limited to bolting on valves that are selected for examination under "Examination Category B-M-2." Visual "VT-1" examination is conducted to determine the condition of the component or surface examined, including such conditions as cracks, wear, corrosion, erosion, or physical damage on the surfaces of the components. Flaws detected in "Examination Category B-G-2" may be acceptable for continued service if they meet the acceptance standards in IWB-3517.

The first and second aging effects applicable to the bolting are cracking and loss of bolting preload. All bolting within the scope of the report is less than 2 inches in diameter. The staff finds the ASME Section XI bolting examination proposed by the B&WOG adequate in managing potential cracking and loss of preload because the bolting will be examined when the valves are disassembled for valve inspections. Mechanical closure integrity can also be monitored through "Examination Category B-P" system leakage and hydrostatic tests and technical specification RCS leakage limits.

The third aging effect applicable to the bolting is loss of material of low-alloy steel bolting. The report describes the boric acid wastage surveillance programs implemented by licensees in response to NRC Generic Letter 88-05 as necessary in managing the potential loss of material of low-alloy steel bolting during the period of extended operation. Similar to the discussion in Section 3.3.2.1 and 3.3.2.3 of this safety evaluation, the staff finds the programs committed to by licensees in response to Generic letter 88-05 to be acceptable for managing the aging effect of loss of material for low alloy steel bolting within the scope of this report during the period of extended operation.

However, information from operating experience indicates that there are additional elements of bolting maintenance procedures that should be considered, such as personnel training, installation and maintenance procedures, plant-specific bolting degradation history, and corrective

measures (Reference 11). The B&WOG has not proposed programs to manage the effects of bolting degradation during the period of extended operation to address this information and this is part of Open Item No. 3.

4.0 CONCLUSIONS

The staff has reviewed the subject B&WOG topical report (Reference 2) and additional information submitted by the B&WOG. On the basis of its review, the staff concludes that, upon resolution of the open items discussed in Section 4.2, the B&WOG report provides an acceptable demonstration that the aging effects of RCS components within the scope of this report will be adequately managed such that there is reasonable assurance that the RCS components will perform their intended function(s) in accordance with the CLB. Any B&WOG GLRP member plant may reference this report in a license renewal application to satisfy the requirements of 10 CFR 54.21(a)(3) for providing a demonstration that the effects of aging on the RCS piping components within the scope of this report will be adequately managed. The staff also concludes that, subject to resolution of the open items of Section 4.2, referencing this report in a license renewal application and summarizing the aging management programs contained in this report in a FSAR supplement will provide the staff with sufficient information to make the necessary findings required by 54.29(a)(1) for components within the scope of this report.

4.1 Renewal Applicant Action Items

When incorporating the B&WOG topical report in its renewal application, the license renewal applicant is to verify that its plant is bounded by the topical report. Further, the renewal applicant is to commit to programs described as necessary in the report to manage the effects of aging during the period of extended operation on the functionality of the RCS piping components. A summary description of these programs is to be provided in the license renewal FSAR supplement in accordance with 10 CFR 54.21(d).

Any deviations from the aging management programs described within this report as necessary to manage the effects of aging during the period of extended operation to maintain the functionality of the RCS piping components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3).

Additionally, since the staff does not make any finding relative to whether the B&WOG report constitutes the complete list of RCS piping components subject to an aging management review or the adequacy of a scoping methodology, individual plant applicants will need to provide a comprehensive list of structures and components subject to an aging management review and a methodology for developing this list as part of their license renewal applications.

4.2 Open Items

1. System versus component level intended function (Section 3.1.2 of this safety evaluation)

Part 54.21(a)(3) of Title 10 of the Code of Federal Regulations requires a demonstration that the effects of aging on a structure and component will be adequately managed so that the intended functions will be maintained for the period of extended operation. The rule requires at 54.21(a)(3) that a renewal applicant demonstrate that the intended functions are maintained at the basic structure or component level rather than at a system level.

Although the B&WOG report provides an acceptable demonstration that the effects of aging of the RCS components within the scope of this report will be adequately managed for the period of extended operation, the characterization in the B&WOG report and RAI responses that only system level functions should be maintained, remains a concern with the staff and given its importance in this first report, is considered an open item until properly clarified by the B&WOG.

2. Alloy 82/182 Cladding (Section 3.2.1 of this safety evaluation)

   A 9-1/2-inch section of each hot leg is cladded with Alloy 82/182. The B&WOG indicates that Alloy 82/182 cladding may be susceptible to cracking. However, the staff can not conclude that cracking of this cladding would not cause degradation in the underlying base metal as contended by the B&WOG. The staff would accept a one-time inspection for renewal, at or near the end of the current license term, of either the Alloy 82/182 cladding or the underlying base metal. The purpose of the inspection is to demonstrate either the Alloy 82/182 cladding is not cracked or the underlying base metal is not degraded with clad cracks as suggested by the B&WOG. This one-time inspection could be performed at one selected site to bound all B&WOG GLRP member plants.

3. Aging management programs to address operating experience (Section 3.3.1 of this safety evaluation)

   Operating experience contains information relating to plant aging relevant to the RCS piping components. However, the staff finds the B&WOG has not properly factored the results of the operating experience review into the programs for managing the effects of aging during the period of extended operation on the functionality of the RCS piping components. In particular, the operating experience review found several generic communications with relevant information on the effects of aging of the RCS piping components. The B&WOG must identify, describe, and justify programs necessary to manage the effects of aging for renewal to address information from operating experience. Further, a summary description of the aging management programs must be provided by the renewal applicant in the FSAR supplement for renewal in accordance with 10 CFR 54.21(d).

4. ASME Section XI edition and Appendix VIII (Section 3.3.2 of this safety evaluation)

   The report does not explicitly identify an edition of Section XI programs for managing the effects of aging during the period of extended

operation to maintain the functionality of the RCS piping components. It appears that the B&WOG intends to maintain the plant-specific edition of ASME Section XI in effect for each plant's current 10-year ISI program, which are the 1983, 1986, and 1989 editions. Without identifying a specific edition of ASME Section XI, the staff cannot fully evaluate the specific elements of the programs for managing the effects of aging during the period of extended operation to maintain the functionality of the RCS piping components. This is because there are variations between the various code editions. The B&WOG should either standardize the reference to the 1989 edition, the latest edition endorsed by the staff in 10 CFR 50.55a, or describe and justify differences applicable to the RCS piping components between the various editions. Further, the staff has a concern about the reliability of ultrasonic examinations. The B&WOG should commit to Appendix VIII of Section XI to provide assurance on the reliability of ultrasonic examinations during the period of extended operation.

5. Piping less than 4 inches in size (Section 3.3.2.1 of this safety evaluation)

   The B&WOG relies on ASME Section XI to manage cracking of the RCS piping. However, ASME Section XI specifies only leakage detection for piping 1 inch or less in size and it specifies a surface examination and leakage detection for piping between 1 and 4 inches. The staff's concern is that leakage detection and surface examination, such as liquid penetrant examination of the outside surface of the piping, do not detect internal part-through wall cracks which could cause a failure of the reactor coolant pressure boundary under design loads. The staff will accept a one-time inspection for renewal of a small sample of between 1 and 4 inches piping using volumetric methods, such as radiography or ultrasonics. For piping 1 inch and less in size, the B&WOG should propose a program to manage cracking so that the reactor coolant pressure boundary function of this piping will be maintained for the period of extended operation.

5.0 <u>REFERENCES</u>

1. Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," <u>Federal</u> <u>Register</u>, Vol. 60, No. 88, May 8, 1995, pp. 22461-22495.

2. BAW-2243, "Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping," Babcock & Wilcox Owners Group, March 1995.

3. Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," The American Society of Mechanical Engineers, 1983, 1986, and 1989 editions.

4. Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," NRC, March 17, 1988.

5. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 3 (for comment), NRC, February 1976.

6. Letter to U.S. NRC from Douglas J. Walters of NEI, dated September 27, 1995, Project No. 690, transmittal of NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Draft Revision C, August 18, 1995.

7. Information Notice 94-63, "Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks," NRC, August 30, 1994.

8. "Assessment of Thermal Embrittlement of Cast Stainless Steels," NUREG/CR-6177, May 1994.

9. "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR-4513, Rev. 1, August 1994.

10. Bulletin 79-17, "Pipe Cracks in Stagnant Borated Water Systems at PWR Plants," NRC, July 26, 1979.

11. Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," NRC, June 2, 1982.

12. Generic Letter 91-17, "Generic Safety Issue 29, 'Bolting Degradation of Failure in Nuclear Power Plants,'" October 17, 1991.

13. Generic Letter 85-20, "Resolution of Generic Issue 69: High Pressure Injection/Make-Up Nozzle Cracking in Babcock and Wilcox Plants," NRC, November 8, 1985.

14. Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," NRC, June 22, 1988, and its three supplements.

15. Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," NRC, December 20, 1988.

16. Information Notice 90-10, "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600," NRC, February 23, 1990.

17. SECY-95-245, "Completion of the Fatigue Action Plan," NRC, September 25, 1995.

6.0 APPENDICES

Appendix A. List of Correspondence

1. "Summary of Meeting Between the U.S. Nuclear Regulatory Commission Staff and B&WOG Representatives to Discuss the B&WOG GLRP Reactor Coolant System Piping Evaluation Report," dated December 30, 1994, prepared by T.G. Hiltz of the NRC for a meeting between the NRC and B&WOG held on December 15, 1994.

2. Letter to U.S. NRC, attention D.M. Crutchfield, from D.K. Croneberger of B&WOG Generic License Renewal Program, dated March 9, 1995, transmittal of, "Demonstration of the Management of Aging Effects for the reactor Coolant System Piping," Topical Report BAW-2243, March 1995.

3. "Summary of Senior Management Meeting Between the U.S. NRC Staff and B&WOG Representatives to Discuss the B&WOG GLRP," dated March 16, 1995, prepared by T.G. Hiltz of the NRC for a meeting between the NRC and B&WOG held on March 14, 1995.

4. Letter to D.K. Croneberger of B&WOG Generic License Renewal Program from T.G. Hiltz of NRC dated March 28, 1995, "Request for Additional Information Regarding the B&W Owners Group Topical report BAW-2243."

5. "Summary of Meeting Between the U.S. NRC Staff and B&WOG Representatives to Discuss the B&WOG GLRP Reactor Coolant System Piping Aging Management Topical Report," dated April 19, 1995, prepared by T.G. Hiltz of the NRC for a meeting between the NRC and B&WOG held on April 3, 1995.

6. Letter to D.K. Croneberger of B&WOG Generic License Renewal Program from T.G. Hiltz of the NRC dated April 20, 1995, "Request for Additional Information Regarding the B&W Owners Group Topical Report BAW-2243."

7. Letter to U.S. NRC, attention T.G. Hiltz, from D.K. Croneberger of B&WOG Generic License Renewal Program, dated May 5, 1995, "Technical Reports," transmittal of 2 reports: E. Howells & L.H. Vaughn, 'Corrosion of Reactor Metal in Boric Acid,' RDE-1086, B&W, Alliance Ohio, August 1960 and C.A. Ouellette, 'B&W Boric Acid Corrosion Research and the Wastage and Inspection Procedures for RCS Leakage,' BAW-2126, B&W Nuclear Services Company, Lynchburg, Virginia, December 1990.

8. Letter to D.K. Croneberger of B&WOG Generic License Renewal Program from T.G. Hiltz of U.S. Nuclear Regulatory Commission dated May 10, 1995, "Request for Additional Information Regarding the B&W Owners group Topical report BAW-2243."

9. Letter to U.S. NRC, from D.K. Croneberger of B&WOG Generic License Renewal Program, dated May 16, 1995, transmittal of response to NRC staff Requests for Additional Information dated March 28, 1995, and April 20, 1995.

10. "Summary of Meeting Between the U.S. NRC Staff and B&WOG Representatives to Discuss the B&WOG Generic License Renewal Program," dated May 21, 1995, prepared by T.G. Hiltz of the NRC for a meeting between the NRC and B&WOG held on May 11, 1995.

11. "Summary of Meeting Between the U.S. NRC Staff and Babcock and Wilcox Owners group (B&WOG) Representatives at Oconee Nuclear Station," dated May 31, 1995, prepared by T.G. Hiltz of the NRC for a meeting between the NRC and B&WOG held on May 3 and 4, 1995.

12. "Summary of Meeting Between the U.S. NRC Staff and B&WOG Representatives," dated June 1, 1995, prepared by T.G. Hiltz of the NRC for a meeting between the NRC and B&WOG held on May 24, 1995.

13. Letter to U.S. Nuclear Regulatory Commission, from D.K. Croneberger of B&WOG Generic License Renewal Program, dated July 7, 1995, transmittal of revised responses to NRC staff Requests for Additional Information dated March 28, 1995, April 20, 1995, and initial responses to NRC staff Request for Additional Information dated May 10, 1995.

14. Letter to U.S. NRC, from R.B. Borsum of B&WOG Generic License Renewal Program, dated August 1, 1995, transmittal of response to NRC staff Request for Additional Information, Question #8, dated March 28, 1995.

15. Letter to U.S. NRC, from D.K. Croneberger of B&WOG Generic License Renewal Program, dated August 11, 1995, transmittal of responses to NRC staff Requests for Additional Information dated March 28, 1995, and April 20, 1995, modified following a conference call on August 3, 1995.

16. "Summary of Teleconference Between the U.S. NRC Staff and B&WOG Representatives to Discuss Their Response to a Request for Additional Information," dated August 17, 1995, prepared by J.P. Moulton of the NRC for a teleconference between the NRC and B&WOG held on August 3, 1995.

17. Letter to U.S. NRC, from D.K. Croneberger of B&WOG Generic License Renewal Program, dated August 25, 1995, transmittal of clarifications to NRC staff Requests for Additional Information and modifications to Chapters 3 and 4 of the subject topical report, BAW-2243.

18. "Summary of Teleconference Between the U.S. Nuclear Regulatory Commission Staff and B&WOG Representatives to Discuss Their Response to the August 3, 1995, Teleconference," dated August 25, 1995, prepared by J.P. Moulton of the NRC for a teleconference between the NRC and B&WOG held on August 17, 1995.

Appendix B. Summary of On-Site Review

This appendix discusses results of the NRC staff's on-site review of the B&WOG RCS piping topical report at Oconee.

To support the NRC staff's review of the B&WOG GLRP topical report, "Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping," three staff members from NRR visited the Oconee site on May 3 and 4, 1995. The on-site review focused on a limited sample of items associated with Oconee Unit 1's reactor coolant system piping, viz., hot leg piping to steam generator "B", the incore monitoring system (IMS) piping, and the high pressure injection (HPI) thermal sleeves.

The following is a summary of the items reviewed by the staff during the on-site visit and they are listed in the order corresponding to the on-site scope of review:

(a) Material properties excluding welds

The staff verified that the material properties listed in the subject report agreed with the site design basis documentation for the hot leg piping, decay heat removal branch connection and attached piping, flowmeter instrumentation line, resistance temperature detector mounting boss, and the IMS lines.

(b) Class 1 design and ISI boundaries

The staff reviewed the plant drawings which identified the Class 1 design boundary and the Class 1 ISI boundary. The staff concluded that the Class 1 design boundary was in compliance with the Oconee licensing basis (that is, out to and including the first isolation valve or to a vessel nozzle) and the Class 1 ISI boundary encompassed the Class 1 design boundary. It was noted that the Class 1 ISI boundary for the decay heat removal drop leg from the RCS, as well as the attached 3-inch drain line, extended out to and included the second isolation valve. Also, as required by the ASME Section XI for the system leakage test, the visual examination was extended to include second closed valve at the boundary extremity. The staff was satisfied that boundaries were properly identified. However, the B&WOG report still needed to be updated to reflect the differences in design boundaries between the various B&WOG GLRP member plants as documented in RAI #3.

(c) Last ISI performed on welds ("Examination Categories B-F and B-J")

For the sample selected, the staff verified that the "Examination Category B-F" pressure retaining dissimilar metal welds and the "Examination Category B-J" pressure retaining welds in piping were examined during the second inspection interval. In addition, the staff reviewed the actual examination reports signed by the Authorized Nuclear Inspector (ANI) for the sampled welds and noted no problems. In summary, based on the sample reviewed, the

staff was satisfied that the ISI program for Class 1 welds was being implemented in accordance with the ASME Section XI requirements.

(d) Last ISI performed on valves equal to or greater than 4-inch nominal pipe size ("Examination Categories B-M-1 and B-M-2"), including bolting. If subject valves were not inspected, provide inspection results of valve within ISI grouping.

The licensee stated that it does not have any valves that are subject to the requirements of "Examination Category B-M-1" for pressure retaining welds in valve bodies. From the sample selected for "Examination Category B-M-2" review, the decay heat removal drop leg valve, "1LP-1," was the only valve included due to the Section XI size exclusion and also had its internal surfaces examined. The staff verified that valve "1LP-1" was included in the second interval inspection scope. From the "ISI Visual Examination VT-3 Form," signed by the Authorized Nuclear Inservice Inspector (ANII), the staff verified that the subject valve had a visual "VT-3" examination which also included an examination of visible threaded parts. Based on the sample reviewed the staff was satisfied that the ISI program for Class 1 valves at Oconee was being implemented in accordance with the ASME Section XI requirements.

Also, the staff verified that the licensee had performed visual "VT-1" examination of pressure retaining bolting 2 inches and less in diameter in accordance with "Examination Category B-G-2" for selected components. The staff reviewed the actual examination reports signed by the ANI for the selected bolting and noted no problems.

(e) Augmented ISI programs

The only augmented ISI applicable to the scope of the on-site review is the HPI nozzle safe end examinations which are addressed in Item (j) below. The staff reviewed Section 7, "Augmented ISI," of the "Oconee ISI Plan General Requirements" volume and verified that no other components were within the scope of this review.

(f) Last system 10-year hydrostatic and last refueling leakage test to verify procedure boundaries versus scope of B&WOG report (for example, is IMS included?)

The staff verified hydrostatic and leakage test boundaries as documented in Item (b) above. The IMS was not specifically verified as part of this on-site review. The staff did verify that the visual "VT-2" examination boundary extended out to and included the second isolation valve and encompassed the Class 1 components for the leakage test. Also, the staff reviewed that the last leakage and hydrostatic tests had been completed on June 22 and 23, 1994, respectively.

(g) For austenitic stainless steel cladding (that is, weld material) verify that ferrite content is equal to or greater than 5 percent.

The staff reviewed the B&W equipment specification for stainless steel welds and weld deposit cladding and noted that it did not support the B&WOG GLRP report claim that cladding had a minimum of 5 percent ferrite since the requirement limits chromium <u>or</u> ferrite. Furthermore, B&W personnel explained that during construction of Oconee, cracks were discovered in the cold leg cladding that were attributed to a faulty manufacturing process. Subsequent testing of the cracked cladding revealed that it contained less than 5 percent ferrite, contrary to what was stated in the B&WOG GLRP report. As a result of this discovery, the manufacturing process was changed, the cracked cladding spool pieces were returned to the manufacturer and redone, and the remaining Oconee cladded piping that was not cracked was liquid penetrant inspected. The staff indicated that these aforementioned details needed to be captured in the subject B&WOG GLRP report in lieu of the assertion that the ferrite content of the cladding was maintained above 5 percent.

(h) Verify existence of stress analysis for:

(1) attachment piping and IMS piping designed to Class 1

(2) "Examination Category B-J" fatigue cumulative usage factors (CUFs) greater than 0.4 or stresses greater than 2.4 times code allowable stress intensity "Sm" and check against ISI program plan

With regard to the attachment piping to the RCS, the licensee explained that on April 27, 1995, they received a letter from the NRC project manager requesting that they perform an evaluation of the attached piping to the RCS to demonstrate compliance with the FSAR criteria of designing to Class 1 standards. The B&WOG report, dated March 1995, implies that all reactor coolant pressure boundary piping is designed to Class 1 standards and this is contradicted by the current situation at Oconee. Therefore, the subject report needs to be updated to reflect the design basis of Oconee as discussed in RAI #3.

For the IMS piping, the staff verified that a Class 1 stress analysis did exist. The staff performed a cursory review of the B&W stress analysis for the IMS piping and noted that the highest fatigue CUF was less than 0.01.

Also the staff verified that the ISI plan for the third inspection interval included "Examination Category B-J" piping welds based on the fatigue usage and stress criteria noted above. The staff examined an internal Oconee memorandum which identified what welds exceeded the aforementioned criteria. Also, the staff reviewed the "Oconee ISI Plan General Requirements" volume and verified that piping welds were being included in the ISI program based on the

fatigue and stress criteria. No hot leg piping to steam generator "B" had welds that exceeded these criteria. The locations effected were the cold leg, pressurizer surge and spray, and the HPI/letdown piping. Since Oconee was not required to utilize these ISI selection criteria for the second 10-year ISI interval which was recently completed, not all of the 39 usage/stress criteria locations identified for the third 10-year ISI interval had been examined. The staff concluded that the usage/stress criteria were being properly utilized and will be fully implemented in the third inspection interval.

(i) Verify use of boric acid corrosion procedures implemented in response to Generic Letter 88-05 relating to when the procedures were last performed and procedure requirements, scope, acceptance criteria, and corrective action.

The staff was aware that the NRC had conducted an audit of Oconee's boric acid leakage monitoring and corrosion prevention program from July 31 through August 2, 1989, and had concluded that an acceptable program was in place although a more formalized program was recommended. From a review of various plant procedures, it was apparent to the staff that the licensee was well aware of the issue and was adequately monitoring the plant for any evidence of RCS leakage and its effects. Additionally, the staff reviewed examples of implemented procedures that demonstrated that evidence of boric acid corrosion/leakage was being documented. The staff was satisfied that specific plant procedures and directives are implemented to detect and mitigate adverse effects from boric acid corrosion/leakage.

(j) Augmented ISI for HPI thermal sleeve

The staff reviewed Section 7 of the "Oconee ISI Plan General Requirements" for the third interval and verified that under Item 7.1.2 that a commitment was made to perform volumetric examinations of the HPI nozzle safe ends. Even though no problems have been experienced with the Oconee Unit 1 HPI thermal sleeves and they are of a different design as those that cracked in service, Unit 1 performs the same augmented ISI examinations as Units 2 and 3 which had to replace their HPI thermal sleeves. The staff reviewed one volumetric test data sheet signed by the ANII for the HPI nozzle safe end and verified that it contained no reportable indications.

In summary, as a result of this on-site review, the B&WOG needed to:

(1) provide a more detailed description of the history of the ferrite content of the pipe cladding (see Item (g)), and

(2) better define the scope of the RCS boundaries (see Items (b) and (h)).

Subsequent to the site visit, the B&WOG provided additional information to adequately address the above two open issues from the on-site review. No open issues remain from the on-site review.