SAFETY EVALUATION

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

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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. After reviewing the RAI response, the staff issued a draft safety evaluation (DSE) on the topical report. Following the issuance of the DSE, the B&WOG representatives responded to the open items in the DSE. Requests for additional information, meeting summaries, DSE, and other correspondence are listed in Appendix A of this safety evaluation.

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 of this safety evaluation. 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. In its January 24, 1996, DSE response, the B&WOG indicated that TLAAs, such as fatigue, will be resolved on a plant-specific basis.

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 January 24, 1996, DSE response, the B&WOG indicated that the report addresses the RCS piping component intended function. The B&WOG also stated, "The component intended function that is applicable to the RCS piping components within the scope of this report is to maintain the pressure boundary so that the RCS may perform its system function(s) within the scope of license renewal in the period of extended operation."

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, with a few exceptions, 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

Response to Bulletin 82-02 on bolting degradation (Reference 5)

Program evaluated in Generic Letter 85-20 on thermal sleeve cracking (Reference 6)

Information resulting from Information Notice 90-10 on Alloy 600 cracking (Reference 7)

Technical specification RCS leakage limits

The exceptions are: augmented inspection of a 9-1/2 inch segment of hot leg containing Alloy 82/182 cladding; augmented inspection of less than 4-inch size RCS piping; and a new procedure to manage the potential for loss of fracture toughness of cast stainless steel valve bodies.

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 8) 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 identify the 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 of this safety evaluation, 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 January 24, 1996, DSE response, the B&WOG indicated that the report addresses the RCS piping component intended function. The B&WOG also stated, "The component intended function that is applicable to the RCS piping components within the scope of this report is to maintain the pressure boundary so that the RCS may perform its system function(s) within the scope of license renewal in the period of extended operation."

In its August 11, 1995, RAI response, the B&WOG provided clarification on the structural integrity of the RCS piping components 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.

In addition, the staff agrees with the B&WOG that the RCS piping component intended function is to be maintained for renewal. This is consistent with Part 54.21(a)(3) of Title 10 of the Code of Federal Regulations which 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.

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. 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.

In B&W plants, a 9-1/2-inch flow meter section of the hot leg is internally clad with Alloy 82/182. In its January 24, 1996, DSE response, the B&WOG indicated that the Alloy 82/182 cladding may be susceptible to cracking. The staff 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. In addition, operating experience has shown that cracked cladding could lead to underlying base metal degradation (Reference 9).

B&W plant high pressure injection branch connections to the cold leg contain stainless steel thermal sleeves. These thermal sleeves have experienced cracking and caused the associated safe-ends to crack. In its January 24, 1996, DSE response, the B&WOG indicated that the high pressure injection thermal sleeves may be susceptible to cracking. On the basis of operating experience at B&W plants, the staff agrees with the B&WOG that the high pressure injection thermal sleeves may be susceptible to cracking.

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 clad 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 10 and 11). 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 B&WOG reviewed information from operating experience of the RCS piping relating to the effects of aging. This review identified the following generic communications concerning the effects of aging of the RCS piping components:

Generic Letter 88-05 on boric acid corrosion (Reference 4)

Bulletin 82-02 and Generic Letter 91-17 on bolting degradation (References 5 and 12)

Generic Letter 85-20 on thermal sleeve cracking (Reference 6)

Information Notice 90-10 on Alloy 600 cracking (Reference 7)

In its January 24, 1996, DSE response, the B&WOG indicated that aging management programs resulting from the above generic communications are necessary for license renewal. The B&WOG also indicated that Bulletin 88-08 on fatigue of attachment piping (Reference 13) and Bulletin 88-11 on fatigue of surge line (Reference 14) are outside the scope of this report because they address fatigue which is a TLAA. The B&WOG stated that TLAAs will be resolved on a plant-specific basis.

The staff finds the B&WOG has properly reviewed information from operating experience of the RCS piping relating to the effects of aging. The staff also finds the B&WOG has identified the appropriate generic communications representing relevant aging effects on the RCS piping components for which the B&WOG is providing aging management programs for the period of extended operation.

3.3.2 Evaluation of Aging Management Programs

The B&WOG evaluated existing programs and found them adequate, with a few exceptions, in managing the effects of aging so that the intended function of the RCS piping components will be maintained consistent with the CLB for any period of extended operation. The exceptions are: augmented inspection of a 9-1/2 inch segment of hot leg containing Alloy 82/182 cladding; augmented inspection of less than 4-inch size RCS piping; and a new procedure to manage the potential for loss of fracture toughness of cast stainless steel valve bodies.

In summary, 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 on boric acid corrosion Program evaluated in Generic Letter 85-20 on thermal sleeve cracking Information resulting from Information Notice 90-10 on Alloy 600 Technical specification RCS leakage limits Augmented inspection of Alloy 82/182 clad hot leg segment Augmented inspection of small bore piping
Valve bodies	ASME Section XI "Examination Categories B-M-1, B-M-2, and B-P" Response to Generic Letter 88-05 on boric acid corrosion 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 on boric acid corrosion Response to Bulletin 82-02 on bolting degradation Technical specification RCS leakage limits

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

<u>ASME Section XI Class 1 ISI</u> - 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. (Reference 3)

<u>Response to Generic Letter 88-05</u> - All pressurized water reactor (PWR) 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. (Reference 4)

<u>Response to Bulletin 82-02</u> - All PWR licensees have responded to the bulletin describing their maintenance procedures for threaded fasteners in the components of the reactor coolant pressure boundary. The staff has inspected licensee programs as part of operating plant activities. (Reference 5)

<u>Program evaluated in Generic Letter 85-20</u> - A B&WOG task force developed a program to manage potential cracking of the B&W high pressure injection nozzle thermal sleeves. B&W plants are implementing the program. The staff has evaluated the program and found it acceptable.

The staff's evaluation is documented in Generic Letter 85-20. (Reference 6)

Information resulting from Information Notice 90-10 - PWR licensees were informed of the potential for primary water stress corrosion cracking of Alloy 600 materials. Licensees have evaluated the applicability of the information to their facilities and considered actions, as appropriate. (Reference 7)

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.

In its January 24, 1996, DSE response, the B&WOG indicated that the ASME Section XI ISI program for the RCS piping components relied on in the topical report is that specified in the 1989 Edition of the ASME Section XI, including mandatory Appendices VII and VIII (1989 Addenda for Appendix VIII). The B&WOG indicated the only exception is "Examination Category B-J" in Section XI which may be conducted in accordance with the 1974 Edition of Section XI as permitted by 10 CFR 50.55a(b)(2)(ii).

The staff finds it appropriate for the B&WOG to identify a specific edition of Section XI for staff review because the code program varies with code editions. The staff also finds the 1989 edition appropriate because this is the latest edition reviewed and endorsed by the staff in 10 CFR 50.55a. Further, the staff finds Appendix VIII provides assurance on the reliability of ultrasonic examinations for managing aging of the RCS piping components during the period of extended operation. In summary, the staff finds the B&WOG reference to the specific ASME Section XI program acceptable.

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, program evaluated in Generic Letter 85-20 on thermal sleeve cracking, information resulting from Information Notice 90-10 on Alloy 600, Technical specification RCS leakage limits, augmented inspection of Alloy 82/182 clad hot leg segment, and augmented inspection of small bore piping, 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 start up 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 is discussed in Section 3.2.1 of this safety evaluation.

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 ASME Section XI ISI program is based on a surface examination of the piping outside surface and leakage detection under "Examination Category B-P." For RCS piping less than or equal to 1 inch, the ASME Section XI ISI program is based 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 plants have extensive leakage monitoring requirements and that corrective actions would be taken to prevent recurrence, 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.

In its January 24, 1996, DSE response, the B&WOG indicated that additional sample inspections of small bore piping are appropriate and will be performed for renewal. Further, the B&WOG indicated that the criteria for additional inspection locations should be based on detailed evaluations of material susceptibility, operating environment, stress, and risk. As a result of Information Notice 90-10 (Reference 7), the B&WOG has instituted a program to catalogue Alloy 600 components at B&W plants and rank their susceptibility to primary water stress corrosion cracking and the B&WOG indicated that this information may be useful in developing the sample inspection program. The B&WOG indicated that either the B&WOG or the industry may provide GLRP member plants with recommendations for the details of the sample inspection program at a later time.

The staff finds the B&WOG commitment to perform additional inspection of small bore RCS piping, that is, less than 4-inch size, for license renewal acceptable because it would provide assurance that potential cracking of small bore RCS piping is adequately managed such that the small bore piping can perform its intended function during the period of extended operation. Because the B&WOG defers developing details of the sample inspection program to the renewal applicant referencing the topical report, the staff will review and approve details of the sample inspection program when a B&WOG GLRP member plant submits its renewal application.

To manage potential cracking of the Alloy 82/182 cladding, the B&WOG, in its January 24, 1996, DSE response, indicated that a one-time volumetric inspection of the Alloy 82/182 clad flow meter section of the hot leg will be performed for renewal at one B&W plant at or near the end of the current license term. The B&WOG also indicated that ultrasonic inspection methods would be used and the calibration block and flaw acceptance criteria would be

in accordance with ASME Section XI Subsection IWB. The B&WOG indicated that it may provide GLRP member plants with recommendations for the details of the inspection program and plant selection for the examination at a later time.

The staff finds the B&WOG commitment to perform a one-time inspection for license renewal of the Alloy 82/182 clad flow meter section of the hot leg acceptable because it would provide information on whether the Alloy 82/182 cladding is not cracked or the underlying base metal is not degraded due to clad cracks. The staff also agrees that 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. Because the B&WOG defers developing details of the inspection program and the plant selection to the renewal applicant referencing the topical report, the staff will review and approve details of the inspection program and plant selection when a B&WOG GLRP member plant submits its renewal application.

To manage potential cracking of the thermal sleeves, the B&WOG has developed a program which has been previously evaluated by the staff as discussed in Generic Letter 85-20 (Reference 6). Under this program, the B&W plant owners have examined the high pressure injection connection thermal sleeves and have replaced those susceptible to cracking 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. In its DSE response dated January 24, 1996, the B&WOG indicated that the program evaluated by the staff in Generic Letter 85-20 is necessary to manage potential cracking of the high pressure injection thermal sleeves during the period of extended operation. The staff finds this acceptable.

In its January 24, 1996, DSE response, the B&WOG indicated that fatigue is a TLAA and will be resolved on a plant-specific basis. The staff recognizes 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 15), 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 (Reference 4) 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 Valve Bodies

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 10 and 11) 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 10 and 11). 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 staff finds the B&WOG's program for managing the loss of fracture toughness of cast stainless steel during the period of extended operation 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, response to Bulletin 82-02, 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.

Further, Bulletin 82-02 (Reference 5) indicates that there should be additional elements of bolting maintenance program, such as maintenance procedures for threaded fasteners and establishment of quality assurance measures for use of lubricants and sealants for connections with threaded fasteners. In its DSE response dated January 24, 1996, the B&WOG indicated that the response to Bulletin 82-02 describes a program necessary for managing the effects of aging of bolts in the RCS during the period of extended operation. The staff finds the programs committed to by the licensees in response to Bulletin 82-02 acceptable in managing the effects of aging of bolting during renewal.

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 the B&WOG report provides an acceptable demonstration that the aging effects of RCS piping components within the scope of this report will be adequately managed such that there is reasonable assurance that the RCS piping 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 referencing this report in a license renewal application, summarizing the aging management programs contained in this report in a FSAR supplement, and completing the action items described in Section 4.1 below, 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 and edition of the ASME Section XI code (including mandatory appendices), 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).

Further, the B&WOG defers the development of details of (1) the inspection of the Alloy 82/182 clad hot leg segment and plant selection for that inspection, and (2) the sample inspection of small bore RCS piping, to the renewal applicant referencing this topical report. The renewal applicant will have to

provide details of these two augmented inspection programs in its renewal application for staff review and approval.

The B&WOG elected to exclude TLAAs applicable to the RCS piping components from the scope of the topical report and indicated that they will be resolved on a plant-specific basis. Thus, any renewal applicant referencing this report will have to evaluate TLAAs applicable to the RCS piping components in its renewal application in accordance with the requirements in 10 CFR 54.21(c).

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

5.0 REFERENCES

1. Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Federal Register, 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. Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," NRC, June 2, 1982.

6. 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.

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

8. 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.

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

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

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

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

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

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

15. 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.

19. Letter to D.K. Croneberger of B&WOG Generic License Renewal Program from D.M. Crutchfield of NRC, dated October 18, 1995, "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."

20. Letter to U.S. NRC, from D.K. Croneberger of B&WOG Generic License Renewal Program, dated January 24, 1996, transmittal of response to open items in the NRC draft safety evaluation.

Appendix B. <u>Summary of On-Site Review</u>

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.