Duke Energy Corporation — Oconee 1, 2, 3
Entergy Operations, Inc. — ANO-1
Florida Power Corporation — Crystal River

The B&W Owners Group

AmerGen Energy Co., LLC. — TMI-1
FirstEnergy Nuclear Operating Co. — Davis-Besse
Framatome Technologies, Inc

Working Together to Economically Provide Reliable and Safe Electrical Power

Suite 525 • 1700 Rockville Pike • Rockville, MD 20852 • (301) 230-2100

April 21, 2000
OG-1788

Project No. 683

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. David B. Matthews
Division of Reactor Program Management

Subject: B&WOG License Renewal Task Force Topical Report BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals"
Transmittal of Approved Version of Topical Report

Reference: Letter from Christopher I. Grimes to William R. Gray, December 9, 1999, entitled "Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled, 'Demonstration of the Management of Aging Effects for the Reactor Vessel Internals,' BAW-2248, July 1997"

Gentlemen:

Enclosed please find twelve (12) copies of BAW-2248A entitled, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals." This is the approved version of this Topical Report and it includes the NRC Safety Evaluation Report that was transmitted by the reference letter.

Please call me at 804-832-2783 if you need any further information.

Sincerely,

WR Gray

W. R. Gray
Project Manager
B&W Owners Group Services

WRG/mcl
Enclosure

A003 1/12

c: R.K. Anand — US NRC/NRR
C.I. Grimes — US NRC/NRR

BAW-2248A
March 2000

THE B&W **OWNERS GROUP**

Licensing Renewal Task Force

Demonstration of the Management of Aging Effects for the

Reactor Vessel Internals

FRAMATOME TECHNOLOGIES

BAW-2248A
March 2000

DEMONSTRATION OF THE MANAGEMENT OF AGING EFFECTS

for the

REACTOR VESSEL INTERNALS

by

R.W. Clark, Entergy Operations, Inc.
F.M. Gregory, FTI

for

B&W Owners Group
Generic License Renewal Program

Duke Power Company
Entergy Operations, Inc.
GPU Nuclear Corporation

(See Section 6 for document signatures)

Framatome Technologies, Incorporated
P. O. Box 10935
Lynchburg, Virginia 24506-0935

NRC FINAL SAFETY EVALUATION

Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," BAW-2248, July 1997

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

December 9, 1999

William R. Gray, Program Director
Generic License Renewal Program
The B&W Owners Group
1700 Rockville Pike, Suite 525
Rockville, MID 20852

SUBJECT: ACCEPTANCE FOR REFERENCING OF GENERIC LICENSE RENEWAL PROGRAM TOPICAL REPORT ENTITLED, "DEMONSTRATION OF THE MANAGEMENT OF AGING EFFECTS FOR THE REACTOR VESSEL INTERNALS," BAW-2248, JULY 1997

Dear Mr. Gray:

The staff of the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation (NRR), has reviewed the topical report entitled, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," BAW-2248, which the Babcock & Wilcox Owners Group (B&WOG) submitted in July 1997 as part of the Generic License Renewal Program (GLRP). The resultant final safety evaluation report (FSER) is transmitted to you as an enclosure to this letter.

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

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

In accordance with the procedures established in NUREG-0390, "Topical Report Review Status," the staff requests that the B&WOG publish the accepted version of BAW-2248 within three months after receiving this letter. In addition, the published version will incorporate this letter and the enclosed FSER between the title page and the abstract.

To identify the version of the published topical report that was accepted by the staff, the B&WOG will include "-A" following the topical report number (e.g., BAW-2248-A).

Sincerely,

CI Grimes

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

Project No. 683

Enclosure: Final Safety Evaluation Report

cc w/enclosure: See next page

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CONCERNING

"DEMONSTRATION OF THE MANAGEMENT OF AGING EFFECTS FOR THE REACTOR VESSEL INTERNALS"

BABCOCK & WILCOX OWNERS GROUP REPORT NUMBER BAW-2248

PROJECT NO. 683

TABLE OF CONTENTS

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CONCERNING

"DEMONSTRATION OF THE MANAGEMENT OF AGING EFFECTS FOR THE REACTOR VESSEL INTERNALS"

BABCOCK & WILCOX OWNERS GROUP REPORT NUMBER BAW-2248

PROJECT NO. 683

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CONCERNING

"DEMONSTRATION OF THE MANAGEMENT OF AGING EFFECTS

FOR THE REACTOR VESSEL INTERNALS"

BABCOCK & WILCOX OWNERS GROUP REPORT NUMBER BAW-2248

PROJECT NO. 683

1.0 INTRODUCTION

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

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

1.1 Babcock & Wilcox Owners Group Topical Report

By letter dated July 29, 1997, the Babcock & Wilcox Owners Group (B&WOG) Generic License Renewal Program (GLRP) submitted topical report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (Ref. 2), for staff review and approval. The purpose of the topical report is to provide a technical evaluation of the effects of aging of the reactor vessel internals and demonstrate that the aging effects within the scope of the report are adequately managed for the period of extended operation associated with license renewal. The topical report provides 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 B&WOG topical report to determine whether the requirements set forth in 10 CFR 54.21(a)(3) and (c)(1) were met. The staff issued requests for additional information (RAIs) after completing the initial review. The B&WOG responded to the staff's RAIs. After reviewing the RAI responses, 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, the DSE, responses to the DSE open items, and other correspondence are listed in Appendix A of this safety evaluation.

For the reasons stated in this safety evaluation report (SER), the staff has found that the B&WOG, as documented in BAW-2248, has (a) properly determined the portion of the reactor vessel internals (RVI) within the scope of license renewal; (b) appropriately treated those structures and components within the scope of license renewal as subject to an AMR except for the thermal shield and the thermal shield upper restraint assembly; (c) identified the applicable RVI component aging effects subject to aging management, with qualifications regarding cracking and reduction of fracture toughness, except for change of dimension (void-swelling); and (d) provided an acceptable AMP for wear. As set forth below, however, the staff has also found that aging management for the other aging effects will have to be addressed on a plant-specific basis by individual applicants for license renewal.

2.0 SUMMARY OF TOPICAL REPORT

The B&WOG topical report, BAW-2248, contains a technical evaluation of aging effects related to B&W reactor vessel internals components, and was provided to the staff to demonstrate that B&WOG member plant owners can adequately manage these effects of aging during the period of extended operation. This evaluation applies to the following B&WOG GLRP member plants:

- Arkansas Nuclear One, Unit 1 (ANO-1)
- Oconee Nuclear Station, Units 1, 2, and 3 (ONS-1, -2, -3)
- Three Mile Island, Unit 1 (TMI-1)

The topical report also contains evaluations of TLAAs, as defined in 10 CFR 54.3, for the reactor vessel internals. However, the topical report indicates that the TLAA of flaw growth acceptance prescribed in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI inservice inspection (ISI) program (Ref. 3) is plant-specific, is not within the scope of the report, and will be resolved on a plant-specific basis.

In addition, the report indicates that inservice inspection programs identified in the ASME Code Section XI may need to be supplemented because certain components are not easily accessible using current technology.

2.1 Components and Intended Functions

2.1.1 Intended Functions

In Section 1.0 of the topical report, the following five intended functions for the Reactor Vessel Internals (RVI), and system components were identified based on the requirements of 10 CFR 54.4(a):

- Provide support and orientation of the reactor core (i.e., the fuel assemblies).
- Provide support, orientation, guidance and protection of the control rod assemblies.
- Provide a passageway for the distribution of the reactor coolant flow to the reactor core.
- Provide a passageway for support, guidance, and protection for the incore instrumentation.
- Provide a secondary core support for limiting the downward displacement of the core support structure in the event of a postulated failure of the core barrel.

An additional function of the RVI, "provide shielding of the reactor pressure vessel," is not covered within the scope of the topical report, as indicated in response to NRC RAI #1 (Ref. 4). The response also indicates that the license renewal applicant should provide justification for a plant-specific conclusion that this is not an intended function. Additional discussion is provided in Section 3.1 of this safety evaluation.

2.1.2 Components

As described in the report, the RVI scope consists of two major structural subassemblies that are located within, but not integrally attached to (i.e., not welded to) the reactor pressure vessel (RPV). These major subassemblies are the plenum assembly (PA) and the core support assembly (CSA). For the purpose of defining materials, fasteners, construction, and assembly, the CSA can be further divided into three principal subassemblies; the core support shield (CSS) assembly, the core barrel assembly (CBA) and the lower internals assembly (LIA). The mechanical fasteners (bolting) joining these subassemblies and associated items are within the scope of this report. The welds within the scope of the reactor vessel internals report include the major structural welds that form or join the major subassembly cylinders and flanges and minor structural welds joining parts such as lifting lugs, support pipes, and tubes to the major subassemblies. There are no pressure-retaining or pressure boundary welds within the scope of this report.

The control rod assemblies (CRA), fuel assemblies (FA), and the incore monitors (IMS) are not considered part of the RVI and are not covered in this report.

The thermal shield and upper thermocouple guide tube assemblies are RVI items; however, it is concluded in the report that they do not perform intended functions as defined in 10 CFR Part 54 and, therefore, are not within the scope of this report. For the thermal shield, plant-specific consideration of the intended function of the RVI by the license renewal applicant would determine if the thermal shield supports an intended function of the RVI; an affirmative determination would result in the plant-specific addition of the thermal shield and the thermal shield upper restraint assemblies to the scope of the IPA (see Section 3.1).

Portions of the internal vent valve assemblies are active components that do not require an aging management review per 10 CFR 54.21.

The surveillance specimen holder tube assemblies (SSHT) are not part of the RVI for the plants included in this report. As such, the SSHT assemblies are not within the scope of this report.

Physical and functional descriptions of the individual items within each of the four principal subassemblies are presented in Sections 2.1 through 2.4 of the topical report.

2.2 Effects of Aging

Section 3.0 of the topical report discusses the aging effects applicable to the reactor vessel internals described above for the period of extended operation for the participating B&W plants.

The topical report states that the following effects of aging could result in adverse impact or loss of any of the reactor vessel internals intended functions:

- cracking (initiation and growth)
- loss of material
- reduction of fracture toughness
- loss of mechanical closure integrity (for bolted connections)

Table 3-2 of the topical report provides a detailed list of the subassemblies in each of the reactor vessel internal assemblies, and identifies the aging effect applicable to each subassembly, as determined by the B&WOG's evaluations. These evaluations included a review of industry operating experience to identify past incidents of aging effects applicable to the reactor vessel internals. This review is discussed in Section 3.5 of the topical report.

The following is a summary of Table 3-2 of the topical report:

Major RVI Assemblies	Applicable Aging Effects
Plenum Assembly	Cracking Loss of material Reduction of fracture toughness Loss of closure integrity

Core Support Shield Assembly	Cracking Loss of material Reduction of fracture toughness Loss of closure integrity
Core Barrel Assembly	Cracking Reduction of fracture toughness Loss of closure integrity
Lower Internals Assembly	Cracking Loss of material Reduction of fracture toughness Loss of closure integrity

2.3 Aging Management Programs

Section 4.0 of the topical report discusses the B&WOG bases for demonstrating that the applicable aging effects identified in Section 3.0 of the topical report can be managed by existing programs at ANO-1; ONS-1, -2, and -3; and TMI-1 during the period of extended operations of those plants. Table 4-1 in the topical report provides a detailed summary of the existing programs that manage aging effects that are applicable to each subassembly of the four major reactor vessel internals assemblies identified above. These programs are the following:

- ASME B&PV Code, Section XI, Inservice Inspection Program
- Reactor Vessel Internals Aging Management Program (RVIAMP)
- Plant Technical Specifications for Vent Valve Bodies in Core Support Shield Assemblies in ANO-1 and TMI-1
- Pump and Valve In-Service Test Programs for Vent Valve Bodies in Core Support Shield Assemblies of ONS-1, -2, and -3

The topical report proposes that the RVIAMP supplement the ASME Section XI ISI program, since the report concludes that the inspection program required by Examination Category B-N-3 of the ASME Section XI program, subsection IWB, may not be adequate to detect aging effects for certain reactor vessel internal components. As described in the topical report, the RVIAMP addresses the specific aging effects of SCC, IASCC, (neutron) irradiation embrittlement and stress relaxation.

2.4 Time-Limited Aging Analyses

Section 4.5 of the topical report identifies the following TLAAs that are applicable to the reactor vessel internals, and presents the B&WOG's proposed aging management programs for each TLAA:

- Fatigue - Cracking (Initiation and Growth)
- Ductility - Reduction of Fracture Toughness

However, the topical report indicates that the TLAA of flaw growth acceptance in accordance with the ASME Section XI ISI program (Ref. 3) is plant-specific, is not within the scope of the report, and will be resolved on a plant-specific basis.

In addition, the report indicates that inservice inspection programs identified in the ASME Code, Section XI, may need to be supplemented because certain components are not easily accessible using current technology.

3.0 STAFF EVALUATION

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

Besides the IPA, Part 54 requires an evaluation of TLAAs in accordance with 10 CFR 54.21(c). The staff reviewed the topical report and additional information submitted by the B&WOG to determine if the evaluations of TLAAs set forth in the report satisfied 10 CFR 54.21(c)(1).

To ensure applicability of the results and conclusions of BAW-2248 to the applicant's plant(s), the license renewal applicant should verify that the critical parameters for the plant(s) are bounded by those used in the report, and should identify any deviations from the aging management programs described as necessary to manage the aging effects identified in the report. This is Renewal Applicant Action Item 1.

In accordance with 10 CFR 54.21(d), each application for license renewal must include an FSAR supplement that provides a summary description of the programs and activities for managing the effects of aging and an evaluation of TLAAs. This is Renewal Applicant Action Item 2.

3.1 Intended Functions

The staff reviewed Sections 1.0 and 2.0 of the topical report to determine whether there is reasonable assurance that the RVI components and supporting structures subject to AMR have been identified in accordance with the requirements of 10 CFR 54.21(a)(1). This was accomplished as described below.

1. As part of the evaluation, the staff determined whether B&WOG has properly identified the systems, structures, and components within the scope of license renewal, pursuant to 10 CFR 54.4. The staff reviewed portions of the Updated Final Safety Analysis Reports (UFSAR) on the RVI for the applicable operating plants, and compared the information in the UFSARs with the information in the report to identify portions of the RVI that the report did not identify as within the scope of license renewal. The staff then reviewed structures and components outside the identified portion, and as described below, requested the B&WOG to provide additional information and/or clarifications for a selected number of structures and components to verify that they do not have any intended functions as delineated in

10 CFR 54.4(a). The staff also reviewed the UFSARs for any safety-related system functions that were not identified as intended functions in the report to verify that structures and components having intended functions were not omitted from consideration within the scope of the rule.

After completing the initial review, by letter dated December 2, 1998 (Ref. 5), the staff issued RAIs regarding the RVI, and by letter dated February 18, 1999 (Ref. 4), the B&WOG provided responses to those RAIs. Section 1.4 of the report identifies the intended functions of the RVI but it does not include, "provide shielding for the RPV" as one of the intended functions. As a result of this omission of an RVI intended function, the components that support this intended function, namely, the thermal shield and the thermal shield upper restraint assemblies were omitted from the scope of license renewal, and from the AMR. NRC RAI # 1 pointed out this omission, and requested clarification. In response, the B&WOG agreed that the function entitled "provide shielding for the RPV" was not included within the scope of the report. Therefore, the B&WOG recommended in its response that each license renewal applicant that references BAW-2248 must determine if the function entitled "provide shielding for the RPV" is an RVI intended function. If not an intended function, the license renewal applicant should provide justification for that conclusion. Should a license renewal applicant determine that the function entitled "provide shielding for the RPV" is an intended function of the RVI, then the items that support this intended function, such as, the thermal shield and the thermal shield upper restraint assemblies, must be identified and reviewed in accordance with 10 CFR 54.21(a)(3). This is Renewal Applicant Action Item 3. The B&WOG also agreed to revise BAW-2248 to provide this clarification. Based on the supporting information in the UFSARs, and the B&WOG's response to the staff's request for additional information, the staff has found no additional omissions in the report and, therefore, concludes that, with the exception of items covered by Renewal Applicant Action Item 3, there is reasonable assurance that the report adequately identified those portions of the RVI and its associated (supporting) structures and components that fall within the scope of license renewal in accordance with 10 CFR 54.4.

As discussed above, the staff has reviewed the information provided in Sections 1.0 and 2.0 of the subject topical report (BAW-2248) and the additional information provided by the B&WOG in response to the staff's RAIs. All structures and components identified in BAW-2248 as being within the scope of license renewal, except for the reactor head vents, are subject to an AMR. The portions of the head vents that are not subject to an AMR perform their functions by changing configuration. Accordingly, the staff concluded that, with the revision described in the previous paragraph, there is reasonable assurance that the report has appropriately identified the portions of the RVI and the associated structures and components thereof, that are within the scope of license renewal and are subject to AMR in accordance with the requirements of 10 CFR Part 54.

3.2 Effects of Aging

As discussed in Section 2.2 above, the effects of aging evaluated in BAW-2248 included reduction of fracture toughness, cracking, loss of material, and loss of mechanical integrity (for bolted connections). The B&WOG reviewed these aging effects for their applicability to the RVI assemblies within the scope of the report. The B&WOG reviewed RVI service history with respect to cracking of weld locations (due to mechanical failure, fatigue and other causes), loss of material (external wall thinning), and loss of closure integrity (wear and erosion). The B&WOG findings about these effects were incorporated into the aging management program.

3.2.1 Cracking

Stress Corrosion Cracking (SCC) and Irradiation-Assisted Stress Corrosion Cracking (IASCC)

The topical report identifies cracking as a potential aging effect due to either SCC or IASCC. SCC results from the synergistic action of a susceptible material subjected to tensile stresses in a corrosive environment, which is specific to the material. The material may be inherently susceptible, or can become sensitized during fabrication. The tensile stresses can be due to the operational loading or residual fabrication stresses. The environmental parameters considered to be critical in SCC are the dissolved oxygen, halide, and sulfide contents. IASCC is a mechanism in which the presence of neutron irradiation can make the material more susceptible to SCC.

For SCC, the report relies on reactor coolant chemistry control, in particular, chemistry control to ensure dissolved oxygen concentration is less than 5 ppb and halide concentration is less than 150 ppb, as the basis for generally ruling out SCC as potentially significant. For RVI bolting applications, the topical report indicates that SCC is a potential aging effect due to the potential for occluded environment conditions in the crevice area typically associated with bolting. The specific applications cited as potentially susceptible to SCC are: core support shield to core barrel bolts, lower internal assembly to core barrel bolts, lower internal assembly to thermal shield bolts, core barrel to thermal shield bolts, shell forging to flow distributor bolts, and Alloy 600 locking devices on the modified vent valve assembly. As discussed in Section 3.3.2 of this SER, additional actions are needed to manage SCC.

For IASCC, the report uses a neutron fluence threshold of 1-2 x 10^{21} n/cm^2 (E>1 MeV) to determine susceptibility to IASCC. The NRC staff has reservations concerning this threshold fluence approach; however, the proposed aging management program will obviate the need for a threshold fluence consideration (see Section 3.3). RVI components determined to be susceptible to IASCC are: core barrel assembly base metal and welds, baffle to baffle and baffle to former bolts, core support shield to core barrel bolts, lower internals assembly to core barrel bolts, and upper and lower grid assembly base metals and welds. Of these components, baffle-former and baffle-baffle bolts are expected to be the first to exhibit the effects of IASCC.

Baffle-Former Bolt Cracking

In Section 3.5 of the BAW-2248 report, the technical evaluation included a review of the historical performance of the reactor vessel internals (RVI) to identify and assess past incidents of aging effects applicable to RVI. The assessment included a review of data from the nuclear plant reliability data system (NPRDS), Licensee Events Reports (LERs), and NRC Generic Letters (GL), Information Bulletins (IE) and Information Notices (IN). In the BAW-2248 report, the B&WOG identified NRC IN 91-05 as providing information regarding cracking in Alloy A-286 bolts used in reactor coolant pumps and the B&W-designed RVI. However, the RVI historical performance review does not include the aging effects applicable to RVI baffle bolting described in the more recently issued NRC IN 98-11, "Cracking of Reactor Vessel Internal Baffle Former

Bolts in Foreign Plants," issued on March 25, 1998. The B&WOG program on baffle-former bolt cracking, as discussed in Section 3.3.2 of this SER, must consider this information.

3.2.2 Loss of Material

Loss of Material Due to Wear

In Section 3.3 of the topical report, the B&WOG identifies the RVI items that are subject to loss of material due to wear. The items identified include the fuel assembly support pads on the upper and lower grid assemblies, the plenum rib pads, the guide blocks on the lower grid, the top flange on the core support cylinder, and the locking devices on the original vent valve assembly. Wear occurs as a result of mating surface contact that may result from flow-induced vibration during plant operation and differential thermal expansion and contraction movements during plant heat-up, cool-down and changes in power operating cycles. The resulting relative movement between the interfacing and mating surfaces causes wear. The severity of the wear depends upon the frequency and duration of the motion, and the loads imposed on the affected surfaces. The cited items are the only RVI components subject to wear covered by the report because these are the only items found in locations of structural interfaces and mating surfaces that experience relative motion during plant operation. The identified RVI items subject to wear require programmatic aging management.

Loss of Material Due to Corrosion

The topical report cites three possible mechanisms for loss of material due to corrosion. These mechanisms are: (1) erosion and erosion-corrosion, (2) uniform attack/general corrosion, and, (3) pitting and crevice corrosion. The staff agrees with the B&WOG's analysis of these mechanisms, as described below for each mechanism:

Erosion and erosion-corrosion are not considered to be applicable since all of the RVI components are fabricated from stainless steel or nickel-base alloys, and these materials have been found to be resistant to erosion and erosion-corrosion.

Uniform attack and general corrosion are not considered to be applicable since all of the RVI components are fabricated from stainless steel or nickel-base alloys, and these materials have been found to be resistant due to protective passivation layers which mitigate the susceptibility of these materials.

Pitting and crevice corrosion are not considered to be applicable due to the low oxygen levels in the reactor coolant as a result of the water chemistry controls.

Therefore, loss of material due to corrosion is not considered to be an applicable aging effect for any of the RVI components.

3.2.3 Reduction of Fracture Toughness

The topical report identifies reduction of fracture toughness in RVI components as an applicable aging effect due to either thermal aging embrittlement or neutron irradiation embrittlement.

Thermal aging embrittlement can occur in cast austenitic stainless steel (CASS) and martensitic stainless steel parts exposed to high temperatures typical of reactor operating conditions. Neutron irradiation embrittlement occurs in all steels due to exposure to high neutron flux conditions typical of many RVI components. Both of these mechanisms result in increased hardness and tensile strength, along with reduced ductility, impact strength, and fracture toughness of the material.

For RVI components fabricated from CASS and hence subject to thermal aging embrittlement, concurrent exposure to high neutron fluence levels can result in synergistic effects wherein the service-degraded fracture toughness is reduced from the levels predicted independently for either of the mechanisms. Therefore, components determined to be subject to thermal aging embrittlement require an additional consideration of the neutron fluence of the component to determine the full range of degradation mechanisms applicable for the component.

Thermal aging embrittlement was determined to be applicable to internal vent valve bodies, vent valve retaining rings, control rod guide tube (CGT) assembly spacer castings, and incore guide tube assembly spider castings. These components are fabricated from CASS, except for the vent valve retaining rings, which are composed of precipitation-hardened stainless steel. In addition, the core support shield outlet nozzles at Oconee Nuclear Station Unit 3 are also composed of CASS (the core support shield outlet nozzles at the other B&WOG GLRP plants are fabricated from wrought stainless steel which is not susceptible to thermal aging embrittlement).

Determination of RVI components subject to neutron irradiation embrittlement was handled in the topical report using a fluence threshold to screen-out components with a neutron fluence level below 5×10^{20} n/cm^2 (E>1 MeV). Components found to be subject to neutron irradiation embrittlement include the upper grid assembly base metal and welds, core support shield to core barrel bolts, core barrel assembly base metal and welds, baffle-baffle and baffle-former bolts, lower internal assembly to core barrel bolts, and the lower grid assembly base metal and welds. The NRC staff has reservations concerning this threshold fluence approach: however, the proposed aging management program, as discussed in Section 3.3 of this safety evaluation, will obviate the need for a threshold fluence consideration.

3.2.4 Loss of Closure Integrity for Bolted Closures

In Section 3.4 of the topical report, the B&WOG indicates that bolting stress relaxation is considered an applicable aging mechanism for those components where maintaining a preload is important to the structural integrity function(s) of the RVI. The B&WOG indicated further that RVI bolts include: the control rod guide tube (CRGT) to upper grid fasteners; the core support shield to core barrel bolts; the core barrel to thermal shield bolts; lower internals assembly to core barrel bolts; the lower grid rib-to-shell fasteners; the shell forging to flow distributor bolts; and the lower internals assembly to thermal shield bolts.

3.2.5 Change of Dimension

Section 3.1 of the topical report dismisses change of dimension of the RVI components due to void swelling as a significant aging effect due to the lack of evidence of void swelling under PWR conditions. However, EPRI TR-107521 (Ref. 6) cites several sources with conflicting results. One source predicts swelling as great as 14 percent for PWR baffle-former assemblies over a 40-year plant lifetime, whereas results from another source indicate that swelling would be less than 3 percent for the most highly irradiated sections of the internals at 60 years. The issue of concern to the staff is the impact of change of dimension due to void swelling on the ability of the RVI to perform their intended functions. The specific impacts of concern are constriction of critical coolant flow paths and interference with control rod insertion. The industry and owners group activities on void selling, as discussed in Section 3.3.6 of this safety evaluation, will address this issue.

3.2.6 Summary

As qualified above, the staff agrees that the B&WOG has identified all applicable RVI component aging effects that are subject to aging management, with the exception of change of dimension. The staff finds that the aging effect of change of dimension due to void swelling should be considered by license renewal applicants. It is the staff's judgment, based on industry experience, that no other aging effects apply to the RVI within the scope of BAW-2248.

3.3 Aging Management Programs

As described in Section 2.3, the aging management programs discussed by the B&WOG include the RVIAMP, ASME Section XI requirements, plant technical specifications, plant-specific test programs and licensee commitments in response to NRC generic communications. Applicants for license renewal will be responsible for describing any such commitments and identifying the appropriate regulatory control.

3.3.1 Reactor Vessel Internals Aging Management Program (RVIAMP)

A key component of the aging management described in the topical report is the addition of the RVIAMP. The generic UFSAR supplement provided as Appendix A to the topical report states that through the RVIAMP an applicant will continue to investigate the potential aging effects identified in BAW-2248, and will establish appropriate monitoring and inspection programs prior to the expiration of the current license. As described in a status report on the B&WOG RVIAMP (Ref. 7), this program is addressing IASCC, SCC, stress relaxation, irradiation embrittlement, irradiation creep and void swelling. As described in the topical report, an applicant would be required to participate in the RVIAMP and to provide to the NRC annual reports or periodic updates (after completion of significant milestones) on the RVIAMP beginning one year after issuance of the renewed license. This is Renewal Applicant Action Item 4.

3.3.2 Cracking

Stress Corrosion Cracking (SCC) and Irradiation-Assisted Stress Corrosion Cracking (IASCC)

Management of SCC and IASCC is achieved through two means. The first means for management of cracking is through the existing inservice inspection (ISI) program, which requires visual VT-3 examination to Examination Category B-N-3 of Section XI of the ASME Code. However, the topical report indicated that this visual examination is only adequate to detect cracking the modified vent valve assembly. Due to accessibility concerns, this examination would not be adequate for other susceptible components. Therefore, the second means to manage cracking is through the Reactor Vessel Internals Aging Management Program (RVIAMP). The purposes of this program are to continue the investigation of the potential aging effects that have been identified in the topical report for the RVI, including cracking in inaccessible locations, and to establish appropriate monitoring and inspection programs that will continue to maintain the intended functions of the RVI for the period of extended operation. Further, in response to NRC RAI #3, the B&WOG indicated that an industry group, the PWR Materials Reliability Project (MRP) is addressing neutron embrittlement of RVI components under the auspices of an RPV Internals Issue Task Group (Ref. 4). The RAI response stated that the results of this MRP activity will be incorporated in the development of the RVIAMP.

The NRC staff proposed to the B&WOG a modified approach to manage cracking of RVI components. In particular, a two-pronged approach was proposed. The two-pronged approach is to perform inspection and/or to perform tests and analysis of irradiated material. The inspection approach would be a supplemental examination of the components believed to be the limiting components for cracking, considering both the susceptibility of the component to the aging mechanism as well as the material properties (in particular the fracture toughness) and the operating stresses on the component. These examinations should be included as part of the 10-year ISI program. This supplemental examination applies to all RVI components except for bolting. Aging management for the limiting RVI bolting, baffle-former bolting, is described below. The test and analysis approach provides for the applicant to evaluate the results of the MRP, or any other industry research activity, to determine the scope of inspection during the period of extended operation.

Since the examination addresses the limiting components, plant-specific neutron fluence evaluations are not necessary. Initial consideration by the B&WOG indicated that the limiting components with respect to highest neutron fluence were the baffle plates and baffle-former bolts.

In response to DSE Open Item 2 (Ref. 8), the B&WOG agreed that an augmented inspection of selected accessible regions of the baffle plates is appropriate to manage cracking (SCC and IASCC) and reduction of fracture toughness during the license renewal period. Details of the augmented inspection (e.g., sample size, examination method and acceptance criteria) are not specified within the topical report, but would be developed by the RVIAMP. Additional details (e.g., timing) of the plant-specific augmented inspections will be addressed at the time of the license renewal application.

Since the topical report concludes that augmented inspection is warranted for cracking (and loss of fracture toughness), each renewal applicant will be required to include plant-specific inspection plans in its application. This is Renewal Applicant Action Item 5.

The second part of the approach for managing cracking would determine the need for continuing the supplemental examination. Should data or evaluations from the MRP or any other industry research activity indicate that the supplemental examinations can be modified or possibly eliminated, each applicant would be required to provide plant-specific justification to demonstrate the basis for the modification or elimination.

Baffle-Former Bolt Cracking

BAW-2248 (Section 3.1 and Table 3-2) identifies baffle-former bolting as a component subject to aging effects. Section 4.1 of the report describes the demonstration of aging management for cracking and identifies baffle-former bolts as the RVI items susceptible to cracking that are subject to programmatic aging management, which require Category B-N-3 examination in accordance with the ASME Code, Section XI, ISI Program, Subsection IWB. Subsection IWB provides requirements for the visual inspection of removable core support structures. Further, the B&WOG indicates that visual inspections may not be adequate to detect bolting cracking that is inaccessible. Therefore, the B&WOG implemented a program to manage the effects of aging due to cracking as described in Section 4.6 of the topical report.

In Section 4.6 of the topical report, the B&WOG outlines generalized activities of the implemented program. One of the defined purposes to implement the aging management program, according to the report, is to establish appropriate monitoring, inspection techniques and inspection programs that will continue to maintain the intended functions of the RVI for the period of extended operation.

During meetings with the staff subsequent to the submittal of BAW-2248, the B&WOG described current and ongoing baffle bolt activities that included preparation for a possible augmented baffle bolt inspection during the next 10-year interval at a B&W lead plant.

By letter dated December 2, 1998 (Ref. 5), NRC forwarded requests for additional information (RAI) with regard to baffle bolt cracking, NRC RAI #12 and RAI #13. NRC RAI #12 requested the B&WOG to describe the baffle bolt inspections that will be conducted prior to the start of the period of extended operation and indicate how these actions provide a basis for assuring that the baffle bolt monitoring and inspection techniques that are planned for implementation during the period of extended operation are appropriate. NRC RAI #13 requested the B&WOG to describe the program that will be implemented as outlined in Section 4.6 of BAW-2248 with regard to the aging management of reactor internals baffle bolts and describe the overall inspection program, including aspects such as intervals, monitoring, and inspection techniques.

By letter dated February 18, 1999, (Ref. 4), the B&WOG provided a response to the RAIs. The responses to NRC RAIs #12 and #13 indicate that the technical elements of the B&WOG RVI aging management program were presented during a meeting with the NRC on April 23, 1998. Since the time of that meeting, the industry has initiated a project to address generic materials issues and the scope of the B&WOG RVI aging management program has changed. The PWR MRP was established during the second quarter of 1998 to address and resolve existing and emerging PWR materials issues. An Issues Task Group (ITG) was formed to manage emerging RVI materials issues. The ITG on reactor vessel internals is currently addressing the issue of cracking, reduction of fracture toughness and loss of preload related to baffle bolts and associated materials. The data and information acquired from these various ITG activities will be used to determine the necessary steps in managing the issues of baffle bolt age-related degradation, including future inspection plans. These plans are expected to be outlined on a plant specific basis, possibly beginning with the inspection at Oconee Unit 1 during their fourth inservice inspection (ISI) interval (Ref. 3).

The renewal applicant will be responsible for using the tools provided by both the ITG and owners groups to determine the necessary steps (e.g., inspections, operability determinations, and replacements) to manage the applicable baffle bolt aging effects. Therefore, the requested information of the B&WOG in RAIs #12 and 13, with regard to the aging management of the effects of baffle bolt age-related degradation, is converted to a renewal applicant action item based on this transfer of responsibility from the B&WOG.
This is Renewal Applicant Action Item 6.

3.3.3 Loss of Material Due to Wear

The B&WOG proposes to continue the ASME Code Section XI program to manage the wear loss of material of RVI items that could cause loss of the RVI function(s) during the period of extended operation. The RVI items are managed by Examination Category B-N-3 of the ASME Code, Section XI, Subsection IWB, which provides requirements for the visual inspection (VT-3) for removable RVI structures. These requirements define conditions in IWB3520.2 which, if detected, must be corrected prior to continued service. These conditions include wear of mating surfaces that may lead to loss of function. Based on industry experience with ASME Code, Section XI, the staff considers these inspections adequate to identify wear. The staff considers the ASME Code Section provisions for inspection adequate for detecting wear for the components within the scope of the topical report.

3.3.4 Reduction of Fracture Toughness

Thermal Aging Embrittlement

In response to Open Item 3 in the draft safety evaluation, the B&WOG described augmented inspection programs for CASS items and precipitation-hardened stainless steel items in the RVI to address loss of fracture toughness by thermal aging embrittlement. The CASS items are the internal vent valve bodies, control rod guide tube (CRGT) assembly spacer castings, and incore guide tube assembly spider castings. In addition, the core support shield outlet nozzles at Oconee Nuclear Station Unit 3 are fabricated from CASS (the core support shield outlet nozzles at the other B&WOG GLRP plants are fabricated from wrought stainless steel which is not susceptible to thermal aging embrittlement). The vent valve retaining rings are fabricated from precipitation-hardened stainless steel.

For the CRGT assemblies, the B&WOG open item response (Ref. 8) indicated that each of the 69 assemblies has 10 spacer castings fabricated from CASS, presumably a static cast process due to the geometry of the items. The specification for the castings was ASTM A 351, Grade CF3M. A review by the B&WOG of chemical composition records for 175 heats of material used to fabricate the spacer castings indicates an average ferrite content of 20 percent, calculated using Hull's equivalent factors. From the susceptibility screening criteria of Ref. 9, this level of ferrite indicates that the spacer castings are potentially susceptible to thermal aging embrittlement.

In addition, the B&WOG has agreed that it is appropriate to manage thermal aging embrittlement for the internal vent valve retaining rings and valve bodies, and incore guide tube assembly spider castings. The core support shield outlet nozzles at Oconee Nuclear Station Unit 3, fabricated from CASS, would also be included in the aging management program.

To manage thermal aging embrittlement of RVI components, the B&WOG proposed augmented inspections of these components, with the details of the augmented inspections (e.g., sample size, examination method, and acceptance criteria) to be determined by the RVIAMP. Each renewal applicant will be required to address the plant-specific plans for these inspections, including timing of the inspections. This is Renewal Applicant Action Item 7.

Neutron Embrittlement

In response to DSE open item 5, the B&WOG linked management of neutron embrittlement to management of cracking, as addressed in Section 3.3.1. Fulfillment of the Renewal Applicant Action Items described in Sections 3.3.1 and 3.3.2, will also address management of neutron embrittlement.

3.3.5 Loss of Closure Integrity for Bolted Closures

In Section 3.4 of the topical report, the B&WOG indicates that bolting stress relaxation is considered an applicable aging mechanism for those components where maintaining a preload is important to the structural integrity function(s) of the RVI. These RVI bolts include: the control rod guide tube (CRGT) to upper grid fasteners; the core support shield to core barrel bolts; the core barrel to thermal shield bolts; lower internals assembly to core barrel bolts; the lower grid rib-to-shell fasteners; the shell forging to flow distributor bolts; and the lower internals assembly to thermal shield bolts.

In Section 4.4 of the topical report, the B&WOG indicates that the required programmatic management for these bolts is the VT-3 visual examination required by Examination Category B-N-3 of the ASME Code, Section XI ISI Program, Subsection IWB. The VT-3 visual examination is specifically designed to determine the general mechanical and structural conditions, including structural distortion and displacements, loose or missing parts, and wear of mating surfaces that may lead to the loss of integrity at bolted connections. IWB-3142 provides options for correcting the relevant condition(s), such as: (1) acceptance by supplemental surface and/or volumetric examination, in order to further characterize the condition; (2) acceptance by corrective measures (i.e., re-establishing the preload) or repairs; or (3) acceptance by replacement of the item. However, the B&WOG indicates that it recognizes that visual examination may not be adequate to detect the loss of mechanical closure integrity of RVI, and the GLRP has implemented a program to manage these aging effects as discussed in Section 4.6 of the topical report.

In Section 4.6 of the report, the B&WOG indicates that the RVIAMP will provide assurance that the RVI can perform their component intended function(s) for the period of extended operation associated with license renewal. The B&WOG indicates the purpose of the program is to: (1) continue the investigation of the aging effects; and (2) establish the appropriate monitoring and inspection programs to continue to maintain the RVI functions. The B&WOG indicates the elements of the program that will be implemented to address stress relaxation of RVI bolting are:

(1) to determine critical locations; and (2) establish appropriate monitoring and inspection techniques.

In the report, the B&WOG indicates that the updated FSAR supplement and the license renewal applications will include a commitment to implement this AMP and to notify the NRC staff regarding the status of the program activities on a regular basis.

Each renewal applicant will be required to address the plant-specific plans for this AMP. This is Renewal Applicant Action Item 8.

3.3.6 Change of Dimension

In response to DSE Open Item 1 (Refs. 8 and 10), the B&WOG indicated that industry and owners group activities are addressing change of dimension due to void swelling. The B&WOG will evaluate the available data and determine the effect of void swelling on the functionality of RVI components. The B&WOG committed to continue participation in the industry activities, and to perform design-specific analyses, as necessary. The B&WOG has stated that, based on currently available data, they conclude that there are no RVI locations that may be adversely impacted by void swelling, and further, that current ASME Section XI inspection requirements (Examination Category B-N-3, VT-3) would be effective in identifying the occurrence of gross deformation by void swelling. Therefore, the B&WOG proposes no additional aging management beyond the current ASME Code requirements.

However, as discussed in Section 3.2.5 of this SER, the evidence shows that void swelling may be significant. Further, the staff does not believe that the visual VT-3 requirements of Section XI are adequate for the detection of void swelling, due to the fact that the VT-3 examinations are conducted to determine the general mechanical and structural condition of components, and to detect discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear or erosion. The effects of void swelling are not expected to provide such gross indications. Therefore, renewal applicants should develop supplemental management programs (AMP) for void swelling. An adequate AMP would include participation in industry program(s) to address the significance of void swelling (either individually or through an owners or industry group), a commitment to develop a sufficient inspection program (including the basis, methods, locations to be examined, timing, frequency and acceptance criteria) for management of the issue based upon the results of the industry programs, and a commitment to implement the inspection program prior to the end of the current license period.

Should industry programs determine that void swelling is not a significant issue in the renewal term, then licensees would not need to develop nor implement any additional inspections beyond the current ASME Code Section XI VT-3 visual inspection of core support structures (Examination Category B-N-3 of Table IWB-2500-1).

Aging management of void swelling is Renewal Applicant Action Item 9.

3.4 Time-Limited Aging Analyses

Time-limited aging analyses are defined in 10 CFR 54.3 as those licensee calculations and analyses that:

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

(2) consider the effects of aging;

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

(4) were determined to be relevant by the licensee in making a safety determination;

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

(6) are contained or incorporated by reference in the current licensing basis.

Paragraph 54.21(c)(1) requires the applicant to demonstrate that:

(i) the analyses remain valid for the period of extended operation;

(ii) the analyses have been projected to the end of the period of extended operation; or

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

In Section 4.5 of the topical report, the B&WOG identified the TLAAs applicable to the RVI items within the scope of the report based on reviewing plant-specific docketed correspondence files, plant-specific FSARs, BAW topical reports, and the ASME Code, Section XI ISI requirements. The topical report identifies four TLAAs appplicable to the RVI:

- flow-induced vibration endurance limit assumptions
- transient cycle count assumptions used for the design of the RVI and used in the fatigue cumulative usage factor calculations
- the effects of neutron irradiation on RVI material properties and deformation limits
- flaw growth acceptance in accordance with the ASME Code, Section XI ISI requirements

The first two of these TLAAs are lumped together in the topical report under the heading "Fatigue - Cracking (Initiation and Growth)," and the third is referred to in the topical report as "Ductility - Reduction of Fracture Toughness." The flaw growth acceptance TLAA is identified in the topical report as requiring a plant-specific evaluation, and as such is not evaluated within the topical report. This is Renewal Applicant Action Item 10.

All of the TLAA analysis documents were identified in the topical report as required by 10 CFR 54.21(c)(1).

3.4.1 Fatigue - Cracking (Initiation and Growth)

The flow-induced vibration endurance limit assumptions were based on 10^{12} cycles for 40 years. The analysis was extended for the license renewal period by conservatively increasing the number of cycles to 10^{13}, and then determining the endurance limit using the latest ASME fatigue curves. The component stress values were found to be less than the endurance limit, rendering the evaluation acceptable, in conformance with the requirements of 10 CFR 54.21(c)(1).

In the topical report, the B&WOG indicates that the design cyclic loadings and thermal conditions used for the analyses are defined in the component design specifications and that the flow-induced vibration input used was obtained from hot functional testing data contained in the listed analyses documents. The ability to withstand cyclic loading without fatigue failure was evaluated using a cumulative usage factor methodology. For each facility, the number of transients accrued to date was conservatively extrapolated, and in all cases it was found that the number of design cycles would not be exceeded in the period of extended operation. The B&WOG reported that each of the participating utilities monitors occurrences of design transients and is thus managing the potential for cracking resulting from fatigue. The topical report indicates that the plants must continue to monitor and track occurrences of design transients. Each renewal applicant will be required to address the plant-specific plans for continuation of this monitoring and tracking. This is Renewal Applicant Action Item 11.

3.4.2 Ductility - Reduction of Fracture Toughness

Section 4.5.2 of BAW-2248 describes a TLAA related to the acceptability of the reactor vessel internals under loss-of-coolant-accident (LOCA) and seismic loading. The topical report states that Appendix E to BAW-10008, Part 1, Rev. 1, concludes "that at the end of 40 years, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits." The topical report indicates that this TLAA will be resolved on a plant-specific basis per 10 CFR 54.21(c)(1)(iii) based on the results and conclusions of the planned RVIAMP.

The aging management approach proposed for neutron irradiation embrittlement (Section 3.3.4) includes examination for cracking as the primary management method for RVI components. This examination is effective in managing neutron embrittlement as it relates to the fracture resistance of the RVI component material in the presence of flaws. The deformation limits and adequate ductility described above in Appendix E to BAW-1 0008, Part 1, Rev. 1, relates to the material behavior in the absence of flaws. Therefore, no examination looking for flaws will be sufficient to demonstrate resolution of this TLAA, which will require determination of the expected material properties at the end of the license renewal period. As described in the topical report, the planned RVIAMP program will provide the data necessary to resolve this TLAA.

Therefore, this item should be addressed as a renewal applicant action item on a plant-specific basis. This is Renewal Applicant Action Item 12.

4.0 CONCLUSIONS

The staff has reviewed the subject B&WOG topical report (Ref. 2) and additional information submitted by the B&WOG. On the basis of its review, the staff concludes that the B&WOG topical report provides an acceptable demonstration that the aging effects of reactor vessel internal components within the scope of this topical report will be adequately managed for the GLRP member plants with the exception of the noted renewal applicant action items. Upon resolution of the applicant action items with respect to its facility, a GLRP member applicant may rely on BAW-2248 to demonstrate that there is reasonable assurance that the reactor vessel internal components will perform their intended functions in accordance with the CLB. The staff also concludes that, upon completion of the renewal applicant action items set forth in Section 4.1 of this safety evaluation, the B&WOG topical report provides an acceptable evaluation of time-limited aging analyses for the reactor vessel internals for the GLRP member plants for the period of extended operation.

Any B&WOG GLRP member plant may reference this topical report in a license renewal application to satisfy the requirements of (1) 10 CFR 54.21(a)(3) for demonstrating that the effects of aging on the reactor vessel components within the scope of this topical report will be adequately managed and (2) 10 CFR 54.21(c)(1) for demonstrating that appropriate findings be made regarding evaluation of time-limited aging analyses for the reactor vessels for the period of extended operation. The staff also concludes that, upon completion of the renewal applicant action items set forth in Section 4.1 below, referencing this topical report in a license renewal application and summarizing in an FSAR supplement the aging management programs and the TLAA evaluations contained in this topical report will provide the staff with sufficient information to make the necessary findings required by Sections 54.29(a)(1) and (a)(2) for components within the scope of this topical report.

4.1 Renewal Applicant Action Items

The following are license renewal applicant action items to be addressed in the plant-specific license renewal application when incorporating B&WOG topical report BAW-2248 in a renewal application:

(1) The license renewal applicant is to verify that the critical parameters for the plant are bounded by the topical report. Further, the renewal applicant is to commit to programs described as necessary in the topical report to manage the effects of aging during the period of extended operation on the functionality of the reactor vessel components. The applicant for license renewal will be responsible for describing any such commitments and proposing the appropriate regulatory controls. Any deviations from the aging management programs within this topical report described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel internal components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

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

(3) The license renewal applicant must identify whether an intended function of the RVI is to provide shielding for the RPV. If not an intended function, the license renewal applicant should provide justification for that conclusion. Should a license renewal applicant determine that the RVI's intended function is to provide shielding for the RPV, then the items that support this intended function, such as, the thermal shield and the thermal shield upper restraint assemblies, must be identified and reviewed in accordance with 10 CFR 54.21(a)(3).

(4) The applicant must commit to participation in the B&WOG RVIAMP, and any other industry programs as appropriate, to continue the investigation of potential aging effects for RVI components and to establish monitoring and inspection programs for RVI components. The applicant shall provide the NRC with either annual reports or periodic updates (after completion of significant milestones) on the status of the RVIAMP, commencing within one year of the issuance of the renewed license.

(5) The applicant must describe plans for augmented inspection of RVI components for management of SCC/IASCC and loss of fracture toughness (neutron embrittlement) of the RVI components. This description should specify the sample size, the examination method, acceptance criteria and timing of the inspection, or the process to be used to specify these items.

(6) According to the B&WOG, one of its objectives in BAW-2248 states, "it is intended that NRC review and approval of this report will allow that no further review of the matters described herein will be needed when the report is incorporated by reference in a plant specific renewal license application." The license renewal applicant must address the baffle-former bolt cracking issues addressed in Section 3.3.2 of this SE pertaining to Refs. 4 and 5, with regard to the industry ITG project, initiated after April 23, 1998, to address generic RVI materials issues. The B&WOG indicates this industry effort resulted in subsequent changes in the B&WOG RVI aging management program. The ITG is currently addressing the issues of cracking of baffle bolts. The B&WOG indicates that the changes in the aging management program now requires the applicants to be responsible for using the industry ITG project developed information to determine the necessary steps (e.g., inspection, operability determinations, and replacements) for the management of the applicable baffle bolt aging effects.

(7) The applicant must describe plans for augmented inspection of RVI components for management of loss of fracture toughness due to thermal aging embrittlement of the RVI components. This description should specify the sample size, the examination method, acceptance criteria and timing of the inspection, or the process to be used to specify these items.

(8) The applicant must describe plans for management of stress relaxation for bolted closures of the RVI. This description should specify the critical locations, and monitoring and

inspection techniques, and timing of the inspection, or the process to be used to specify these items.

(9) The applicant must address aging management of void swelling. An adequate aging management program (AMP) would include participation in industry program(s) to address the significance of void swelling (either individually or through an owners or industry group), a commitment to develop a sufficient inspection program (including the basis, methods, locations to be examined, timing, frequency and acceptance criteria) for management of the issue based upon the results of the industry programs, and a commitment to implement the inspection program prior to the end of the current license period.

(10) If flaws have been detected in the reactor vessel internals, a TLAA plant-specific evaluation must be performed to determine the flaw growth acceptance in accordance with the ASME B&PV Code, Section XI, inservice inspection requirements.

(11) The applicant must address the plant-specific plans to continue monitoring and tracking design transient occurrences.

(12) Plant-specific analysis is required to demonstrate that, under loss-of-coolant-accident (LOCA) and seismic loading, the internals have adequate ductility to absorb local strain at the regions of maximum stress intensity and that irradiation accumulated at the expiration of the renewal license will not adversely affect deformation limits. The RVIAMP must develop data to demonstrate that the internals will meet the deformation limits at the expiration of the renewal license.

5.0 REFERENCES

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Federal Register, Vol. 60, No. 88, pp. 22461-22495, May 8, 1995.

2. BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," Babcock & Wilcox Owners Group, July 1997.

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

4. Letter from William R. Gray (B&WOG) to David B. Mathews (NRC), dated February 18, 1999, "B&WOG Generic License Renewal Program Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (RAIs 1 through 14 from December 4, 1998)."

5. Letter from Raj K. Anand (NRC), to David J. Firth (B&WOG), December 2, 1998, "Request for Additional Information Regarding the Babcock & Wilcox Owners Group Generic License Renewal Program Topical Report Entitled Demonstration of the Management of Aging Effects for the Reactor Vessel Internals, BAW-2248, July 1997."

6. EPRI Technical Report TR-1 07521, "Generic License Renewal Technical issues summary," Electric Power Research Institute, April 1998.

7. Enclosure to letter from P. K. Goyal (B&WOG) to J. L. Birmingham (N RC), dated May 10, 1999, OG-1753, "Status Report of B&W Owners Group Reactor Vessel Internals Program."

8. Letter from William R. Gray (B&WOG) to David B. Mathews (NRC), dated June 29, 1999, "B&WOG Generic License Renewal Program Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" - Response to DSE Topical Report Open Items."

9. EPRI Technical Report TR-1 06092, "Evaluation of Thermal Aging Embrittlement for Cast Austenitic Stainless Steel Components in LWR Reactor Coolant Systems," Electric Power Research Institute, September 1997.

10. Letter from William R. Gray (B&WOG) to David B. Mathews (NRC), dated August 16, 1999, "B&WOG Generic License Renewal Program Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" - Response to DSE Topical Report Open Item 1 Regarding Void Swelling."

APPENDIX A

LIST OF CORRESPONDENCE

1. Letter from David J. Firth (B&WOG) to Marylee Slosson (NRC), July 29,1997, transmitting B&WOG Generic License Renewal Program Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," July 1997.

2. Letter from Raj K. Anand (NRC) to David J. Firth (B&WOG), December 2, 1998, "Request for Additional Information Regarding the Babcock & Wilcox Owners Group Generic Licensee Renewal Program Topical Report Entitled Demonstration of the Management of Aging Effects for the Reactor Vessel Internals, BAW-2248, July 1997."

3. Letter from William R. Gray (B&WOG) to David B. Mathews (NRC), February 18, 1999, "B&WOG Generic License Renewal Program Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (RAIs 1 through 14 from December 4 1998)."

4. NRC Meeting Summary dated May 6, 1998, entitled " Summary of Meeting on April 23, 1998, Between the U.S. NRC Staff and B&WOG Representatives to Discuss the Status of the B&WOG Generic License Renewal Program."

5. Letter from Raj K. Anand to William R. Gray, May 26, 1999, Draft Safety Evaluation Concerning the B&WOG License Renewal Program Topical Report entitled "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," BAW-2248, July 1997.

6. Letter from William R. Gray (B&WOG) to David B. Mathews (NRC), dated June 29, 1999, "B&WOG Generic License Renewal Program Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" - Response to DSE Topical Report Open Items."

7. Letter from William R. Gray (B&WOG) to David B. Mathews (NRC), dated August 16, 1999, "B&WOG Generic License Renewal Program Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" - Response to DSE Topical Report Open Item 1 Regarding Void Swelling."

ACKNOWLEDGMENTS

Preparation of this topical report was performed under the direction of the B&W Owners Group Generic License Renewal Program (GLRP), with technical support provided by the GLRP Project Engineers: Greg Robison, Duke Power Company; Dave Masiero, GPU Nuclear Corporation; and Garry Young, Entergy Operations, Inc. In addition, the GLRP wishes to thank the B&WOG Materials Committee and the many others from the nuclear industry who contributed to the development of this report.

ABSTRACT

The B&WOG reactor vessel internals report provides a technical evaluation of the effects of aging on the reactor vessel internals, including the plenum assembly, the core support shield assembly, the core barrel assembly, and lower internals assemblies. The reactor vessel internals evaluation applies to the following B&W operating plants: Arkansas Nuclear One Unit 1 (ANO-1), Oconee Nuclear Stations 1, 2, and 3 (ONS-1, 2, 3), and Three Mile Island Unit 1 (TMI-1). This report, developed through the B&W Owners Group Generic License Renewal Program, will support individual utility applications for license renewal.

This report demonstrates that existing utility programs adequately monitor and manage the relevant effects of aging of the reactor vessel internals and permit continued safe operation over the extended operating period associated with license renewal. Where existing programs are not adequate, actions have been identified and will be taken with respect to managing effects of aging during the period of extended operation.

ACRONYMS AND ABBREVIATIONS

AMS...............Aerospace Material Specification
ANO-1............Arkansas Nuclear One Unit 1
ANSI ....American National Standards Institute
ASME ....American Society of Mechanical Engineers
BL ....Bulletin (from the NRC)
B&PV............Boiler & Pressure Vessel
B&W..............Babcock & Wilcox
B&WOG ....Babcock & Wilcox Owners Group
BWNT............B&W Nuclear Technologies
CASS ....Cast Austenitic Stainless Steel
CBA ....Core Barrel Assembly
cc....................cubic centimeters
CFR ....Code of Federal Regulations
CLB................Current Licensing Basis
CRA ....Control Rod Assembly
CRDM............Control Rod Drive Mechanism
CRGT............Control Rod Guide Tube Assemblies
CE ....Combustion Engineering (now ABB-CE)
CS...................Carbon Steel
CSA................Core Support Assembly
CSS ....Core Support Shield Assembly
EPRI...............Electric Power Research Institute
FA ....Fuel Assemblies
FIV ....Flow-Induced Vibration
ft.....................Foot or Feet
GL ....Generic Letter (from the NRC)
GLRP ....Generic License Renewal Program
GPM...............Gallons Per Minute
HAZ ....Heat Affected Zone
HFT................Hot Functional Test(ing)
IASCC............Irradiation-Assisted Stress Corrosion Cracking
ID. ....Inside Diameter
IE....................Irradiation Embrittlement
IGA ....Intergranular Attack
IGSCC............Intergranular Stress Corrosion Cracking
IMS ....Incore Monitoring System
IN ....Information Notice (from the NRC)
IR....................Industry Report (a EPRI/NUMARC/NEI document)
ISI...................Inservice Inspection
IST..................Inservice Testing (or Integrated System Test)
kg....................kilogram
LAS ....Low Alloy Steel

LER................Licensee Event Report
LIA.................Lower Internals Assembly
LOCA.............Loss of Coolant Accident
NDE ....Nondestructive Examination
NEI.................Nuclear Energy Institute (formerly NUMARC)
NPRDS...........Nuclear Plant Reliability Data System
NPS ....Nominal Pipe Size
NRC ...............U.S. Nuclear Regulatory Commission
NUMARC......Nuclear Management And Resources Council (now NEI)
OD..................Outside Diameter
ONS................Oconee Nuclear Station
PA ....Plenum Assembly
PPB ....Parts per Billion
PT...................Penetrant Test
PWHT ....Post-Weld Heat Treatment
PWR...............Pressurized Water Reactor
PPM ....Parts per Million
PSI..................Pounds per Square Inch
PWSCC..........Primary Water Stress Corrosion Cracking
RCS................Reactor Coolant System
RT ....Radiographic Test
RV ....Reactor Vessel
SCC................Stress Corrosion Cracking
SS...................Stainless Steel
SSHT..............Surveillance Specimen Holder Tube
TLAA.............Time-Limited Aging Analysis
TMI-1 ....Three Mile Island Unit 1
VT ....Visual Test

DEFINITIONS

Casting is an item at or near finished shape obtained by solidification of a substance in a mold.

Crud is corrosion products which develop within the RCS and become radioactive due to direct radiation or indirectly through exposure to the reactor coolant.

Forging is plastically deforming metal, usually hot, into a desired shape by means of localized compressive forces exerted by presses, special forging machines, or by manual or power hammers.

Fracture Toughness is a generic term for measures of resistance to extension of a crack.

Intended Functions are those functions of a system, structure, or component which satisfy the requirements established in 10 CFR 54 for functions within the scope of license renewal.

Level A Service Conditions (Normal Conditions) are any condition in the course of system start-up, operation in the design power range, hot standby, and system shutdown other than upset, emergency, or faulted conditions.

Level B Service Conditions (Upset Conditions) are any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. Upset conditions include those transients that result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power. Upset conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair or mechanical damage. That is, deviations from normal operations that result from a malfunction or failure of a component which requires isolation but allows for continual operation under restricted conditions are classified as upset. Examples of upset transients for B&W-designed plants include step load reduction, reactor trip, rapid depressurization, change of reactor coolant flow, rod withdrawal, rod drop, loss of station power, and loss of feedwater flowrate.

Level C Service Conditions (Emergency Conditions) are those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of damage developed in the system. Examples of emergency events for B&W-designed plants include a stuck open steam bypass valve and steam generator tube rupture.

Level D Service Conditions (Faulted Conditions) are those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities. Specifically, deviations from normal operations that do not allow the further operation of the plant due to public health or safety concerns are classified as faulted conditions. A faulted condition has a very low probability of occurrence. Examples of faulted events for B&W-designed plants include steam line breaks and loss-of-coolant accidents.

Nozzles are manufactured vessel penetrations, often with thermal sleeves or other features, which are welded to the vessel shell. A nozzle does not have any special fluid flow directing/modifying properties or geometry. Nozzles are attached to vessels and are not part of piping systems.

Ratcheting is a progressive incremental inelastic deformation or strain which can occur in a component that is subjected to variations of mechanical and/or thermal stress.

Regulated Programs are those inservice inspection programs prescribed for implementation through 10 CFR 50.55a, plant Technical Specifications, or other regulatory directives.

Schedule is a standardized coding for piping wall thickness per ANSI/ASME B36.10 and B36.19.

Test Conditions are defined as those tests including hydrostatic tests, high pressure injection tests, and hot functional tests.

CONTENTS

CONTENTS (Continued)

LIST OF FIGURES

LIST OF TABLES

1. INTRODUCTION

1.1 Background

The reactor vessel internals in B&W operating plants have been reviewed in accordance with 10 CFR 54 [1]. The reactor vessel internals evaluation applies to the following B&W operating plants: Arkansas Nuclear One Unit 1 (ANO-1), Oconee Nuclear Station Units 1, 2, and 3 (ONS-1,-2,-3), and Three Mile Island Unit 1 (TMI-1).

One of the objectives of the B&W Owners Group Generic License Renewal Program (GLRP) is to perform generic activities that will minimize the resource requirements for the individual owners and thus limit the scope of work performed on a plant-specific basis. It is intended that the NRC review and approval of this report will allow that no further review of the matters described herein will be needed when the report is incorporated by reference in a plant-specific renewal license application.

Time limited aging analyses (TLAA) applicable to the scope of reactor vessel internals and defined by 10 CFR 54.21 include transient cycle count assumptions used in fatigue usage factor calculations, flaw growth acceptance under the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI, flow-induced vibration endurance limit assumptions, and neutron embrittlement with respect to faulted loadings. Transient cycle count assumptions used in fatigue cumulative usage factor calculations, flow-induced vibration endurance limit assumptions, and neutron embrittlement are addressed in Section 4.5. Flaw growth acceptance in accordance with the requirements of Section XI will be addressed on a plant-specific basis.

1.2 Purpose

The purpose of the report is to demonstrate that the aging effects for the reactor vessel internals within the scope of this report are adequately managed for the period of extended operation associated with license renewal. The original issue of this report was revised substantially through interaction with the NRC between May 1998 to December 1999, as documented in the Request for Additional Information (RAI) (Appendix B), GLRP responses to the RAIs (Appendix C), the NRC Draft Safety Evaluation (DSE) (Appendix D), and GLRP responses to the DSE (Appendix E). The purpose of this revision of the RV internals report is to update the document in accordance with documented correspondence between the NRC and the GLRP. Utilities that take credit for this report in their plant-specific license renewal application need to address the action items as listed in Section 4.1 of the NRC final safety evaluation.

1.3 Scope

The scope of this report includes all major structural components, henceforth referred to as "reactor vessel internals", that are located within, but not integrally attached to, (i.e., not welded to) the reactor vessel, (RV). The reactor vessel internals consists of two major sub-assemblies: the plenum assembly (PA) and the core support assembly (CSA). The CSA can be subdivided into three principle sub-assemblies: the core support shield assembly (CSS), the core barrel assembly (CBA) and the lower internals assembly (LIA). The fuel assemblies (FA), control rod assemblies (CRA), and incore instrumentation (IMS) are not part of the reactor vessel internals and are not evaluated in this report.

The reactor vessel internals scope is discussed further in Section 2.0. The reactor vessel internals general arrangement is shown in Figure 1-1. Figure 1-2 is a fabrication flow diagram depicting how the various sub-assemblies of the reactor vessel internals were assembled.

1.4 Reactor Vessel Internals Component Intended Functions

The reactor vessel internals are a component in the reactor coolant system (RCS). There are five reactor vessel internals component intended functions that are within the scope of license renewal:

1) Provide support and orientation of the reactor core (i.e., the fuel assemblies).

2) Provide support, orientation, guidance and protection of the control rod assemblies.

3) Provide a passageway for the distribution of the reactor coolant flow to the reactor core.

4) Provide a passageway for support, guidance, and protection for the incore instrumentation.

5) Provide a secondary core support for limiting the downward displacement of the core support structure in the event of a postulated failure of the core barrel.

Maintaining the structural integrity of the reactor vessel internals within the scope of this report will assure that their component intended functions are maintained within the scope of license renewal in the period of extended operation. Maintaining the structural integrity of the reactor vessel internals through management of aging effects will assure that the RCS system function(s) within the scope of license renewal will not be impacted by the aging of these components.

1.5 Approach

The approach for demonstrating management of aging effects on the reactor vessel internals is to define applicable aging effects for the reactor vessel internals items and then to determine how existing regulated programs and commitments manage those effects. The aging effects that could challenge the reactor vessel internals functions that are within the scope of license renewal are identified in Section 3.0. In addition, Section 3.0 contains an overview of industry reactor vessel internals performance history to identify past incidents of the applicable aging effects.

The identified aging effects are evaluated against existing plant programs in Section 4.0. The demonstration process is complete when the evaluation of existing programs demonstrates that the applicable aging effects will be managed for the period of extended operation. If existing programs cannot be shown to demonstrate that aging effects will be managed for the period of extended operation, then action will be taken, when appropriate, to manage the identified effects of aging.

Figure 1-1 Reactor Vessel Internals (Cross Section)

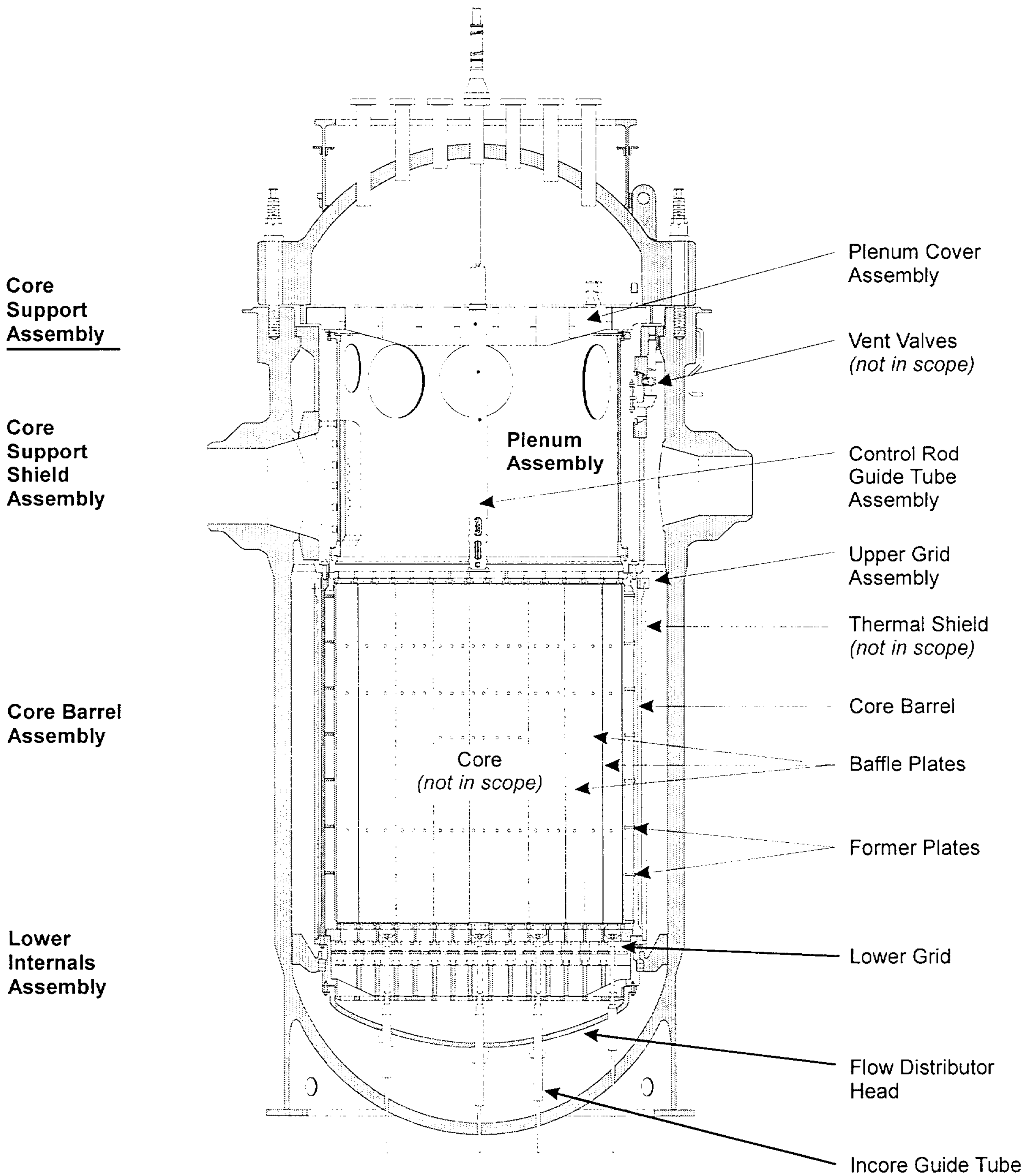
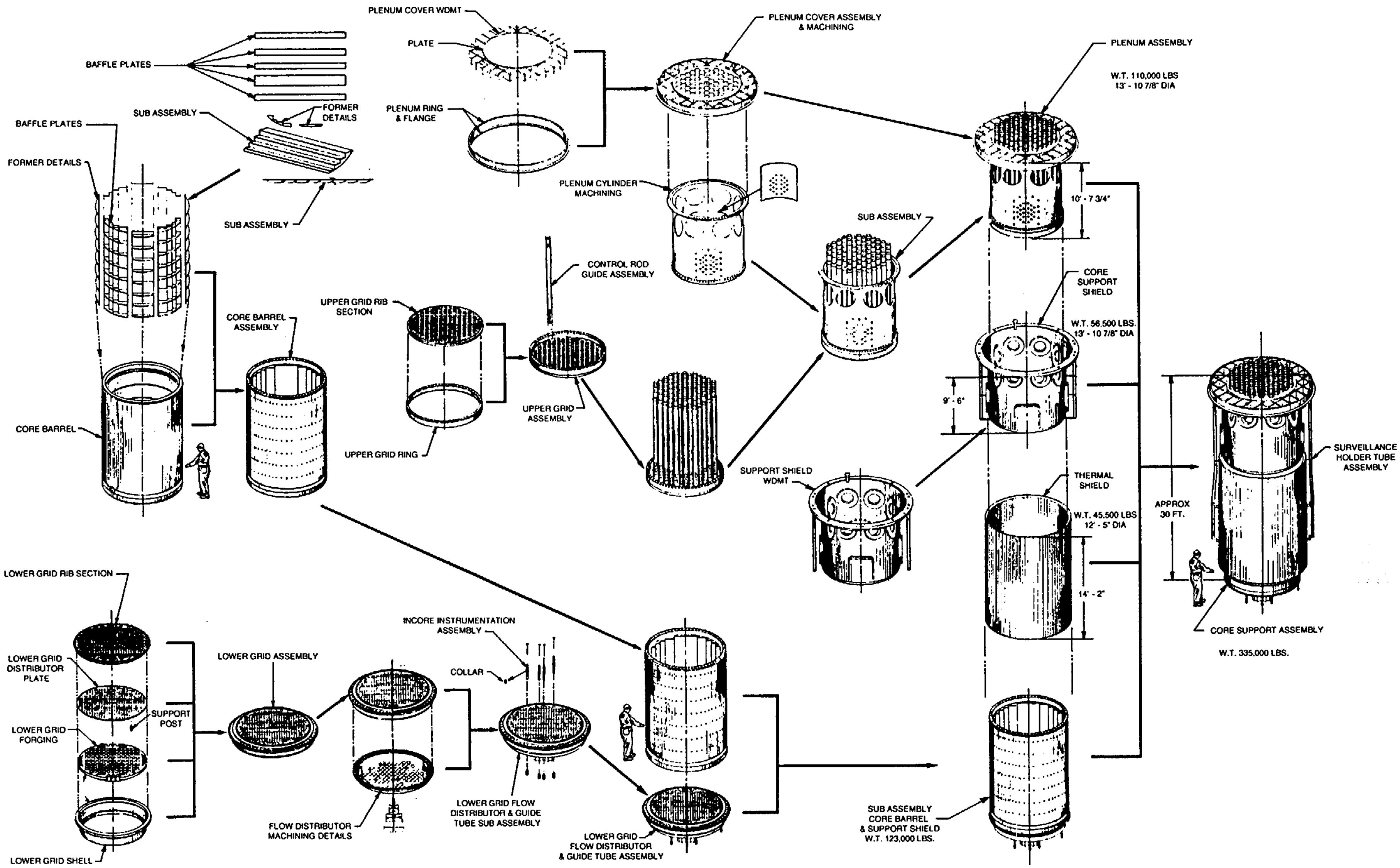


Figure 1-2 Reactor Vessel Internals Fabrication Flow Diagram



2. REACTOR VESSEL INTERNALS SCOPE

The reactor vessel internals scope consists of two major structural sub-assemblies that are located within, but not integrally attached to (i.e., not welded to) the reactor vessel (RV). These major sub-assemblies are the plenum assembly (PA) and the core support assembly (CSA). For the purpose of defining materials, fasteners, construction, and assembly, the CSA can be further divided into three principle sub-assemblies: the core support shield assembly (CSS), the core barrel assembly (CBA) and the lower internals assembly (LIA). The mechanical fasteners (bolting) joining these sub-assemblies and associated items are within the scope of this report. The welds within the scope of the reactor vessel internals report include the major structural welds that form or join the major sub-assembly cylinders and flanges and minor structural welds joining parts such as lifting lugs, support pipes and tubes to the major sub-assemblies. There are no pressure-retaining or pressure boundary welds within the scope of this report.

The control rod assemblies (CRA), fuel assemblies (FA), and the incore monitoring system (IMS) are not considered part of the reactor vessel internals and are not covered in this report. The thermal shield is not within the scope of this report (See the discussion in Section 2.5.1). The thermocouple guide tube assemblies do not perform intended functions as defined in 10 CFR 54 [1] and are, therefore, not within the scope of this report. Portions of the internal vent valve assemblies are active components that do not require an aging management review per 10 CFR 54.21. The surveillance specimen holder tube assemblies (SSHT) are not part of the reactor vessel internals for the plants included in this report. As such, the SSHT assemblies are not within the scope of this report.

Physical and functional descriptions of the individual items within each of the four principle sub-assemblies are presented in Sections 2.1 through 2.4. The reactor vessel internals items not within the scope of this report, e.g., thermal shield and upper thermocouple guide tube assemblies, are discussed in Section 2.5. Material specifications for the items that are evaluated in the report are listed in Table 2-1.

Reactor Vessel Internals Construction

Because of the lack of specific ASME design rules for core support structures at the time of design and construction, Section III of the ASME code was used as a guideline for the design criteria for the reactor vessel internals. The qualification of the internals was accomplished by both analytical and test methods. BAW-10038 [22] described the hot functional test and vibration analysis program for the reactor vessel internals. BAW-10038 used the results of BAW-10051 [2], which predicted the natural frequencies and responses to flow loadings. BAW-10051 extrapolated existing fatigue curves for an assumed endurance limit of 10^{12} cycles. (BAW-10051 qualifies as a TLAA and is addressed in Section 4.5.) BAW-10039 [23] compares measured stress values from the reactor vessel internals vibration and hot functional test to the predicted values and endurance values. It was shown that the actual test values were much lower than the predicted values from BAW-10038, which in turn are less than the endurance values from BAW-10051.

BAW-10008, Part 1, Rev. 1 [24] documents the acceptability of the reactor vessel internals under accident conditions. This report analyzes the internals in the event of a LOCA and a combination of LOCA and seismic loadings. It is shown that although there are some internals deflection, failure of the internals will not occur since the stresses are within established values. Effects of asymmetric cavity loadings are evaluated and shown acceptable for the reactor vessel internals in BAW-1621 [32].

The reactor vessel internals were constructed in accordance with ASME B&PV Code, Sections II, III, and IX, as applicable [26]. ASME Section III or ASTM materials were principally used in the internals. As shown in Table 2-1, all major reactor vessel internals items are made from stainless steel plate, forging, pipe or bar. The CRGT assembly spacer castings, vent valve bodies, incore guide spiders and the outlet nozzles at ONS-3 are made of cast austenitic stainless steel. Most bolting and dowels used for internals fastening are made of stainless steel or Ni-Cr alloys.

The large cylindrical sub-assemblies (e.g., the core barrel, the core support shield, and the plenum cylinder) were fabricated by welded plate material. Forged flanges were joined to the cylinders by welding processes. All austenitic stainless steel welds were required to contain controlled 5% delta ferrite minimum or percentage of Cr $\geq$ 1.9 x % Ni. The sub-assemblies are joined to each other by mechanical fasteners, i.e., bolting. The use of bolting was dictated by spacing and tolerances which made welding the sub-assemblies together difficult or impossible.

The major cylinders required final machining after fabrication welding. The welded items were typically given post-weld treatments prior to machining. To preclude sensitizing the stainless steel materials, either low temperatures (typically below 850°F, approximate threshold for sensitization) or vibratory treatments were used. If heat treating was used, the specifications required heating to 850°F $\pm$ 25 F° at a rate not to exceed 100 F°/hour, holding for a minimum of 48 hours, cooling to 400°F at a rate not to exceed 100 F°/hour and then furnace or air cooled to room temperature. After residual stresses had been equalized and reduced to a minimum, additional residual stresses due to machining were minimized.

All examinations were made to ASME B&PV Code, Section III [26] for a Class "A" vessel. Wrought or forged raw material forms were 100 percent ultrasonically examined. Cast raw material forms were 100 percent radiographically examined. All circumferential full-penetration structural weld joints that support the core were 100 percent radiographically examined. Cast raw material form surfaces were 100 percent liquid penetrant examined. Full penetration welds that were not radiographed and partial penetration structural welds were liquid penetrant inspected by examination of the root and cover passes. All circumferential full penetration structural welds that support the core had cover passes that were liquid penetrant inspected.

Visual (5X magnification) examinations were performed for the following welded joints. For each of these joints the entire weld pass and adjacent base metal were inspected prior to the next pass from the root to and including the cover passes.

1) Partial penetration non-radiographically or non-ultrasonically feasible structural weld joints were 100 percent inspected.

2) Partial or full penetration attachment weld joints for non-structural materials or parts were 100 percent inspected.

3) Partial or full penetration weld joints for attachment of mechanical devices which lock and retain structural fasteners were inspected.

The brazing of the CRGT assembly rod guide sectors and CRGT assembly rod guide tubes to the spacer castings were performed in a hydrogen atmosphere using nickel-based braze alloys (Coast Metals N.P. or low melting Nicrobraze).

Functional and Operational Description

The plenum assembly, described in Section 2.1, is a cylindrical assembly with perforated grids on top and bottom. The plenum assembly fits inside the core support shield, positions the fuel assemblies, and provides the core holddown required for hydraulic lift forces. The plenum assembly provides continuous guidance and protection of the control rods. In addition, the plenum assembly directs flow out of the core to the vessel outlet nozzles. The plenum assembly is removed every refueling outage to permit access to the fuel assemblies.

The CSA remains in place in the reactor vessel and is only removed to perform scheduled inspections of the reactor vessel interior surfaces or of the internals. The CSA is assembled from three separate sub-assemblies which bolt together to form one tall cylinder.

Functionally the CSA provides a structure to physically support the reactor core, and a flow boundary to direct coolant flow. Static loads from the assembled components and fuel assemblies, and dynamic loads from CRA trip, hydraulic flow, thermal expansion, seismic disturbances, and LOCA loads are all carried by the CSA and transferred to the RV closure flange. The CSA walls direct incoming RCS coolant from the cold leg inlet nozzles down the annulus formed between the CSA and the inner RV wall to the lower plenum below the CSA. The coolant then flows up through the bottom portion of the CSA, past the core, into the upper plenum region above the core, and out the outlet nozzles to the hot leg piping.

The core support shield assembly, described in Section 2.2, is the top portion of the CSA. It is a large cylinder with an upper flange that rests on a circumferential support ledge in the reactor vessel closure flange and supports the entire CSA. The core barrel assembly, described in Section 2.3, is a second cylinder bolted to the bottom of the core support shield assembly. The 177 fuel assemblies that make up the core are loaded into the core barrel assembly. The lower internals assembly, described in Section 2.4, is bolted to the bottom of the core barrel assembly. The lower internals support the bottom of the core and direct the coolant flow up past the fuel assemblies. In addition, the lower internals provide guidance of the incore monitoring instrumentation from the reactor vessel interface to the lower fuel assembly end fitting.

2.1 Plenum Assembly

The plenum assembly is a large cylindrical assembly approximately 11-feet tall, located inside of the core support shield directly above the reactor core. This assembly holds down the tops of the fuel assemblies, directs the flow of reactor coolant from the core area to the reactor outlet nozzles and supports the 69 CRGT assemblies. The plenum assembly is removed during every refueling outage to access the fuel assemblies.

The plenum assembly is made up of the following sub-assemblies: the plenum cover assembly, the plenum cylinder assembly, the upper grid assembly, and the CRGT assemblies, as shown in Figure 2-1.

2.1.1 Plenum Cover Assembly

The plenum cover assembly is bolted to the top of the plenum cylinder. It provides support for the top of the 69 CRGT assemblies and the three lifting lugs.

The plenum cover assembly consists of the weldment, the plenum bottom and support flanges, the plenum cover support ring, the cover plate and the lifting lugs. The weldment is a series of parallel flat plates (ribs) assembled to form a square lattice. The ring and flanges provide the vertical and horizontal seating surfaces for the plenum cover and assembly. The cover assembly is bolted to the plenum cylinder top flange. The perforated top plate has matching holes to position the upper end of the CRGT assemblies. Lifting lugs are provided to allow lifting the plenum assembly out of the RV for refueling. Figure 2-2 shows the plenum cover assembly.

Plenum Cover Weldment
The plenum cover weldment is a lattice construction assembled from two sets of ten parallel flat plates intersecting perpendicularly with 10-inch spacing between ribs. The individual ribs are 2-inch thick flat stainless steel plates of varying lengths and heights. Small rib or compression pads are welded to the top outer edge of each rib where it mates with the plenum ring, forming a mating surface on which the reactor vessel head sits.

Plenum Cover Bottom Flange
The plenum cover bottom flange is a flat ring welded to the bottom of the weldment to provide a surface to attach the plenum cover to the plenum cylinder. It is smaller and located inside of the plenum cover support flange. The bottom flange has 64 tapped holes to which the upper flange of the plenum cylinder is bolted.

Plenum Cover Support Flange
The plenum cover support flange is also welded to the bottom of plenum cover weldment assembly. It provides the seating surface that rests on the top of the core barrel shield assembly and against the inner RV wall. At each of the four axes locations, the support flange has keyways which mate with reactor vessel flange keys to align the plenum assembly with the reactor vessel, the reactor closure head control rod drive penetrations, and the CSA.

Plenum Cover Support Ring
The plenum cover support ring is an 8½-inch high, 2-inch thick ring welded onto the top of the support flange and outer vertical edges of the plenum cover weldment. The support ring provides a surface which mates with the reactor vessel head. At each of the four axes locations, the support ring has keyways which mate with reactor vessel flange keys to align the plenum assembly.

Plenum Cover Plate
The plenum cover plate is a ½-inch thick 124-inch diameter disk that is welded to the top center of the plenum cover weldment. It has 69 holes through which the tops of the CRGTs are fitted and welded. The cover plate size allows some reactor coolant flow up past the plenum cover into the upper reactor head region.

Lifting Lugs
Three lifting lugs are located about the plenum cover assembly to be used when removing the plenum assembly, as during refuelings. There are two types of lifting lug arrangements. In all plants but ONS-1, "T"-shaped lifting lugs are fastened with two 1¾-inch diameter bolts to the base blocks. The bolts are secured with locking cups. The base blocks are welded between two of the weldment ribs. At ONS-1, the base blocks and the lifting lugs are one-piece, which is similarly welded between ribs on the plenum cover weldment.

2.1.2 Plenum Cylinder Assembly

As shown in Figure 2-3, the plenum cylinder assembly consists of the plenum cylinder with attached top and bottom flanges, two reinforcing plates welded to the inside surface opposite the outlet nozzles, and two sets of 13 round bars.

Plenum Cylinder
The plenum cylinder is a large 7-foot high cylinder made from 1½-inch thick stainless steel plate. Flanges for connecting the cylinder to the plenum cover and the upper grid are welded to the plenum cylinder ends. Two reinforcing plates are welded to the lower inner surface opposite the core support shield outlet nozzles. Both the reinforcing plates and the plenum cylinder have twenty-four small holes to permit some of the reactor coolant coming up into the plenum to flow directly to the outlet nozzles. Ten large holes (six 34 inches in diameter and four 22 inches in diameter) at the top of the cylinder let the majority of the reactor coolant pass out into the annulus between the plenum cylinder and the core support shield and ultimately down and out through the outlet nozzles.

Top Flange
The plenum top flange is welded to the top of the plenum cylinder. The plenum cover assembly is bolted to the upper flange with sixty-four 1⅛ -inch large hex head bolts held in place with locking cups.

Bottom Flange
The plenum bottom flange is welded to the bottom of the plenum cylinder. The plenum upper grid assembly is bolted to the bottom flange with thirty-six 1-inch diameter large hex head bolts held in place with locking cups.

Reinforcing Plates
The two 3-inch thick reinforcing plates are welded to the inner surface of the plenum cylinder. Twenty-four holes permit some RCS flow directly out from the plenum area to the outlet nozzles.

Round Bars
In order to insure that the space between the plenum cylinder and the core support shield does not collapse and prevent flow through the outlet nozzles, a set of 13 small stainless steel round bars or lugs are welded to the outer surface of the plenum cylinder at each of the outlet nozzle areas. These lugs fit against similar lugs welded to the inner surface of the core support shield. The round bars are 4 inches long and 2½ inches in diameter. (These are frequently referred to as the "LOCA lugs" or "LOCA bosses".)

2.1.3 Upper Grid Assembly

The upper grid assembly is bolted to the plenum cylinder lower flange. It provides the support and seating surface for the tops of the fuel assemblies located in the core barrel below, and support and alignment for the bottoms of the CRGT assemblies. The upper grid assembly sits inside the lower flange of the core support shield.

The upper grid consists of the upper grid ring forging, the upper rib section, and the fuel assembly support pads, as shown on Figure 2-4.

Upper Grid Ring Forging
The upper grid ring forging is a ring with an inward flange on the upper end. The top of the upper grid ring forging is machined to accept the thirty-six 1-inch bolts fastening the upper grid assembly to the plenum cylinder bottom flange. The upper grid rib section is bolted to the bottom of the ring assembly with thirty-six ¾-inch diameter screws held in place with welded lockpins.

Upper Grid Rib Section
The upper grid rib section is a 3-inch thick, 136-inch diameter disk, with 177 squares machined out, leaving a grid with 1-inch wide "ribs". The square holes align with the fuel assembly locations in the core below. Pads to support and align the fuel assemblies are doweled and bolted into the ribs on the bottom side. The topside of the rib section is drilled and tapped to accept the dowels and screws, which hold the bottom flange of the 69 CRGT assemblies to the upper grid.

Fuel Assembly Support Pads
There are 384 small 2-inch high pads attached to the bottom of the upper grid rib section to provide a seating surface and support for the tops of the fuel assemblies. The pads are each held

in place by two Alloy X-750 dowels and a $1/2$-inch diameter cap screw. The dowels and cap screws are all welded in place.

2.1.4 Control Rod Guide Tube (CRGT) Assemblies

The 69 vertical CRGT assemblies are welded to the plenum cover plate and bolted to the upper grid. The CRGT assemblies provide CRA guidance, protect the CRA from the effects of coolant cross-flow, and structurally connect the upper grid assembly to the plenum cover. Design clearances in the guide tube accommodate misalignment between the guide tubes and the fuel assemblies.

The end of each of the 69 control rod assemblies consists of a spider plate through which 16 individual control rods are suspended. As shown in Figure 2-5, the 139-inch long control rods are arranged in two concentric rings, four rods in the middle ring and twelve in the outer ring. The rods have no other support other than the spider head at the top. (The control rod assemblies and the control rod drive mechanisms are not within the scope of this report.) The CRGT assemblies provide support both for the CRGT assemblies as a whole and for each of the 16 individual control rods within each CRA, as well as accommodating the control rod spider that travels the entire length of the CRGT assembly.

The outer portion of the CRGT assemblies consists of tall pipes (or guide housings) welded to the CRGT assembly flanges at their bottoms. The inside of each CRGT assembly consists of an internal sub-assembly with ten parallel horizontal spacer castings to which are brazed 12 perforated vertical rod guide tubes and 4 pairs of vertical rod tube guide sectors, also called "C-tubes". These internal sub-assemblies of spacers, rod guide tubes and rod guide sectors are referred to as the "rod guide brazements". Figure 2-6 shows the CRGT assembly spacer castings and the rod guide brazement configuration.

CRGT Assembly Pipes

The CRGT assembly pipes (or guide housings) are approximately 12-foot tall, 8-inch diameter, Schedule 40 stainless steel pipes. At ten elevations, they are drilled at 4 equally spaced locations to accommodate the $3/8$-inch diameter cap screws that hold the CRGT assembly spacer castings in place.

Four equally spaced 3-inch diameter holes are located 2 inches from the bottom of the CRGT assembly pipes. Above them are two rows of four 3-inch wide, $8\,3/4$-inch high oval shaped holes. These holes allow some of the reactor coolant traveling up the CRGT assembly pipes to exit out into the plenum and to ensure that the pressures are equalized on both sides of the CRGT assembly pipes and prevent hydraulic effects from impeding control rod travel.

The pipes are welded to the CRGT assembly flanges at the bottom, and to the top of the plenum cover plate. The top of the CRGT assembly pipes extend $21\,1/8$ -inches above the plenum cover plate into the upper head area. The CRGT assembly pipes are shown in Figures 2-1 and 2-7.

CRGT Assembly Flanges
The CRGT assembly flanges are 1 1/4-inch thick, 4 1/4-inch sided square plates with a hole in the center to match the inner diameter of the CRGT assembly pipes. Four additional small semicircular flow paths are equally spaced about the center to permit RCS flow upward through the flange on the outside of the CRGT assembly pipe. Each flange is drilled to accept two 1/2-inch diameter dowels and four 1/2-inch diameter hex head cap screws which are torqued into the upper grid rib section.

CRGT Assembly Spacer Castings
The CRGT assembly spacer castings are 3/4-inch thick disks, with internal spaces to conform to the general shape of the control rod spider, with margins to permit RCS flow and to accommodate the rod guide tubes and rod guide sectors.

CRGT Assembly Rod Guide Tubes
Within each CRGT assembly are 12 guide tubes. These are long 0.750-inch ID, 0.095-inch thick tubes with a 5/16-inch wide vertical slot. The tubes have a vertical row of 1/4-inch holes to permit RCS flow into the area. The CRGT assembly guide tubes are brazed into holes in the CRGT assembly spacer castings, with the slots aligned to match where the control rod assembly spider arms pass.

CRGT Assembly Rod Guide Sectors
The CRGT assembly rod guide sectors are similar to the CRGT assembly rod guide tubes however they are for the 4 inner individual control rods in each assembly which are suspended from the middle of a spider arm and thus require slots on both sides of the tubes. They are long, 0.109-inch thick plates with a curved cross section. They are brazed in pairs in holes in the CRGT assembly spacer castings, facing each other with a gap between them to permit travel of the spider arm between them. The rod guide sectors do not have cooling holes like the rod guide tubes, since they are open on two sides.

2.2 Core Support Shield Assembly (CSS)

The CSS is a large flanged cylinder, which sits on top of the core barrel and fuel assemblies. The CSS provides a boundary between the incoming cold reactor coolant, which is directed downward on the outside of the CSS, and the heated reactor coolant flowing upward from the core area into the plenum area and out through the reactor outlet nozzles, bounded by the inside core support shield wall and nozzles.

The top flange of the CSS rests on and is supported by a circumferential ledge in the reactor vessel closure flange. Ultimately this provides the support for the entire reactor vessel internals assembly. (The core barrel assembly bolts to and is suspended from the core support shield bottom flange; the lower internals assembly is similarly bolted to and suspended from the lower core barrel flange.)

The plenum assembly is supported by and fits inside the CSS. The bottom flange of the CSS is bolted to the core barrel. The inside surface of the core support shield bottom flange provides the lower seating surface and aligns the plenum assembly.

The core support shield cylinder wall has two openings with nozzles for RCS outlet flow. These openings are formed by two forged rings which seal to the reactor vessel outlet nozzles by the differential thermal expansion between the stainless steel CSS and the low alloy steel RV. The nozzle seal surfaces are finished and fitted to a predetermined cold gap providing clearance for CSA installation and removal. At operating temperature, the mating metal surfaces are in contact to make a seal without exceeding allowable stresses in either the RV or internals.

Eight vent valve mounting rings are welded in the cylinder wall. Internals vent valves are installed in the CSS cylinder wall to allow steam flow from the core should a cold leg (reactor coolant inlet) pipe rupture occur.

Core Support Shield Cylinder

The core support shield cylinder is a 145-inch ID, approximately 7-foot high cylinder. There are ten major penetrations through the core support shield cylinder: two 67-inch OD core outlet nozzles and eight vent nozzles. Figure 2-8 shows most of the items in the core support shield assembly.

The core support shield top flange is welded to the top of the core support cylinder. The core support shield top flange extends out from the inner diameter. The bottom of the top flange rests on a circumferential ledge in the reactor vessel closure flange. The top of the flange provides the seating surface to support the bottom of the plenum cover support flange, and thus supports the entire plenum assembly. The bottom of the top flange is penetrated in eight places by the vent nozzles.

The core support shield bottom flange is welded to the bottom of the core support shield cylinder. The bottom of the plenum assembly is guided by the inside surface of the bottom flange of the core support shield. The core support shield bottom flange is bolted to the top flange of the core barrel with one hundred and twenty 1¾-inch diameter core barrel bolts, secured with locking clips or locking cups.

Outlet Nozzles

The two outlet nozzles are 67-inch OD, 8¾-inch thick curved ring-shaped inserts that are welded into the support shield cylinder with full penetration welds, (i.e., the inner surfaces are welded flush with the inner cylinder wall and extend out horizontally 8 inches towards the inner RV wall). The wall thickness of the nozzle tapers, with the inner hole having an oval shape. The outlet nozzles at ONS-3 are castings; all other plants have forged nozzles.

Vent Valve Nozzles

The eight vent nozzles are flat ring-shaped inserts welded into the core barrel support cylinder to provide support for the internal vent valve sub-assemblies. The nozzles are welded into the core barrel cylinder using full penetration welds with their inner edges matching the inner core cylinder wall (when viewed from above, the inner flat surface of the nozzles crosses the inside of the core support cylinder). The nozzles are approximately 38-inch OD and are 6¼-inches thick in cross section. To accommodate the internal vent valves, the inner surfaces of the rings have lips and flanges.

Two small guide blocks are welded to the top outside surface of each vent nozzle. The guide blocks are machined to provide a small triangular seating surface for the vent valve assemblies.

Internal Vent Valves

Eight internal vent valve assemblies are installed in the core support shield as shown in Figure 2-11. For all normal operating conditions, the vent valve is closed. In the event of a pipe rupture in the reactor vessel inlet pipe, the valve will open to permit steam generated in the core to flow directly to the break, and will permit the core to be flooded and adequately cooled after emergency core coolant has been supplied to the reactor vessel.

Each valve assembly consists of a hinged disc, valve body with sealing surfaces, a split-retaining ring and fasteners, which retain and seal the perimeter of the valve assembly, and an alignment device to maintain the correct orientation of the valve assembly for hinged-disc operation. Each valve assembly can be remotely handled as a unit for removal or installation. Valve component parts, including the disc, are designed to minimize the possibility of loss of parts to the coolant system, and all operating fasteners include a positive locking device. The hinged-disc includes a device for remote testing and verification of proper disc function. The external side of the disc is contoured to absorb the impact load of the disc on the RV inside wall without transmitting excessive impact loads to the hinge parts as a result of a loss-of-coolant accident.

The hinge assembly consists of a shaft, two valve body journal receptacles, two valve disc journal receptacles, and four flanged shaft journals (bushings). Loose clearances are used between the shaft and journal inside diameters, and between the journal outside diameters and their receptacles. The valve disc hinge journal contains integral exercise lugs for remote operation of the disc with the valve installed in the core support shield. The hinge assembly provides eight loose rotational clearances to minimize any possibility of impairment of free motion of the disc in service. In addition, the valve disc hinge loose clearances permit disc self-alignment so that the external differential pressure adjusts the disc seal face to the valve body seal face. This feature minimizes the possibility of increased leakage and pressure induced deflection loadings on the hinge parts in service. The hinge assembly and disc are active components and are exercised at each refueling outage to evaluate performance. In accordance with 10 CFR 54.21(a)(1)(i) which exempts functions accomplished through changes of states or moving parts from a license renewal evaluation [1], these vent valve parts are not subject to aging management review and thus are not within the scope of this report.

The vent valve materials were selected on the basis of their corrosion resistance, surface hardness, anti-galling characteristics, and compatibility with mating materials in the reactor coolant environment. The jackscrews, once installed, may need to be cut out in order to replace the vent valve assembly. As such, vent valve assemblies with modified locking devices were made available. These modified locking devices consist of different materials as noted in Table 2-1.

Round Bars
At each outlet nozzle area, thirteen round bars are located on the inner side of the core support shield to mate with the similar lugs welded on the outer side of the plenum cylinder. The round bars insure that the two cylinders are kept apart so RCS flow is not disrupted under any conditions. (These are frequently referred to as the "LOCA lugs" or "LOCA bosses".)

Flow Deflectors ("U"- Baffles)
A baffle consisting of three 1-inch thick plates shaped to form an inverted "U" is welded to the outer side of the core support cylinder around the area opposite each of the four inlet (cold leg) nozzles. These baffles help divert the incoming flow downward to the bottom of the core, and minimize the upward flow that might damage the internal vent valve assemblies. The baffle plates originally were of a uniform 4-inch width (i.e., extended out 4 inches from the cylinder) and blocked most of the annulus between the core support shield cylinder and the RV shell. Following flow induced vibration problems during hot functional testing at ONS-1, however, the side baffle plates were tapered down to 5/8-inch width, so that only the top horizontal baffle plate spans most of the annulus. This helps reduce the flow velocities seen at the bottom of the core.

Lifting Lugs
Three lifting lugs are welded on the inside of the core support shield top flange. These lugs permit lifting the CSA out of the core when required, such as for vessel inspections.

2.3 Core Barrel Assembly (CBA)

The core barrel supports the fuel assemblies. The core barrel consists of a flanged cylinder, rings of internal horizontal former plates bolted to the cylinder at eight elevations, and vertical baffle plates bolted to the former plates to produce an inner wall enclosing the fuel assemblies. The bottom flange of the CSS is bolted to the top flange of the core barrel cylinder and the lower grid assembly bolts to the core barrel bottom flange.

Incoming cold RCS flow is downward along the outside of the core barrel cylinder and upward through the fuel assemblies contained inside the core barrel. A small portion of the coolant flows upward through the space between the core barrel outer cylinder and the baffle plates.

At the fourth elevation from the bottom, near the hottest section of the core, the ring of former plates are narrower than those at the other elevations, and the baffle plates are bolted to these narrower formers with special shoulder bolts that maintain a 1/4-inch gap between the baffle plates and former plates. This arrangement provides additional cooling flow to the hottest portion of the baffle plates and some flexibility to the assembly under unusual loads.

The thermal shield is attached to the outside of the core barrel assembly by an upper thermal shield restraint assembly. The thermal shield and the restraint assembly, except the thermal shield-to-core barrel bolts, are not within the scope of this report; however, descriptions of these items have been provided in Section 2.5. Figure 2-8 shows most of the items in the core barrel assembly.

Core Barrel

The core barrel is a cylinder approximately 12¼ feet high and 2 inches thick. It is formed from two cylinders welded together circumferentially. The core barrel top and bottom flanges are welded to the ends of the cylinder. The core support shield assembly bolts to the top flange with one hundred and twenty 1¾-inch diameter core barrel bolts secured with locking clips or locking cups. The lower grid shell forging is bolted to the core barrel bottom flange with one hundred and eight 1¾-inch diameter core barrel bolts secured with locking clips or locking cups (except for ONS-1 which has 12 additional bolts over the guide lug locations).

At twenty equally spaced locations, the thermal shield upper restraint assemblies are bolted on to the outer vertical wall of the core barrel top flange with three 1½-inch diameter bolts secured with locking clips.

Baffle Plates

The vertical baffle plates form an outer perimeter of the core area to confine and direct the flow of reactor coolant. The baffle plates do not ordinarily provide any structural support to or affect the alignment of the fuel assemblies since there is a clearance between the outer fuel assemblies and the baffle plates. The baffle plates are approximately 13¼ feet high, ¾-inch thick, with widths varying from about 8 to 45 inches. At various elevations the baffle plates have rows of 1⅜-inch diameter flow holes to help equalize the coolant behind the baffle plates with the main coolant flow. Figure 2-9 is a sketch showing the inside of the core barrel with the baffle plates and one representative fuel assembly in place.

At seven elevations the baffle plates are bolted to the formers with seven hundred and fifty-six ⅝-inch diameter hex head bolts secured in place by stainless steel locking pins.

At the fourth highest elevation, the narrow baffle plates are bolted to the formers with one hundred and eight ⅝-inch diameter core barrel shoulder screws which have a shoulder which holds the baffle plates ¼-inch out from the former. The shoulder screws are secured in place by ⅛-inch diameter dowels which are welded in place.

At the tall vertical joints where two baffle plates meet to form corners, a total of six hundred and twelve 7/16 -inch diameter bolts secured with locking rings hold the plates together.

Formers

The 1¼-inch thick former plates provide horizontal framing to support the vertical baffle plates at eight elevations. The outside edges of the formers curve to match the inside surface of the core barrel to which they are bolted. Inside surfaces of the formers are either flat or step shaped to support the various baffle plates. (The eight former plates at the fourth elevation are ¼" narrower and offset that distance from the baffle plates.) The formers have small holes to permit some reactor coolant to flow up through and cool the void behind the baffles.

To hold the formers in place, a total of seven hundred and four 5/8-inch diameter socket head cap screws are bolted through the outer side of the core barrel into the formers. The screws are held in place with locking pins.

At 16 locations on the top and bottom rows of formers there are 0.625-inch diameter Alloy X-750 dowels used to locate the formers on the core barrel.

2.4 Lower Internals Assembly (LIA)

As shown in Figure 2-10, the lower internals assembly consists of the lower grid assembly, the flow distribution assembly, and the incore guide tube assemblies. The lower grid assembly is a series of grids and support structures bolted to the bottom of the core barrel to provide structural support to the core. The flow distribution assembly is a set of flow distribution plates located below the lower grid, bowing out into the lower reactor vessel plenum region. The flow distribution assembly helps direct coolant flow upwards towards the core. The incore guide tube assemblies run through and are supported by both the flow distribution assembly and the lower grid assembly. The incore guide tube assemblies provide support and protection for the incore monitoring detectors.

2.4.1 Lower Grid Assembly

The lower grid assembly provides alignment and support for the fuel assemblies, supports the thermal shield and flow distributor, and aligns the incore instrument guide tubes with the fuel assembly instrument tubes. The lower grid consists of three grid structures or flow plates. From top to bottom they are the lower grid rib section, the flow distributor plate, and the lower grid forging. Each of these flow plates has holes or flow-ports to direct reactor coolant flow upward towards the fuel assemblies. The lower grid assembly is surrounded by the lower grid shell forging. The lower grid shell forging is a forged flanged "ring" cylinder, which supports the various horizontal grid structures and flow plates.

The top flange of the lower grid shell forging is bolted to the lower flange of the core barrel by the 108 (plus an additional 12 bolts at ONS-1), 13/4-inch diameter core barrel bolts described previously.

Alignment between fuel assemblies and incore instruments is provided by pads bolted to the top of the lower grid rib section.

Twelve pairs of guide blocks bolted to the outer surface of the lower grid shell forging mate with the guide lugs welded on the inside RV wall. The guide blocks are not credited with preventing internals motion or as an internals support during normal and upset operating conditions and also for seismic and design break events.

Lower Grid Rib Section

The lower grid rib section is a 5-inch thick, 141-inch diameter disk through which 177 squares are machined out, leaving a grid with 1-inch wide "ribs". The square holes align with the fuel assembly locations in the core above. There are additional holes about the periphery of the disk to permit a small bypass flow of reactor coolant up behind the baffle plates in the core barrel.

There are 384 small fuel assembly support pads attached to the top of the rib section to provide a seating surface and support for the bottoms of the fuel assemblies. A 1/2-inch diameter, 2 1/4-inch long cap screw is used to hold each pad in place. Two 3/8 -inch diameter Alloy X-750 dowels position each pad. Below the rib plate at 48 grid intersections, there are support post assemblies that provide support from the lower grid forging. The support post assemblies are bolted in place with 1-inch diameter socket head cap screws secured with welded locking pins.

Incore guide tube spider castings are welded in 52 of the holes to provide support for the tops of the incore instrument guide tubes. The spider castings are cylinders with four legs that are welded to the walls of the holes in the lower grid rib section.

Lower Grid Flow Distributor Plate

The lower grid flow distributor plate, located midway between the lower grid rib section and the lower grid forging, aids in distributing coolant flow. It is a flat 1-inch thick, 135 7/8-inch diameter perforated plate with a 1/8-inch lip around the bottom. The flow distributor plate rests on and is welded to a 1/2-inch lip on the lower grid shell forging.

The flow distributor plate has six hundred and seventy-seven 3 3/8-inch diameter flow holes (177 of which are aligned with the center of the fuel assemblies). Twelve of the normal flow holes near the center of the flow distributor plate are fitted with orifice plugs which reduce the diameter of the flow port down to 1 7/8-inches. There are also 24 smaller flow holes and 48 holes to accommodate the support posts. The support posts are welded to the lower grid flow distribution plate.

Lower Grid Forging

At all plants except ONS-1, the lower grid forging is a single 135-inch diameter forged disk that serves as the main weight-bearing structure in the lower grid. The majority of the lower grid forging, i.e., the center 96 inches of the disc, is 13 1/2 inches thick. The disc tapers up to 6 inches thick at its edges. There are 177 flow holes machined out of the lower grid forging, aligned with the fuel assemblies above. The lower grid forging is welded to the lower grid shell forging. At ONS-1 only, the lower grid forging is fabricated as a lattice grid from ribs, similar to the plenum cover weldment described in Section 2.1.1. The lower ends of the 48 support post assemblies are welded to the top of the lower grid forging.

Lower Grid Shell Forging

The lower grid shell forging is a 2-foot high, 136-inch ID cylinder with numerous internal and external flanges and lips that support the various items of the lower grid assembly. The lower grid shell forging is 4-inches thick at its thinnest cross-section.

The lower grid shell forging is bolted to the core barrel lower flange with 108 core barrel bolts, described previously. The lower end of the thermal shield is shrunk fit on the lower grid flange and fastened by ninety-six 1-inch diameter bolts or studs and nuts secured with locking clips or locking cups. The lower grid rib section is bolted to the shell forging with thirty-six 3/4-inch diameter socket head cap screws secured with welded locking pins. The flow distributor plate rests on and is welded to a 1/2-inch lip on the lower grid shell forging. The lower grid forging rests on and is welded to the top surface of the lower grid shell forging lower flange. The flow distributor assembly bolts into the bottom of the shell forging with ninety-six 1-inch diameter high strength bolts secured with locking clips. The lower surface of the bottom flange of the lower grid shell forging holds the clamping ring in place, which holds the incore guide support plate in place against the flow distributor flange.

Guide blocks are bolted at twelve equidistant locations around the outside vertical wall of the lower grid shell forging. These blocks engage the guide lugs welded to the wall of the RV and serve to maintain alignment and prevent rotation of the core internals. Shock pads are bolted to the underside of the upper flange of the lower grid shell forging, directly above the guide blocks.

Guide Blocks

The 24 guide blocks are each 6 1/2-inches wide, 5-inches high with beveled guiding/mating surfaces extending out 3 inches from the shell forging wall. Each is held in place with a 1-inch diameter hex head bolt and washer and a 1 1/2-inch diameter Alloy X-750 dowel.

Shock Pads

Twelve shock pads are bolted to the lower surface of the upper flange of the lower grid shell forging, located directly above the reactor vessel guide lugs. In the unlikely event of a core barrel joint failure, the RV core guide lugs and lower grid shock pads will limit the core to drop approximately one-half inch. An assessment has been made that failures of the core barrel joint would not result in the loss of integrity of the RCS pressure boundary, core coolability, or core shutdown capability [12].

Support Post Assemblies

The support posts are 48 cylinders placed between the lower grid forging and the lower grid rib section to provide support. The support post assemblies consist of the support pipes and the associated bolting plugs. The support pipes are made from 10 1/2-inch high sections of 4-inch schedule 160 pipe. There are four equally spaced notches at the bottom of the cylinders, where they are welded to the top of the lower grid forging that allow coolant flow up from below. The bolting plugs are 1 3/4-inch high disks welded to the top of the support pipes. The bolting plugs have four scallops shaped holes machined out of the edges so that the tops have a cruciform shape through which coolant can flow. The top of each bolting plug is drilled and tapped to accept the cap screw used to hold it to the lower grid rib section.

The support post assemblies rest on top of the lower grid forging, straddling over grid intersections. They are fitted through matching holes in the flow distribution plate and rest against the bottom of the lower grid rib section. The support posts are welded to the top of the lower grid forging and on both sides of the penetration of the lower grid flow distribution plate. They are bolted to the bottom of the lower grid rib section by 1-inch diameter socket head screws secured with locking pins.

2.4.2 Flow Distributor Assembly

Flow Distributor Head and Flange

The flow distributor is a perforated dished head with an external flange that is bolted to the bottom flange of the lower grid. The flow distributor supports the incore instrument guide tubes and directs the inlet coolant entering the bottom of the core.

The flow distributor head is a 2-inch thick, 136-inch ID bowl-shaped plate that bows downward about 20 inches. The head is welded to the flow distributor flange, which is 5 inches high, with an approximately 3-inch thick flange extending out to a 142 inch OD. The incore guide support plate fits across the flange, resting in a lip in the flange. The clamping ring fits against the inside diameter of the flange on top of and holding the incore guide support plate in place. This whole assembly is bolted to the bottom of the lower grid shell forging with ninety-six 1-inch diameter high strength bolts secured with locking clips.

There are 52 approximately 4½-inch diameter holes through which the incore instrument guide tubes pass. Fifteen of these incore holes have shallow counterbores on the bottom edge to permit welding the instrument guide tubes directly to the flow distributor head plate. The remaining 37 guide tubes are secured by a set of four gussets which are ¾-inch thick triangular shaped pieces, 6-inches high and 1¾-inches wide. The long sides of the gussets are welded to the guide tubes and the bases are welded to the distributor head.

There are one hundred and fifty-six 6-inch diameter holes and five 3½-inch diameter holes in the flow distributor head to permit reactor coolant flow upward through the lower grid assembly.

Incore Guide Support Plate

The incore guide support plate is a 134-inch diameter, 2-inch thick disk, with 52 shaped holes to accommodate the incore instrument guide tubes. The guide tubes are held in place by washers and guide tube nuts secured by welded locking clips.

At 46 of the guide tube holes there are also four oval-shaped flow ports machined through the guide support plate to permit reactor coolant flow parallel to the incore guide tubes. There are also numerous holes between 6½ and 7½ inches in diameter for reactor coolant flow.

The incore guide support plate rests on a lip in the top of the flow distributor assembly. The clamping ring sits on top of the plate and holds it in place.

Clamping Ring

The clamping ring is an approximately 4-inch high, 132-inch diameter, 1-inch thick ring that fits against the inner surface of the flow distributor ring forging, atop the incore support plate. When the flow distributor assembly is bolted to the lower grid shell forging, this clamping ring holds the incore support plate in place.

2.4.3 Incore Guide Tube Assemblies

The incore instrument guide tube assemblies guide the 52 incore instrument assemblies from the instrumentation nozzles in the RV bottom head to the instrument tubes in the fuel assemblies. Horizontal clearances are provided between the RV instrumentation nozzles and the instrument guide tubes in the flow distributor to accommodate misalignment. The incore instrument guide tubes are designed so they will not be affected by core drop.

The guide tubes are long tapered tubes through which the incore nuclear detectors and thermocouples are fed up into the fuel assemblies. The diameters vary along the length of the guide tubes. At the top, where they are held in place by the spiders welded into the lower grid rib section, the guide tubes have a 1-inch OD with a 0.60 to 0.67-inch center bore. At the bottom, the guide tubes have a $4\frac{1}{2}$-inch OD with a $3\frac{1}{2}$-inch ID. The top 32-inches of all 52 guide tubes, from where they penetrate the flow distributor up to the spiders in the lower grid rib section, are essentially identical. There are ten different guide tube models, however, which differ in their overall length, varying from $77\frac{3}{4}$ to $51\frac{1}{4}$ inches. The length required depends upon the location within the core, as the distances vary between the incore guide support plate and the flow distributor head and between the flow distributor head and the bottom of the RV.

The guide tube assemblies are attached to the bottom of the flow distributor head either by a weld bead around the full circumference of the guide tube, or by four gussets which are welded to the flow head and the guide tubes. The guide tubes then have an interference fit through holes in the incore guide support plate. The guide tubes are held to the top of the incore support plate with washers and the guide tube nuts. The outside of the guide tubes have a $1\frac{3}{4}$-inch section of threading at this location to engage with the guide tube nuts. The guide tubes have an approximate 2-inch diameter where they pass up through $6\frac{1}{2}$-inch diameter holes in the lower grid forging and the $3\frac{3}{8}$-inch diameter holes in the flow distributor plate.

Gussets
Thirty-seven guide tubes whose projected length below the flow distributor require additional stiffness are secured by sets of four gussets which are $\frac{3}{4}$-inch thick triangular shaped pieces, 6-inches high and $1\frac{3}{4}$-inches wide. The long sides of the gussets are welded to the guide tubes and the bases are welded to the distributor head.

Guide Tube Nuts
The guide nuts are $2\frac{1}{2}$-inch tall, $\frac{1}{2}$-inch thick nuts that fit over the guide tubes and secure them to the top of the incore support plate.

Guide Tube Spiders
Spider castings are welded in 52 of the holes to provide support for the incore instrument guide tubes. The spider castings are $1\frac{3}{4}$-inch high, 1-inch ID cylinders with four $\frac{1}{4}$-inch thick "L"-shaped legs that extend out to and are welded to the walls of the holes in the lower grid rib section. The inner diameters of the spider tube cylinders are chrome plated 0.0002 to 0.0004 inches thick. The chrome-plated bore of the spider hub forms a guide bushing for the top of the incore instrument guide tube assembly to accommodate longitudinal thermal expansion [2].

2.5 Reactor Vessel Internals Not Within the Scope of This Report

2.5.1 Thermal Shield

The thermal shield and its restraining devices are described here for the benefit of the reviewers and are not within the scope of this report. Failures of the thermal shield bolts, which are documented in Section 3.6, resulted in the B&WOG reactor internals bolting integrity program. As such, the thermal shield bolts are subject to aging management review and will be addressed in this report.

The cylindrical stainless steel thermal shield is installed in the annulus between the core barrel cylinder and RV inner wall. The thermal shield reduces the incident gamma and neutron flux and hence gamma absorption internal heat generation in the RV wall, thereby reducing the resulting thermal stresses and radiation effects on the RV wall [2]. The thermal shield upper end is restrained against inward and outward vibratory motion by 20 equally spaced restraint sub-assemblies bolted to the core barrel cylinder. The lower end of the thermal shield is shrunk fit on the lower grid flange and fastened by ninety-six 1-inch diameter bolts or studs and nuts secured with locking clips or locking cups.

The thermal shield is assembled from two 146-inch ID, 3-inch thick cylinders welded together circumferentially to obtain an overall height of 14-feet, 3 1/4-inches.

Thermal Shield Upper Restraint Assemblies
There are twenty thermal shield upper restraint assemblies used to bolt the upper end of the thermal shield to the outer wall of the core barrel cylinder top flange. Each assembly consists of three 10-inch wide by 6 1/4-inch high rectangular blocks that are bolted together. The inner block, the shim, is 1 1/8 inches thick. It serves to keep the assembly at the correct distance out from the core cylinder wall. The inner "B" and outer "A" blocks are 1 3/4- and 2 1/8 -inch thick blocks, respectively, slotted at the bottom to hold the top of the thermal shield wall. Each assembly is held together with two 1/2-inch diameter, 1 1/4-inch long hex head cap screws bolted from the rear (shim) side. The restraint assemblies are then positioned and secured with three 1/2-inch dowels and plugs and three 1 1/2-inch diameter, 7 1/2-inch long restraint bolts secured with locking clips welded to the restraints.

2.5.2 Upper Thermocouple Guide Tube Assemblies

ONS-1 and TMI-1 originally had eight upper thermocouple guide tube assemblies that were welded to the plenum cover and also to the plenum cylinder. The eight thermocouple guide tube assemblies were removed from ONS-1 after the first HFT. The thermocouple guide tube assemblies are currently still part of the reactor vessel internals package at TMI. These assemblies mate up with eight penetrations in the RV closure head. These assemblies were intended for thermocouples but are no longer used. As such, they do not support a reactor vessel internals intended function, so they are not subject to aging management review and thus are not within the scope of this report.

Table 2-1 Reactor Vessel Internals Materials of Construction

Item	Plant	Material	Specification Number	Type or Grade/Class	Product Form
PLENUM ASSEMBLY					
Plenum Cover Assembly					
Plenum Cover Weldment Ribs	All	Stainless Steel	A 240-63	Tp. 304	Plate
Rib Pads	All	Stainless Steel	A 276-65	Tp. 304	Bar
Plenum Cover Bottom Flange	All	Stainless Steel	A 240-63	Tp. 304	Plate
Plenum Cover Support Flange	All	Stainless Steel	A 240-63	Tp. 304	Plate
Plenum Cover Support Ring	All	Stainless Steel	A 240-63	Tp. 304	Plate
Plenum Cover Plate	All	Stainless Steel	A 240-63	Tp. 304	Plate
Lifting Lugs	ANO-1 ONS-2 ONS-3 TMI-1	Stainless Steel	A 240-63	Tp. 304	Plate
Base Blocks	ANO-1 ONS-2 ONS-3 TMI-1	Stainless Steel	A 240-63	Tp. 304	Plate
Lifting Lug-to-Base Block Bolts	ANO-1 ONS-2 ONS-3 TMI-1	Stainless Steel	A 193-65	Gr. B-8	Bar
Locking Cups	ANO-1 ONS-2 ONS-3 TMI-1	Stainless Steel	A 167	Tp. 304	Plate
Integral Lifting Lug w/Base Block	ONS-1	Stainless Steel	A 240-63	Tp. 304	Plate

Table 2-1 Reactor Vessel Internals Materials of Construction (continued)

Item	Plant	Material	Specification Number	Type or Grade/Class	Product Form
PLENUM ASSEMBLY (Continued)					
Plenum Cylinder					
Cylinder	All	Stainless Steel	A 240-63	Tp. 304	Plate
Top Flange Bottom Flange	All	Stainless Steel	A 473-63	Tp. 304	Forging
Reinforcing Plates	All	Stainless Steel	A 240-63	Tp. 304	Plate
Round Bars	All	Stainless Steel	A 276-65	Tp. 304	Bar
Top Flange-to-Cover Bolts	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Locking Cups	All	Stainless Steel	A 167	Tp. 304	Plate
Bottom Flange-to-Upper Grid Screws	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Locking Cups	All	Stainless Steel	A 167	Tp. 304	Plate
Upper Grid Assembly					
Upper Grid Rib Section	All	Stainless Steel	A 240-63	Tp. 304	Plate
Upper Grid Ring Forging	All	Stainless Steel	A 473-63	Tp. 304	Forging
Rib-to-Ring Screws	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Lockpins	All	Stainless Steel	AISI	Tp. 304	Bar
Fuel Assembly Support Pads	All	Stainless Steel	A 276-65	Tp. 304	Bar
Dowels	All	NiCr Alloy (Alloy X-750)	AMS 5667F	UNS NO7750	Bar
Cap Screw	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Control Rod Guide Tube Assembly					
CRGT Pipe	All	Stainless Steel	A 312-64	Tp. 304	Pipe
CRGT Flange	All	Stainless Steel	A 240-63	Tp. 304	Plate
Flange-to-Upper Grid Screws	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Dowel	All	Stainless Steel	A 276-65	Tp. 304	Bar
CRGT Spacer Castings	All	Stainless Steel	A 351-65	Gr. CF-3M	Casting
Spacer Castings Screws	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Spacer Casting Washer	All	Stainless Steel	AISI	Tp. 304	Plate

Table 2-1 Reactor Vessel Internals Materials of Construction (continued)

Item	Plant	Material	Specification Number	Type or Grade/Class	Product Form
PLENUM ASSEMBLY (Continued)					
Control Rod Guide Tube Assembly (continued)					
CRGT Rod Guide Tubes	All	Stainless Steel	SA-240-63	Tp. 304L	Plate
CRGT Rod Guide Sectors	All	Stainless Steel	SA-240-63	Tp. 304L	Plate
CORE SUPPORT SHIELD ASSEMBLY (CSS)					
Core Support Shield Cylinder	All	Stainless Steel	A 240-63	Tp. 304	Plate
Top Flange	All	Stainless Steel	A 473-63	Tp. 304	Forging
Bottom Flange	All	Stainless Steel	A 473-63	Tp. 304	Forging
Core Support Shield-to-Core Barrel Bolts	ANO-1	Stainless Steel (Alloy A 286) (6 original 114 - replacement)	A 453	Gr. 660, Cond A	Bar
	ONS-1 ONS-2 ONS-3	Stainless Steel (Alloy A 286) (all original)	A 453	Gr. 660, Cond A	Bar
	TMI-1	NiCr Alloy (Alloy X-750)	A 637-70	Gr. 688	Bar
Locking Clips (original bolts)	All	Stainless Steel	A 240-63	Tp. 304L	Plate
Locking Cups & Tie Plates (replacement bolts)	ANO-1	Stainless Steel	A 240-74 or A 479-75	Tp. 304L	Plate
Outlet Nozzles	ANO-1 ONS-1 ONS-2 TMI-1	Stainless Steel	A 473-63	Tp. 304	Forging
	ONS-3	Stainless Steel	A 351-65	Gr. CF-8	Casting

Table 2-1 Reactor Vessel Internals Materials of Construction (continued)

Item	Plant	Material	Specification Number	Type or Grade/Class	Product Form
CORE SUPPORT SHIELD ASSEMBLY (CSS) (Continued)					
Vent Valve Nozzles	All	Stainless Steel	A 473-63	Tp. 304	Forging
Vent Valve Guide Blocks	All	Stainless Steel	A 276-65	Tp. 304	Bar
Vent Valve Body	All	Stainless Steel	A 351	Gr. CF-8	Casting
Retaining Rings	All	Stainless Steel	AMS 5658	Type 15-5 PH	Forging
Jack Screws	All	Stainless Steel	AMS 5737C	Alloy A-286	Bar
Misc locking device parts (Original)	All	Stainless Steel	A 240 A 276 A 276 A 286	Tp. 304 Tp. 304 Tp. 431 ---	---
Misc locking device parts (Modified)	All	Stainless Steel	A 240 SA-479 SA-182 SA-193	Tp. 304 Tp. 304 Tp. 304 Gr. B8 or B8M	---
		NiCrFe Alloy (Alloy 600)	MIL-N-23228A or SB-168 MIL-N-23229A AM.2T1 or SB-166	Cond. A --- Cond. A ---	---
		NiCr Alloy (Alloy 718)	AMS-5662C or SA-637	--- Gr. 718	---
Round Bars	All	Stainless Steel	A 276-65	Tp. 304	Bar
Flow Deflectors	All	Stainless Steel	A 240-63	Tp. 304	Plate
Lifting Lugs	All	Stainless Steel	A 276-65	Tp. 304	Bar

Table 2-1 Reactor Vessel Internals Materials of Construction (continued)

Item	Plant	Material	Specification Number	Type or Grade/Class	Product Form
CORE BARREL ASSEMBLY (CBA)					
Core Barrel Cylinders	All	Stainless Steel	A 240-63	Tp. 304	Plate
Top Flange	All	Stainless Steel	A 473-63	Tp. 304	Forging
Bottom Flange	All	Stainless Steel	A 473-63	Tp. 304	Forging
Lower Internals Assembly-to-Core Barrel Bolts	ANO-1 ONS-2 ONS-3	Stainless Steel (Alloy A 286) (all original)	A 453	Gr. 660, Cond A	Bar
	ONS-1	Stainless Steel (Alloy A 286) (108 - original)	A 453	Gr. 660, Cond A	Bar
		Stainless Steel (12 - original)	A 193-65	Gr. B-8	Bar
	TMI-1	NiCr Alloy (Alloy X-750)	A 637-70	Gr. 688	Bar
Locking Clips	All	Stainless Steel	A 240-63	Tp. 304L	Plate
Core Barrel -to-Thermal Shield Bolts	ONS-1 ONS-2 ONS-3 ANO-1	Stainless Steel (Alloy A 286) (all original)	A 453	Gr. 660, Cond A	Bar
	TMI-1	NiCr Alloy (Alloy X-750)	A 637-70	Gr. 688	Bar
Locking Clips	All	Stainless Steel	A 240-63	Tp. 304L	Plate
Baffle Plates	All	Stainless Steel	A 240-63	Tp. 304	Plate
Formers	All	Stainless Steel	A 240-63	Tp. 304	Plate
Barrel-to-Former Bolts	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Locking Pins	All	Stainless Steel	AISI	Tp. 304	Bar
Dowels	All	NiCr Alloy (Alloy X-750)	AMS 5667F	UNS NO7750	Bar
Baffle-to-Former Bolts	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Locking Pins	All	Stainless Steel	AISI	Tp. 304	Bar

Table 2-1 Reactor Vessel Internals Materials of Construction (continued)

Item	Plant	Material	Specification Number	Type or Grade/Class	Product Form
CORE BARREL ASSEMBLY (CBA) (Continued)					
Baffle-to-Former Shoulder Screws	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Locking Dowel	All	Stainless Steel	A 276-65	Tp. 304	Bar
Baffle-to-Baffle Bolts	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Locking Rings	All	Stainless Steel	AISI	Tp. 304	Bar
LOWER INTERNALS ASSEMBLY (LIA)					
Lower Grid Assembly					
Lower Grid Rib Section	All	Stainless Steel	A 240-63	Tp. 304	Plate
Fuel Assembly Support Pads	All	Stainless Steel	A 276-65	Tp. 304	Bar
Dowels	All	NiCr Alloy (Alloy X-750)	AMS 5667F	UNS NO7750	Bar
Cap Screw	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Rib-to-Shell Forging Screw	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Locking Pin	All	Stainless Steel	AISI	Tp. 304	Bar
Lower Grid Flow Distributor Plate	All	Stainless Steel	A 240-63	Tp. 304	Plate
Orifice Plugs	All	Stainless Steel	A 276-65	Tp. 304	Bar
Lower Grid Forging	ANO-1 ONS-2 ONS-3 TMI-1	Stainless Steel	A 473-63	Tp. 304	Forging
(Ribs - ONS-1 only)	ONS-1	Stainless Steel	A 240-63	Tp. 304	Plate
Lower Grid Shell Forging	All	Stainless Steel	A 473-63	Tp. 304	Forging
Lower Internals Assembly-to-Thermal Shield Bolts	ONS-1 ONS-2 ONS-3	NiCr Alloy (Alloy X-750) (replacement)	FTI Proprietary Specification 27-1127244	---	Bar
	TMI-1	NiCr Alloy (Alloy X-750) (original)	A 637-70	Gr. 688	Bar
	ANO-1	Stainless Steel (Alloy A 286) (replacement)	A 453	Gr. 660, Cond A	Bar

Table 2-1 Reactor Vessel Internals Materials of Construction (continued)

Item	Plant	Material	Specification Number	Type or Grade/Class	Product Form
LOWER INTERNALS ASSEMBLY (LIA) (Continued)					
Lower Grid Assembly (continued)					
Locking Cups & Tie Plates (replacement bolts)	ONS-1 ONS-2 ONS-3 ANO-1	Stainless Steel	A 240-74 or A 479-75	Tp. 304L	Plate
Locking Clips (original bolts)	TMI-1	Stainless Steel	A 240-63	Tp. 304L	Plate
Guide Blocks	All	Stainless Steel	A 276-65	Tp. 304	Bar
Guide Block Bolts	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Guide Block Washers	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Dowel	All	NiCr Alloy (Alloy X-750)	AMS 5667F	UNS NO7750	Bar
Shock Pads	ANO-1 ONS-1 ONS-2 ONS-3	Stainless Steel	A 276-65	Tp. 304	Bar
	TMI-1	Stainless Steel	A 240-63	Tp. 304	Bar
Shock Pad Bolts	ANO-1 ONS-1 ONS-2 ONS-3	Stainless Steel	A 193-65	Gr. B-8	Bar
	TMI-1	NiCr Alloy (Alloy X-750)	A 637-70	Gr. 688	Bar
Support Post Pipes	All	Stainless Steel	A 312-64	Tp. 304	Pipe
Bolting Plugs	All	Stainless Steel	A 276-65	Tp. 304	Bar
Support Post Cap Screws	All	Stainless Steel	A 193-65	Gr. B-8	Bar
Locking Pin	All	Stainless Steel	AISI	Tp. 304	Bar

Table 2-1 Reactor Vessel Internals Materials of Construction (continued)

Item	Plant	Material	Specification Number	Type or Grade/Class	Product Form
LOWER INTERNALS ASSEMBLY (LIA) (Continued)					
Flow Distributor Assembly					
Flow Distributor Head	All	Stainless Steel	A 240-63	Tp. 304	Plate
Flow Distributor Flange	All	Stainless Steel	A 473-63	Tp. 304	Forging
Shell Forging-to-Flow Distributor Bolts	ANO-1 ONS-1 ONS-2 ONS-3	Stainless Steel (Alloy A 286)	A 453-65	Gr. 660	Bar
	TMI-1	NiCr Alloy (Alloy X-750)	A 637-70	Gr. 688	Bar
Locking Clip	All	Stainless Steel	A 240-63	Tp. 304	Plate
Incore Guide Support Plate	All	Stainless Steel	A 240-63	Tp. 304	Plate
Clamping Ring	All	Stainless Steel	A 240-63	Tp. 304	Plate
Dowel	All	Stainless Steel	A 276-65	Tp. 304	Bar
Incore Guide Tube Assembly					
Incore Guide Tubes	All	Stainless Steel	A 276-65	Tp. 304	Bar
Gussets	All	Stainless Steel	A 276-65	Tp. 304	Bar
Guide Tube Nuts	All	Stainless Steel	A 276-65	Tp. 304	Bar
Guide Tube Washers	All	Stainless Steel	A 276-65	Tp. 304	Bar
Locking Clips	All	Stainless Steel	A 240-63	Tp. 304	Plate
Spiders	All	Stainless Steel	A 351-65	Gr. CF-8	Casting

Figure 2-1 Plenum Assembly

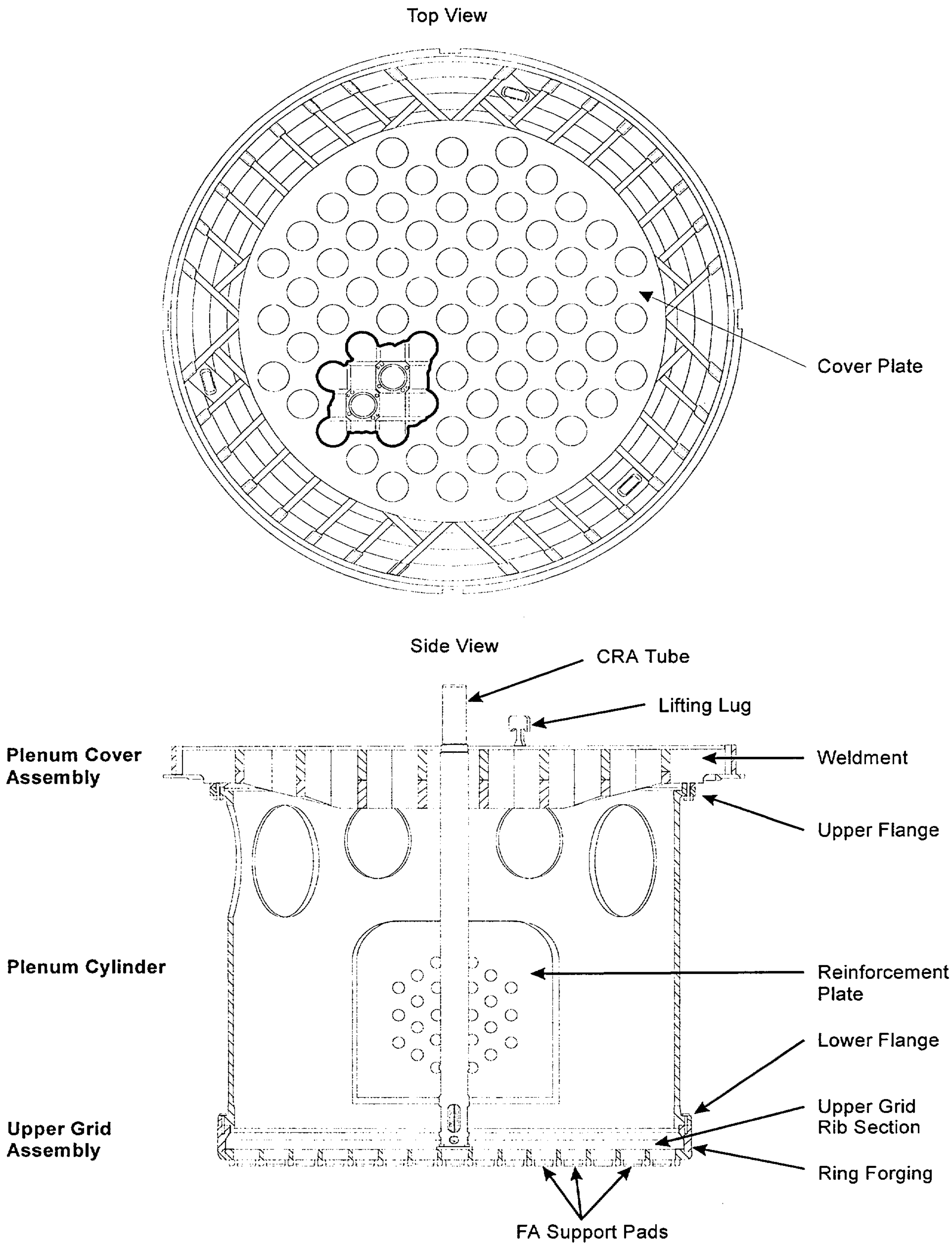


Figure 2-2 Plenum Cover Assembly

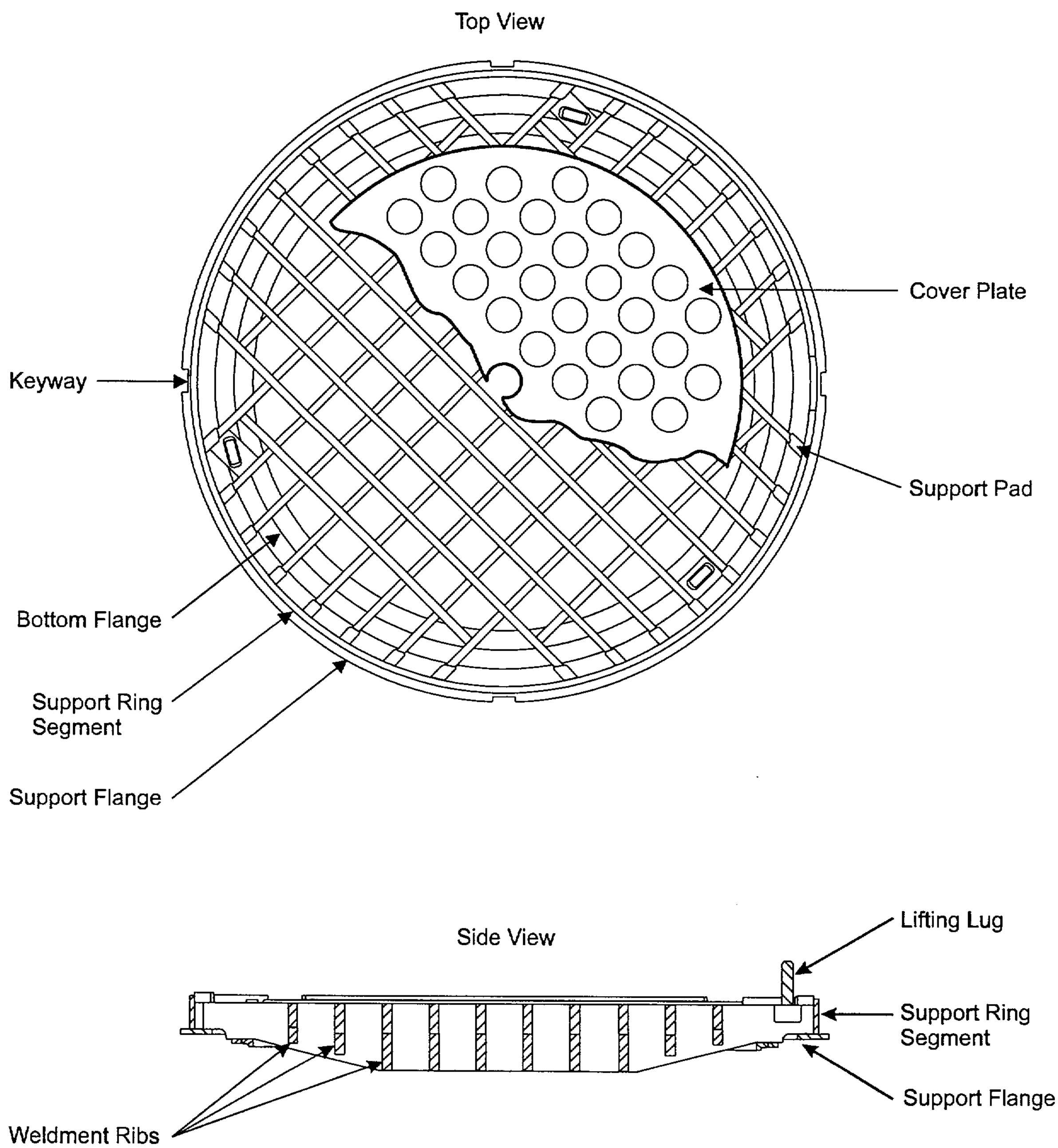
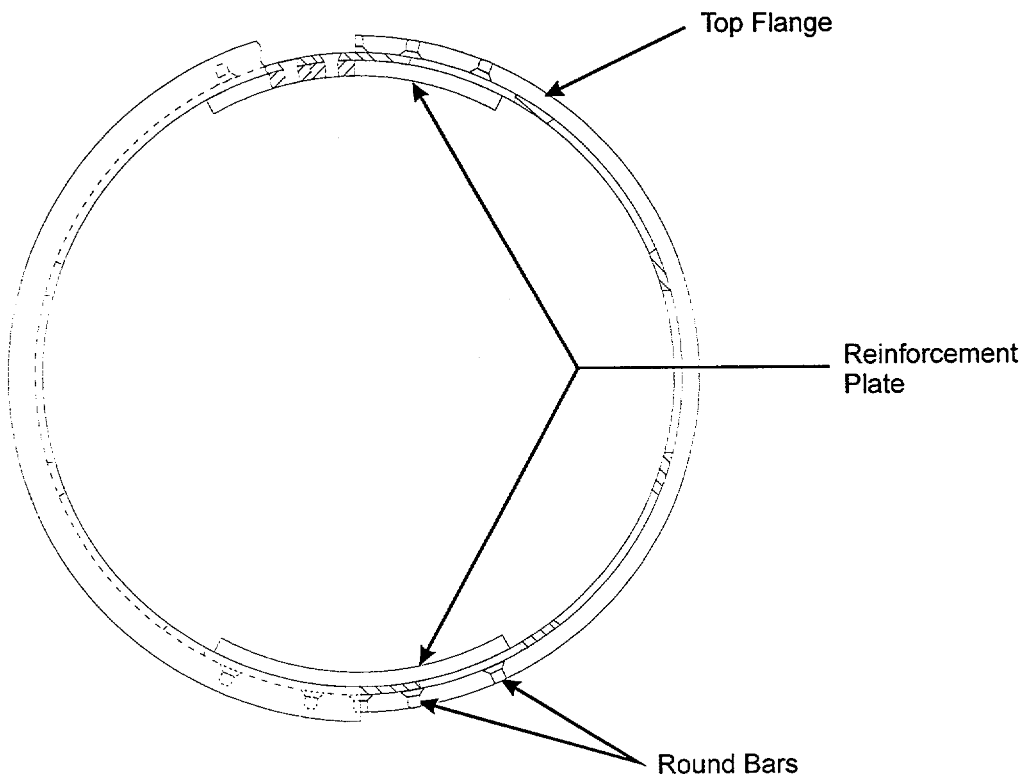


Figure 2-3 Plenum Cylinder Assembly

Top View



Side View

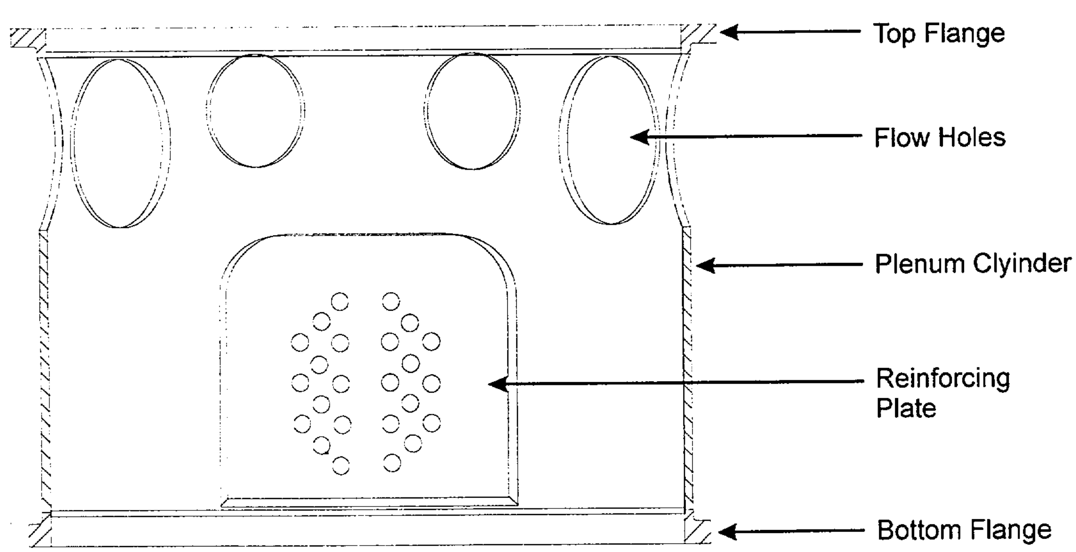


Figure 2-4 Upper Grid Assembly

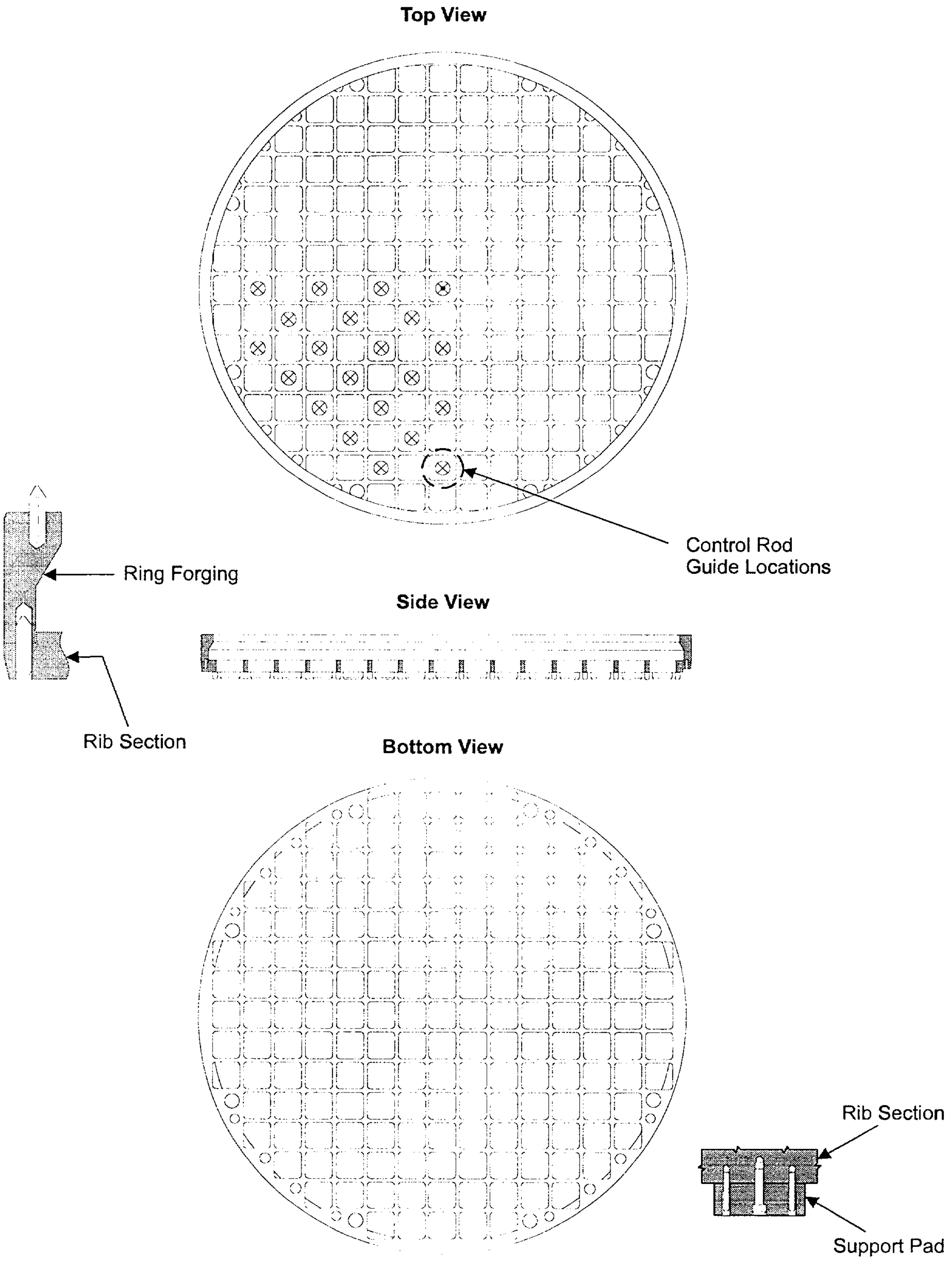


Figure 2-5 Control Rod Assembly (Not in Scope)

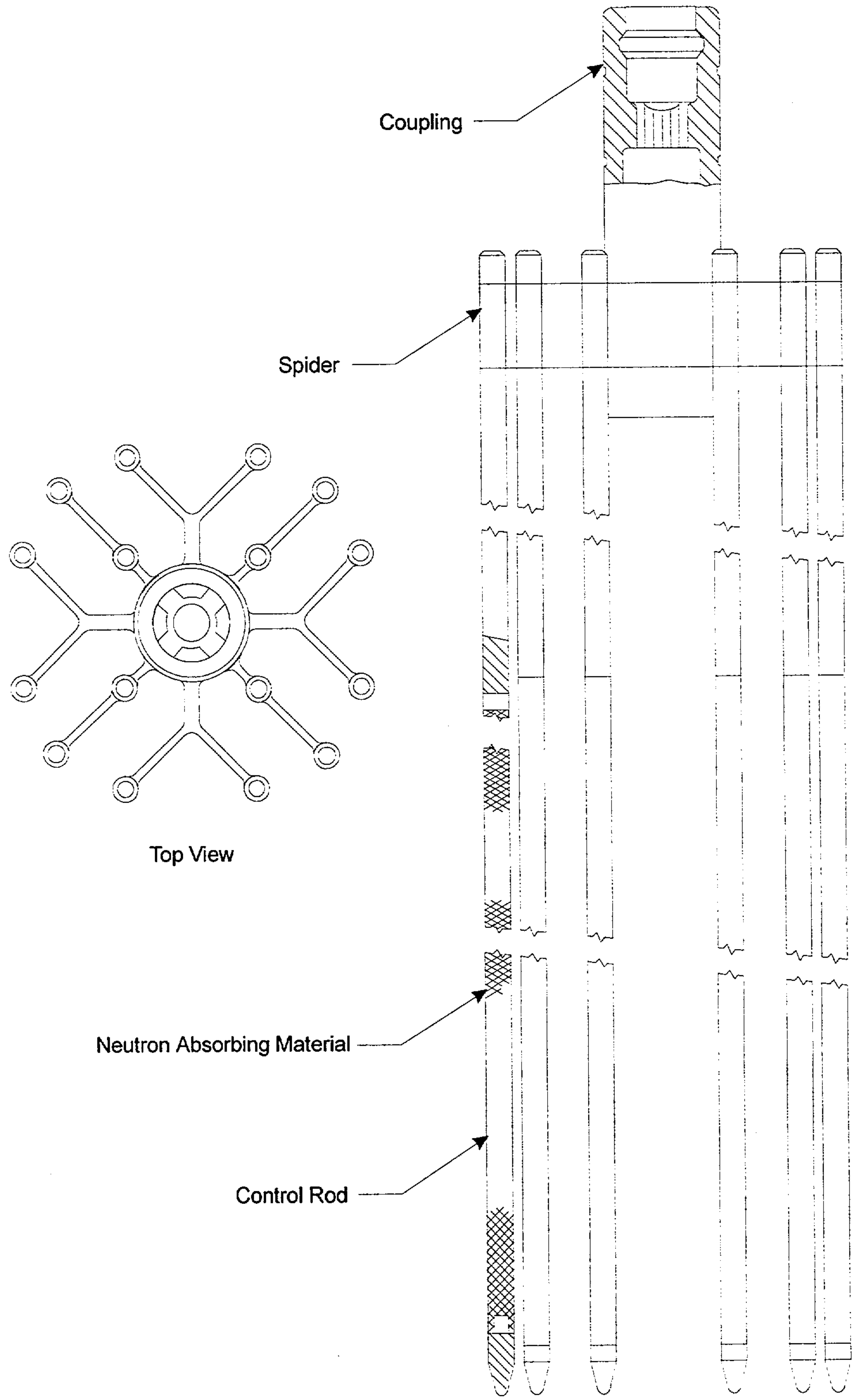


Figure 2-6 Rod Guide Brazement and Spacer

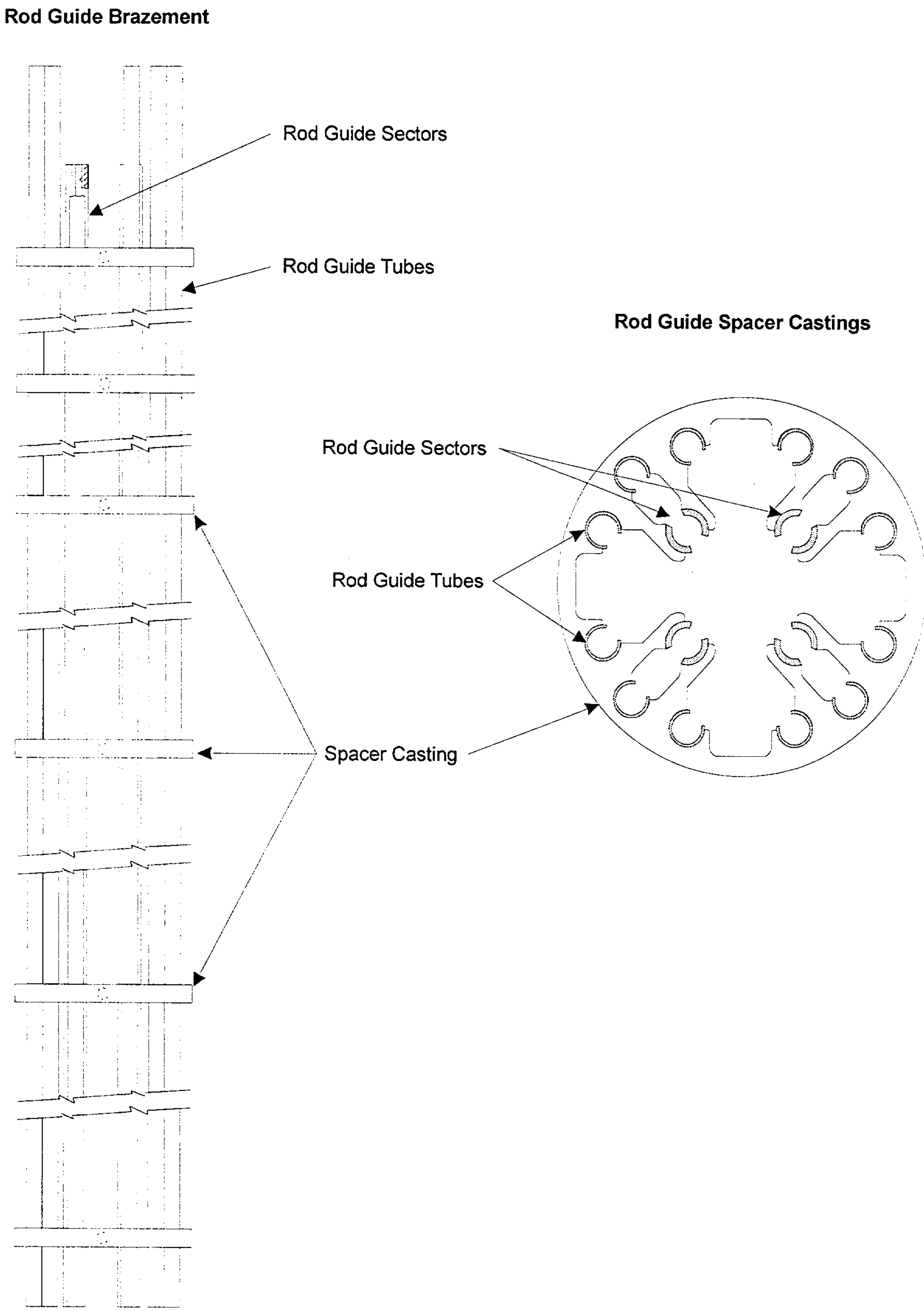


Figure 2-7 Control Rod Guide Tube Assembly

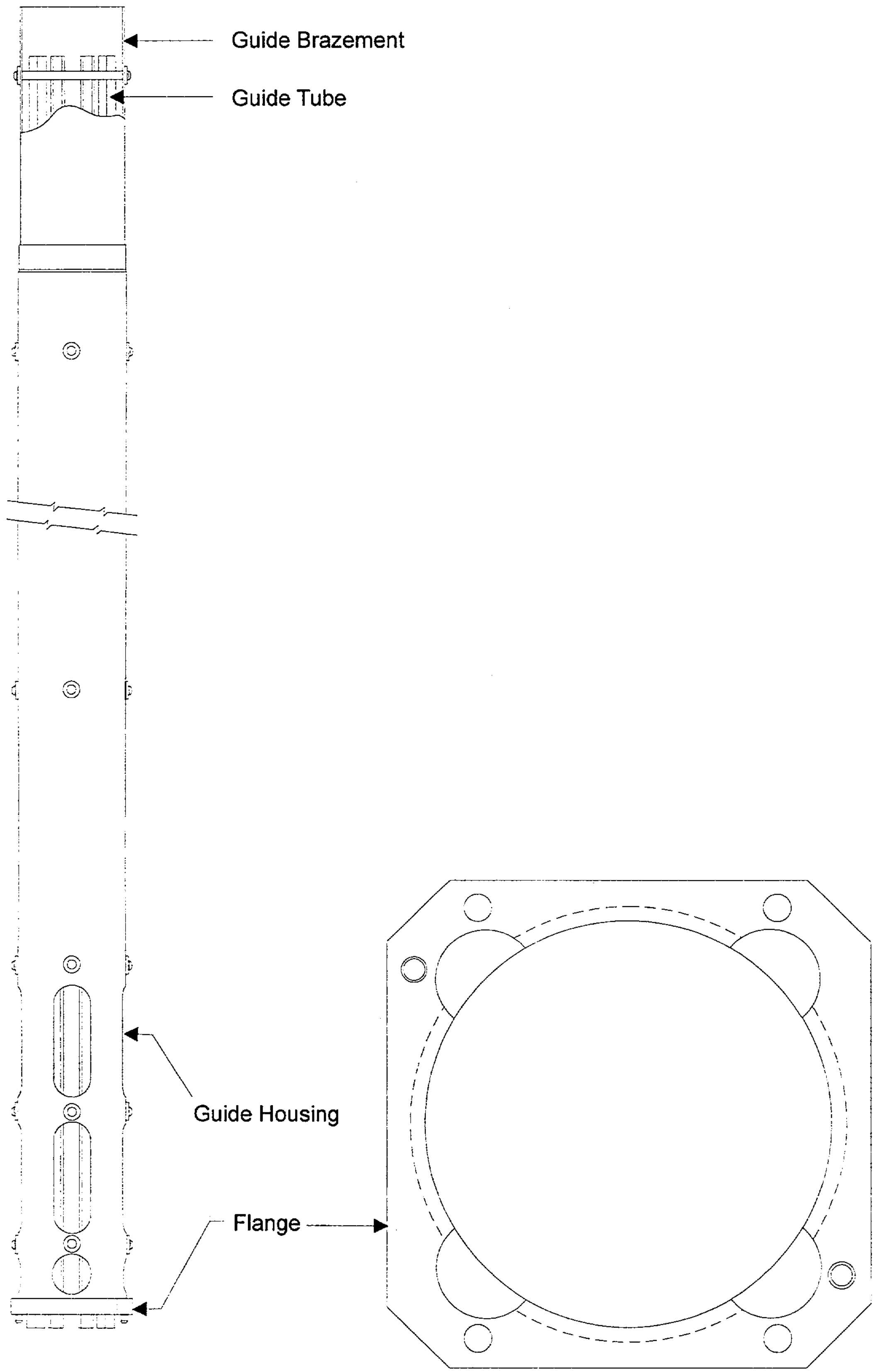


Figure 2-8 Core Support Assembly

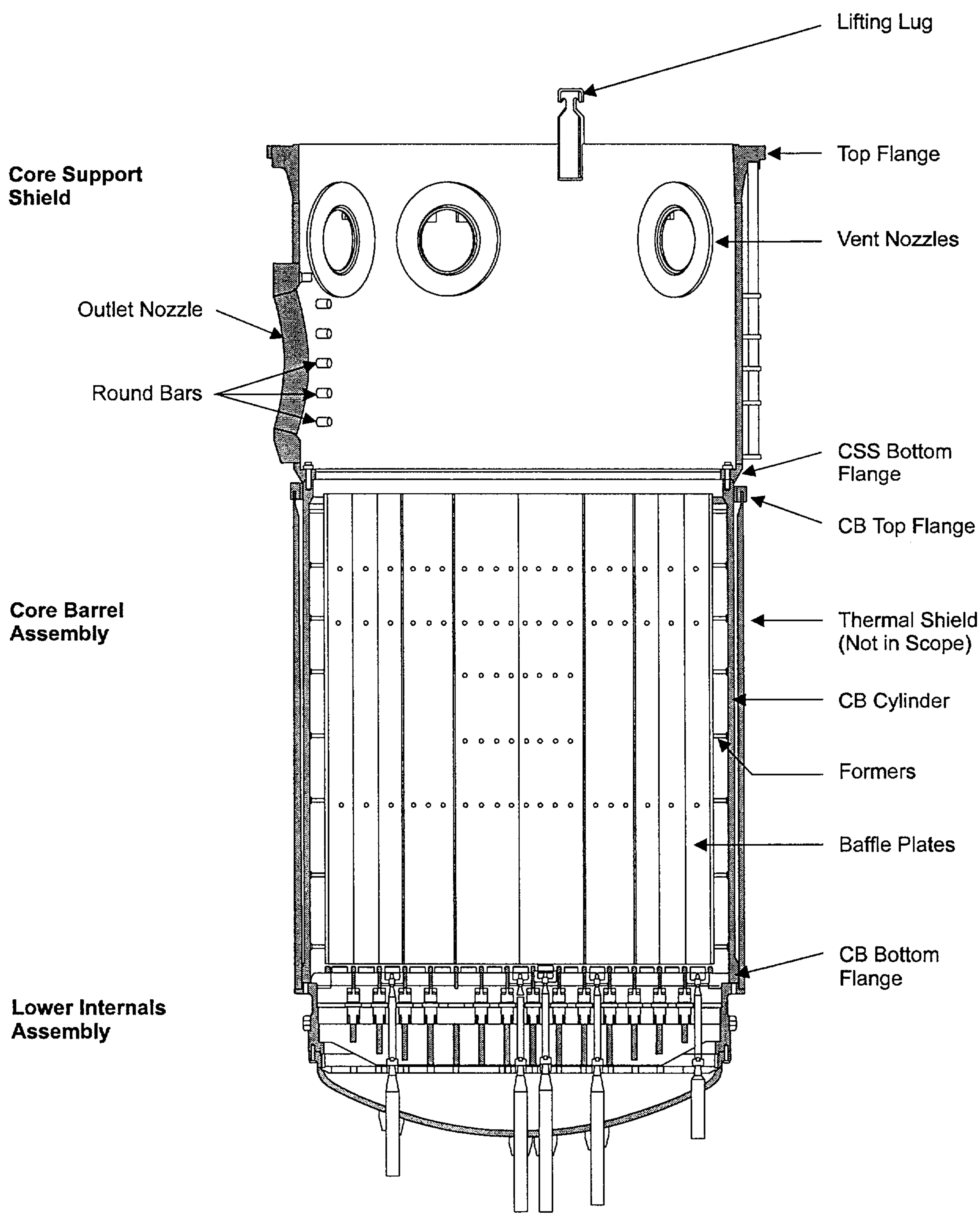


Figure 2-9 Core Barrel Interior

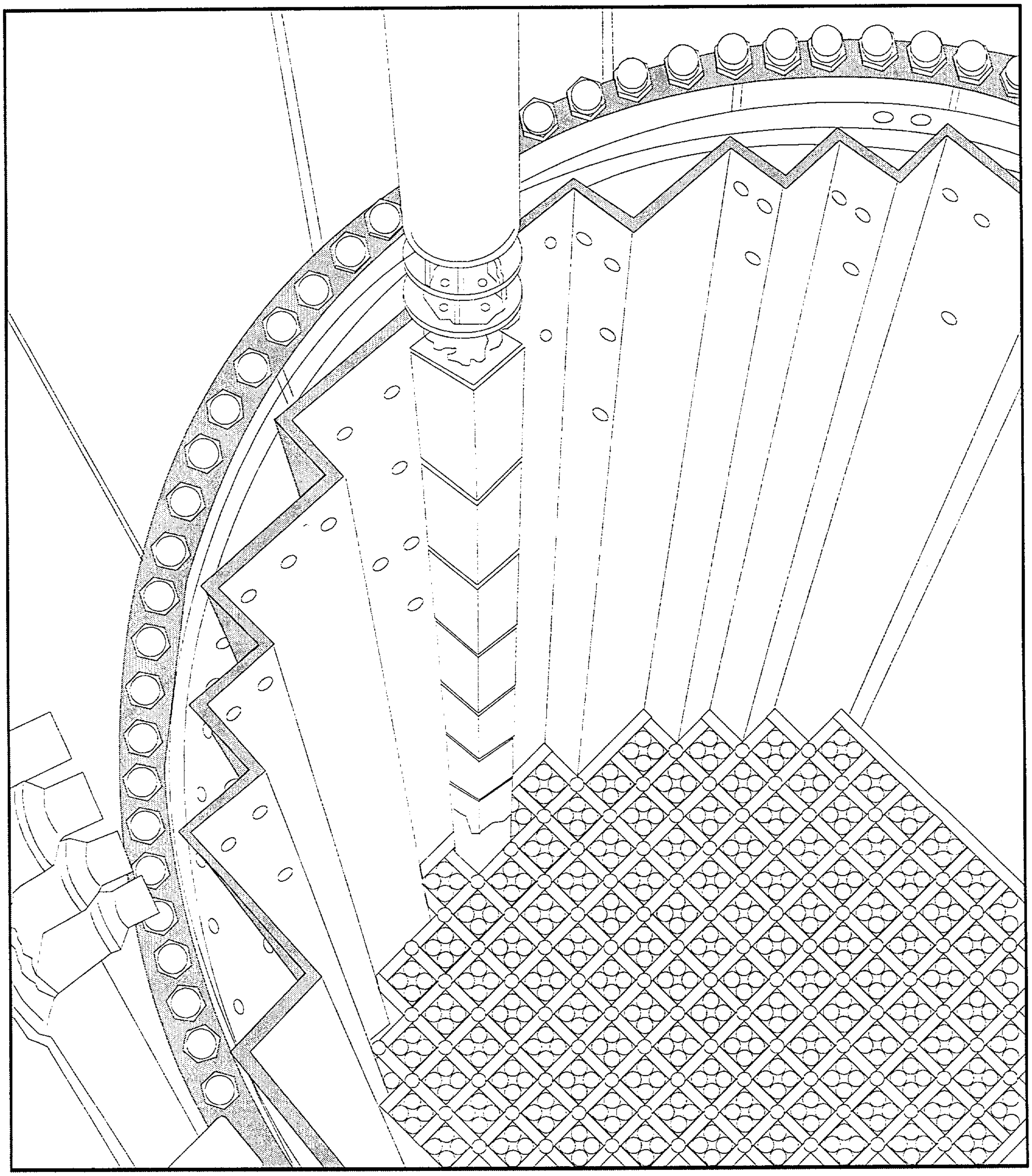


Figure 2-10 Lower Internals Assembly

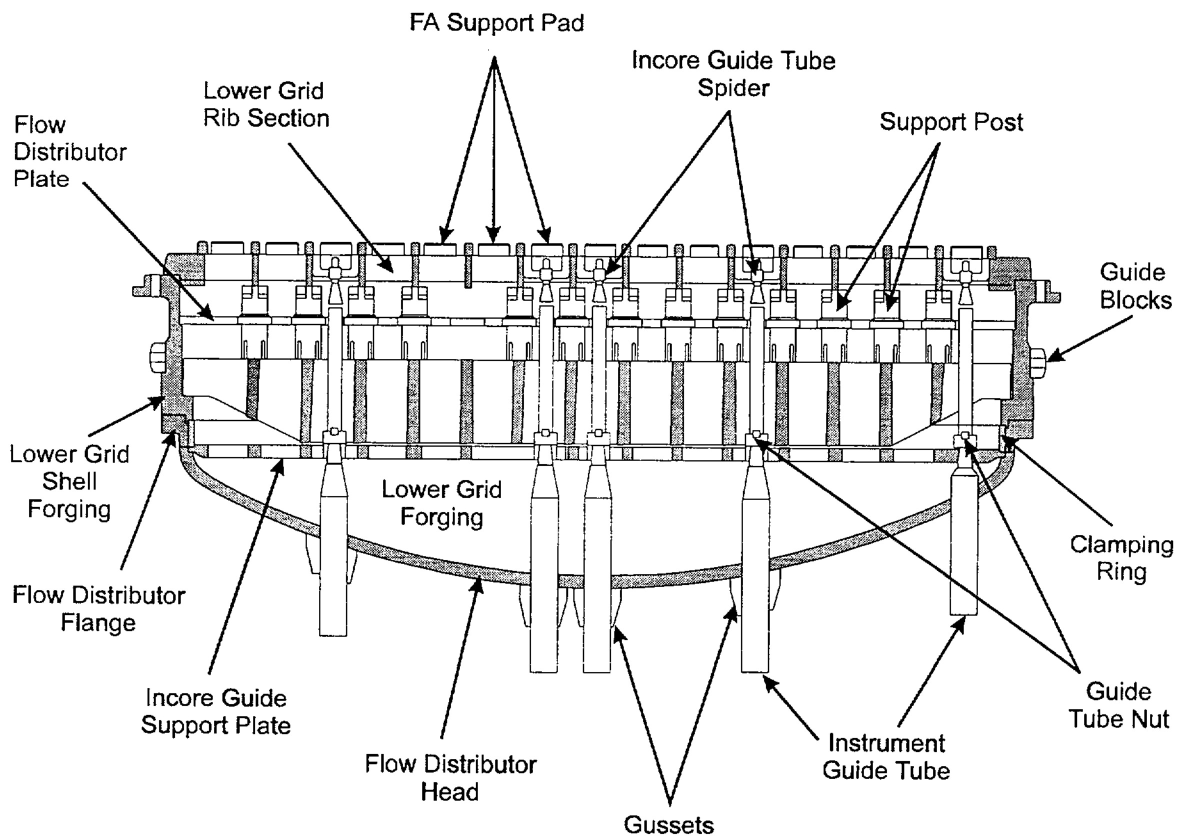
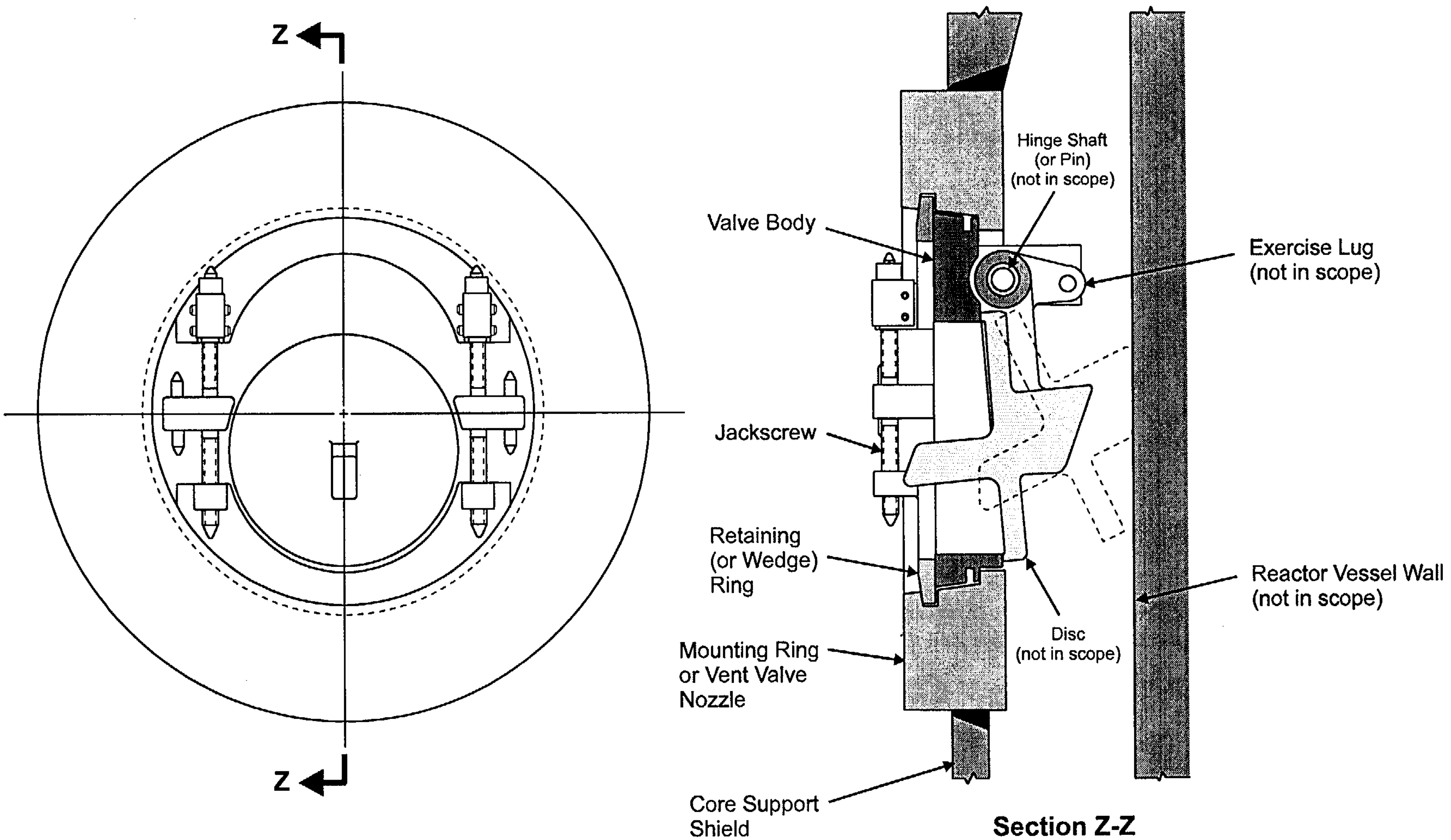


Figure 2-11 Vent Valve Assembly



3. EFFECTS OF AGING ON THE REACTOR VESSEL INTERNALS

This section discusses the aging effects applicable to the reactor vessel internals items for the period of extended operation based on the current design and licensing bases of the B&W participating plants. All items within the scope of the reactor vessel internals report are exposed to RCS water chemistry and stresses associated with Level A and B Service Conditions. The reactor vessel internals components have been designed to accommodate all service loadings (i.e., Levels A-D); however, operation under Service Levels A and B contribute to the normal aging stresses for the reactor vessel internals items. Emergency (Level C) and/or faulted (Level D) events, which have a low probability of occurrence during the life of the plant, do not contribute to normal aging stresses. Aging management for component aging effects combinations that result from these regularly experienced conditions (i.e., Levels A and B) will ensure that the reactor vessel internals items can sustain a Level C or D event during the period of extended operation.

The reactor vessel internals support the following five functions that are within the scope of license renewal as listed in Section 1.4:

1) Provide support and orientation of the reactor core (i.e., the fuel assemblies).

2) Provide support, orientation, guidance and protection of the control rod assemblies.

3) Provide a passageway for the distribution of the reactor coolant flow to the reactor core.

4) Provide a passageway for support, guidance, and protection for the incore instrumentation.

5) Provide a secondary core support for limiting the downward displacement of the core support structure in the event of a postulated failure of the core barrel.

Maintaining the structural integrity of the reactor vessel internals, within the scope of this report, will assure that their component intended functions are maintained so that the RCS may perform its system function(s) within the scope of license renewal in the period of extended operation. The impact of the effects on aging on the structural integrity of the reactor vessel internals is the focus of this section.

The full set of aging effects that could result in loss of the structural integrity of the reactor vessel internals include: (1) cracking (whose stages include crack initiation and growth); (2) reduction of fracture toughness; (3) loss of material (thinning); (4) mechanical distortion and/or ratcheting; and (5) loss of mechanical closure integrity (for bolted connections). The expected temperatures in the reactor vessel internals for normal and upset conditions are well below the threshold values for creep (austenitic stainless steel alloys - 1000°F; nickel base alloys - 1400°F) [3]. Therefore, distortion and ratcheting are not considered to be aging effects requiring further consideration for the reactor vessel internals. [Note: The NRC in its Final Safety Evaluation required that change in dimensions by void swelling be identified as an applicable aging effect and that an aging management program be established to manage this effect. See Section 3.6 for a discussion of void swelling.] Cracking, reduction of fracture toughness, loss of material, and loss of mechanical closure integrity are the aging effects that will be considered in this section for the reactor vessel internals components within the scope of this report.

The material, functional, and operational requirements for the reactor vessel internals scope are initially defined within the component design specifications. The design specifications address loadings and stresses that result from service conditions such as dead weight, earthquake loads, thermal gradients, and thermal expansion. The design specifications define the design envelope for each item, and ensure a margin of safety for the item when operated within the design envelope. Leading to the specific consideration of how the effects of aging will manifest themselves on the hardware, it must also be recognized that current equipment design envelopes have evolved over the life of the plant. Issues have arisen after design that have called into question features in the design. Challenges to the design envelope have been accommodated through plant-specific actions in response to NRC Bulletins (BL), Generic Letters (GL), and Information Notices (IN). These and other generic communications are discussed in Section 3.5.

The commitments that have been implemented as a result of generic communications may be divided into those that are required to validate the design and those that are required to manage aging. Both may contain technical elements to be carried forward to the period of extended operation. The commitments to generic communications required for design validation that should be carried forward to the period of extended operation are discussed in Sections 3.0 through 3.5 and are summarized in Section 3.6. Commitments to generic communications required to manage the aging effects discussed in Sections 3.0 through 3.5 that are to be carried forward to the period of extended operation are discussed in Section 4.0. In all instances, commitments remaining in effect that were made in docketed licensee responses to generic communications must be continued through the period of extended operation unless modified under 10 CFR 50.59.

In general, age related degradation mechanisms manifest themselves in the aging effects noted above, i.e., cracking, reduction of fracture toughness, loss of material, and loss of mechanical closure integrity. Aging mechanisms for the reactor vessel internals items have been studied in detail in the industry report [3]. From this and other references, it is noted that the aging mechanisms that may lead to cracking include stress corrosion (including irradiation-assisted stress corrosion), intergranular attack, void swelling, and fatigue (high or low cycle). Aging mechanisms that may lead to reduction of fracture toughness include both thermal and irradiation induced embrittlement. Aging mechanisms that may lead to loss of material include corrosion (pitting and crevice or uniform attack/general corrosion), erosion and erosion-corrosion, and wear. Finally, aging mechanisms that may result in loss of mechanical closure integrity include loss of preload/stress relaxation, or failure of the bolting materials themselves due to any of the previously mentioned aging effects.

Existing aging management programs have been designed to detect aging effects and not aging mechanisms. The approach presented in the following sections is to define applicable aging effects for items within each of the four major subassemblies based upon materials of construction, environment, and the Level A and B Service Condition stresses.

Aging effects are determined to be applicable:

- if plausible for a given material, environment and stress combination, and,
- if undetected, could result in a condition in which the reactor vessel internals component functions may not be maintained and thus the RCS could not perform its system function(s) in the period of extended operation.

In addition, operating experience, such as NRC generic correspondence, NPRDS, and LER data, were reviewed to confirm applicable aging effects that apply to the reactor vessel internals scope. Therefore, the determination of applicable aging effects have been qualitatively assessed in this evaluation based on operating conditions and operating experience. The applicable aging effects are evaluated against existing aging management programs in Section 4.0. If the applicable effects of aging are detected by existing aging management programs, and the programs contain appropriate acceptance criteria to trigger corrective action, then the reactor vessel internals component intended functions are assured such that the RCS may perform its system function(s) within the scope of license renewal in the period of extended operation.

3.1 Cracking of Reactor Vessel Internals

Aging mechanisms that may lead to cracking of the reactor vessel internals items include stress corrosion cracking (SCC), irradiation-assisted stress corrosion cracking (IASCC), intergranular attack (IGA), void swelling, and fatigue. The various mechanical/fatigue-related mechanisms that could result in cracking are all TLAA and are addressed in Section 4.5. Dimensional changes due to void swelling are believed to be measurable only after significant irradiation-induced changes in material properties (fracture strength, tensile strength, and percent elongation to fracture) have occurred. Void swelling has only been observed and measured in materials that have been irradiated in high flux reactors such as the HFIR (High Flux Isotope Reactor, in Oak Ridge, TN) and in liquid metal cooled fast reactors (e.g., FFTF and EBR-11); no evidence of void swelling under PWR conditions exists. Hence, void swelling of the RV internals is considered to be a non-significant age related degradation mechanism. [Note: The NRC in its Final Safety Evaluation required that change in dimensions by void swelling be identified as an applicable aging effect and that an aging management program be established to manage this effect. See Section 3.6 for a discussion of void swelling.] The susceptibility of the reactor vessel internals austenitic stainless steel items to cracking mechanisms is dependent upon such factors as material composition, manufacturing process, product form, and the operational environment. The susceptibility of the nickel-base alloy dowels and bolting is similarly dependent upon the same factors. The following is a general discussion of the aging mechanisms involved.

SCC, IASCC, and IGA

Stress corrosion cracking (SCC) is produced by the synergistic action of tensile stresses, a specific corrosive environment, and a susceptible material. The process involves both crack initiation and growth. The stresses can be either residual (i.e., resulting from manufacture), applied (i.e., operational), or both. The chemical environmental agents which favor development of SCC are dissolved oxygen, halides, and sulfides. Austenitic stainless steel can be susceptible to SCC, especially if it has been sensitized. Sensitization is generally caused by slow cooling from elevated temperatures during the manufacturing process. The manufacturing processes used to fabricate the reactor vessel internals specifically sought to preclude sensitization of the stainless steels. Depending upon the ferrite content, cast austenitic stainless steel is generally less susceptible to SCC than other product forms.

Irradiation-assisted stress corrosion cracking is a degradation mechanism where materials exposed to radiation become more susceptible to stress corrosion cracking with increasing exposure to neutron irradiation. Stainless steel (e.g., Type 304) internals component items have been shown susceptible to IASCC in boiling water reactors at a neutron fluence threshold of approximately 5×10^{20} n/cm^2 (E > 1 MeV) [3,5,6,10,13]. However, the occurrence of IASCC in PWRs has not been clearly determined. A PWR threshold fluence for IASCC of 1 to 2×10^{21} n/cm^2 (E > 1 MeV), where radiation damage effects begin to be observed, has been estimated [19]. There are no known threshold levels established for PWR environmental conditions. The relatively benign environment of the PWR coolant, which incorporates hydrogen overpressure and achieves reduced oxygen levels in comparison to BWR coolant, reduces (but does not necessarily eliminate) the potential for IASCC degradation of PWR reactor internals component items.

Intergranular attack (IGA), also known as intergranular corrosion, is similar in some respects to SCC; however, it is distinguished from SCC in that stress is not necessary for it to proceed. IGA is characterized by deterioration of grain boundaries without appreciable attack of adjacent grains. That is, the rate of attack on grain boundaries greatly exceeds that of the matrix material. Generally, materials and conditions that are susceptible to intergranular SCC will also be susceptible to IGA.

In general, cracking is not expected to be a significant aging mechanism for the RV internals since there are reactor coolant chemistry controls in place, as required by plant Technical Specifications, to prevent the coolant from becoming an environment favorable to SCC. Dissolved oxygen, halides, and other impurities in the primary coolant are monitored by plant surveillance testing in accordance with plant Technical Specifications (typically every 72 hours or 3 days/week), and are maintained in accordance with the EPRI PWR Primary Water Chemistry Guidelines for all modes of operation at all participating utilities. Corrective action is required by plant procedures or Technical Specifications if the specified limits are exceeded. Actual dissolved oxygen concentrations are usually maintained below 5 ppb by applying a hydrogen overpressure to the coolant system (25-50 cc/kg H_2O). During shutdown, the aerated primary coolant may contain as much as 8 ppm dissolved oxygen, but is below the temperature range where SCC is typically observed [7]. Based on the above chemistry controls, it is concluded that an environment conducive to SCC of austenitic stainless steel component items does not exist in a properly managed primary system.

Cracking, however, is potentially a concern for materials that may have been sensitized due to welding or post-weld heat treatment, materials that have been cold worked, materials that have been exposed to high levels of irradiation, and in areas where crevices can concentrate halogens or oxygen.

Sensitization is the precipitation of chromium carbides, usually at grain boundaries, on exposure to high temperatures (approximately 800°F to 1500°F), leaving the grain boundaries depleted of chromium and susceptible to preferential attack by an oxidizing medium. However, due to strict RCS water chemistry requirements, the susceptibility of the stainless steel reactor vessel internals is expected to be low. In addition, B&W extensively tested austenitic stainless steels exposed to typical RCS boric acid solutions [8]. The tests included low temperature beaker tests, boiling beaker tests, autoclave tests, and dynamic loop tests at temperatures between 220°F and 650°F. Both annealed and sensitized U-bend stainless steel specimens, which were stressed to 75% of the material yield strength, were tested. Test results indicated extremely low corrosion rates and no evidence of stress corrosion cracking.

Cracking of reactor vessel internals items could occur for those items that have been cold worked. The CRGT assemblies are not expected to experience SCC due to the use of cast stainless steel spacers (ferrite content > 5%) and the use of brazing in the fabrication of the assembly. Unlike welding, brazing uniformly heats the entire component and is performed at a lower temperature, thus preventing the accumulation of residual stresses in the base metal. However, the ONS-1 CRGT assembly sectors required straightening after the first hot functional test (HFT). Although straightening may have introduced residual stresses due to cold work from the straightening process, the reactor coolant environment is not conducive to SCC since halogens are controlled to less than 150 ppb and oxygen is controlled to less than 5 ppb during operation. Further, the 48 EFPY fluence at the guide tube location is below the extrapolated PWR threshold for IASCC of $1x10^{21}$ n/cm^2. Therefore, cracking by either SCC or IASCC is not an applicable aging effect for the CRGT assembly sectors.

IASCC is a result of fluence exposure that is highly dependent upon physical location with respect to the core. The baffle/former plates and associated bolts will accumulate the highest neutron fluence of any reactor internals. Table 3-1 shows an estimate of the fluence levels that are expected in the baffle-to-former and baffle-to-baffle bolt locations at the end of 32 and 48 EFPY, respectively. These values are expected to be bounding among the plants participating in this report. The expected fluence on these bolts exceed the current BWR threshold. Therefore, these items can be expected to be the first to experience IASCC if it occurs in a PWR environment. Cracking due to IASCC is considered a potential aging effect for the component items in the vicinity of the core. Specifically, the lead component items affected are the core barrel assembly base metal and welds, baffle-to-baffle bolts, baffle-to-former bolts, core support shield to core barrel bolts, lower internals assembly to core barrel bolts, upper grid assembly base metal and welds, and the lower grid assembly base metal and welds.

Cracking due to SCC is considered a potential aging effect for reactor vessel internals bolts. The local environment in the creviced area associated with bolts may differ from the bulk RCS environment due to stagnant conditions and a potentially higher oxygen concentration from radiolysis affects. The affected bolts are those for which cracking could impact the structural integrity of the reactor vessel internals. These bolts include the core support shield to core barrel bolts, the lower internals assembly to core barrel bolts, the core barrel to thermal shield bolts, the lower internals assembly to thermal shield bolts and the shell forging to flow distributor bolts.

The few historical occurrences of SCC for reactor vessel internals items have been limited to specific age-hardenable alloys (i.e., Alloy X-750 and Alloy A-286[1]) and fabrication induced conditions. The SCC observed in the Alloy X-750 split pins used in Westinghouse-designed internals has been attributed to the very susceptible AH[2] heat treatment and high applied loads [11]. Replacement split pins have been fabricated from Alloy X-750 in the HTH[2] heat treat condition, which is very resistant to SCC in the PWR environment. SCC has occurred in Alloy A-286 internals bolting in B&W designed internals [12]. The Alloy A-286 bolt failures in B&W Owners Group reactor vessel internals were subjected to a comprehensive failure analysis. It was concluded that this material would exhibit SCC when the applied stress approaches its yield strength. Decreasing the initial stress levels on replacement Alloy A-286 bolts has removed this concern. Also replacement bolts fabricated from Alloy X-750 HTH have been installed in the locations where significant bolting failures were detected. Regardless of the material and heat treatment used for those bolted joints (Alloy A-286, replacement bolts with Alloy X-750, or TMI-1 original bolts with Alloy X-750), they are considered potentially susceptible to SCC.

Finally, the locking devices on the modified vent valve assemblies contain several parts that are fabricated from Alloy 600 materials. Alloy 600 components have experienced primary water stress corrosion cracking in the past [4]. As such, the Alloy 600 vent valve locking device parts are considered potentially susceptible to PWSCC.

Summary
In summary, the reactor vessel internals items that are subjected to cracking that require programmatic management during the period of extended operation are:

1) Cracking of the following bolts due to SCC:
   - Core support shield to core barrel bolts
   - Lower internals assembly to core barrel bolts
   - Core barrel to thermal shield bolts
   - Lower internals assembly to thermal shield bolts
   - Shell forging to flow distributor bolts
   - Alloy 600 locking devices on the modified vent valve assembly

---

[1] Except for the TMI unit, all B&W 177-FA plant designs used SA-453 Gr. 660 Condition A or B bolting material, commercially termed Alloy A-286 in the original design of the reactor vessel internals joints. The latter specification is used in Table 2-1 although the following discussions use the more traditional "A-286" designation as it appears in the various references about the bolting problems experienced in the industry in the early 1980's.

[2] AH and HTH are heat treatments used for Alloy X-750 material. AH is defined as hot rolled material, heat treated for 24 hours at 1625°F followed by 20 hours at 1300°F. HTH is defined as hot rolled material, heat treated for 1 hour at 2000° followed by 20 hours at 1300°F.

2) Cracking of the following items due to IASCC:
    - Core barrel assembly base metal and welds
    - Baffle-to-baffle bolts
    - Baffle-to-former bolts
    - Core support shield to core barrel bolts
    - Lower internals assembly to core barrel bolts
    - Upper grid assembly base metal and welds
    - Lower grid assembly base metal and welds

3.2 Reduction of Fracture Toughness in the Reactor Vessel Internals

Reduction of fracture toughness may be an applicable aging effect for some reactor vessel internals due to either thermal or irradiation embrittlement.

Thermal Embrittlement
Thermal embrittlement is a potential aging effect for cast austenitic stainless steel and martensitic stainless steel items exposed to reactor operating temperatures. Thermal embrittlement increases hardness and tensile strength and decreases ductility, impact strength, and fracture toughness of these materials.

Table 2-1 shows that the reactor vessel internals items that were fabricated from cast austenitic stainless steels are the CRGT assembly spacer castings, the vent valve bodies, the core support shield outlet nozzles at ONS-3, and the incore guide tube spider castings. The CRGT spacer castings support the guide tubes and guide sectors. The guide tubes and guide sectors provide a path for travel of the control rods and spider. Failure of the castings could impede control rod motion. The incore guide tube spider castings provide support for the incore guide tubes. The guide tubes guide the incore instrument assemblies from the instrument nozzles in the RV bottom head to the instrument tubes in the fuel assembly. Failure of these spider castings could result in the loss of the ability to insert the incore nuclear detectors and thermocouples. The body of the internal vent valve provides a seal along with the disc to prevent flow from the downcomer side of the CSS to the plenum side. Failure of this part along with the CSS outlet nozzles will cause unwanted core bypass flow. In addition, the martensitic materials in the reactor vessel internals include the vent valve retaining rings and as such, are susceptible to thermal embrittlement. Therefore, thermal embrittlement is considered a potential aging effect for the CRGT assembly spacer castings, the incore guide tube spider castings, the internal vent valve bodies, vent valve retaining rings, and the ONS-3 core support shield outlet nozzles.

Irradiation Embrittlement
The high neutron fluence levels to which the core support assembly items will receive are expected to lead to irradiation embrittlement concerns. Decreased toughness and loss of fatigue resistance are the expected consequences of irradiation embrittlement. Tests conducted in water on Type 304 SS at PWR operating temperatures indicate that the effects of irradiation

embrittlement become noticeable at a fluence level of $\sim 5x10^{20}$ n/cm^2 [5]. Reduction of fracture toughness due to irradiation embrittlement is considered a potential aging effect for the components in the vicinity of the core. These items include the core barrel assembly base metal and welds, baffle-to-baffle bolts, baffle-to-former bolts, core support shield to core barrel bolts, lower internals assembly to core barrel bolts, upper grid assembly base metal and welds, and the lower grid assembly base metal and welds. As with IASCC, it is expected that the baffle/former plates and bolts (based on fluence estimate in Table 3-1) will be the lead items relative to this mechanism.

Summary
In summary, the reactor vessel internals items that are subjected to reduction of fracture toughness that require programmatic management during the period of extended operation are:

1) Reduction of fracture toughness of the following items due to thermal embrittlement:
    - CRGT Assembly Spacer Castings
    - Incore Guide Tube Spider Castings
    - ONS-3 core support shield outlet nozzles
    - Internal vent valve bodies
    - Internal vent valve retaining rings

2) Reduction of fracture toughness of the following items due to irradiation embrittlement:
    - Core barrel assembly base metal and welds
    - Baffle-to-baffle bolts
    - Baffle-to-former bolts
    - Core support shield to core barrel bolts
    - Lower internals assembly to core barrel bolts
    - Upper grid assembly base metal and welds
    - Lower grid assembly base metal and welds

3.3 Loss of Material in the Reactor Vessel Internals

There are two generic types of aging mechanisms which could manifest themselves as a loss of material in reactor vessel internals: mechanical mechanisms (e.g., wear, erosion) and corrosion.

Erosion and Erosion-Corrosion
Susceptibility to loss of material due to erosion or erosion-corrosion is determined by the materials of construction, fluid velocity and extent of suspended particulate matter, and the occurrence of such phenomenon as cavitation. All RV Internals are fabricated from stainless steel or nickel-base alloys. Stainless steels and nickel-base alloys are resistant to erosion and erosion-corrosion in a PWR environment [3]. Therefore, loss of material due to erosion and erosion-corrosion is not considered an aging effect for the reactor vessel internals components.

Uniform Attack/ General Corrosion
Loss of material due to uniform attack/general corrosion is not considered a potential aging effect for RV Internals. All RV Internal items are fabricated from stainless steel or nickel-base alloys.

Stainless steels and nickel-base alloys are resistant to uniform attack/general corrosion in a PWR environment because they passivate to form protective layers which mitigate the potential for corrosion degradation [3].

Pitting and Crevice Corrosion

Pitting and crevice corrosion are generally associated with stagnant or low flow conditions. Pitting corrosion can be considered a special instance of crevice corrosion in that when a pit is formed, it essentially becomes a crevice. Corrosion in crevices may be caused by: (1) an increase in metal ion concentration with the crevice as compared with the concentration outside the crevice (concentration cell corrosion), (2) a decrease in oxygen concentration inside the crevice (oxygen concentration cell corrosion), or (3) increased corrodent activity resulting from the accumulation of corrosion products within the crevice (stagnant area corrosion). All three of these mechanisms are the result of restricted fluid circulation throughout the crevice. Literature surveys of stainless steel corrosion reveal that the presence of a crevice condition would not greatly increase the corrosion rate of stainless steels in a PWR environment due to the low oxygen levels present during reactor operation [14,15,16]. During shutdown, aerated primary coolant can have dissolved oxygen contents of approximately 8 ppm when the reactor vessel head is removed for refueling; however, impurities are controlled during both refueling and extended outages at levels that will preclude crevice or pitting corrosion. Similar conclusions can be expected for Ni-base alloys.

Wear

Wear or fretting can result in the loss of material due to mechanical abrasion in circumstances where items are physically in contact and must move, either by design or due to FIV. Plant operating conditions usually determine the severity of wear. Loss of material due to wear is not considered a potential aging effect for bolting provided the bolts continue to maintain sufficient preload, as will be discussed under stress relaxation in Section 3.4.

Loss of material due to wear in the plenum assembly may be significant at the interface of the plenum rib pad to the reactor vessel closure head, and at the interface between the upper grid fuel assembly support pads and fuel. Wear at these locations may be a result of installation and removal of the plenum assembly and fuel and also due to flow-induced vibration and thermal effects.

Wear is a potential aging mechanism at the interface of the control rod guide tubes and sectors with the control rods. The control rods remain either "parked" in position within the guide tube assembly or are inserted and withdrawn through the guide tubes/sectors to control core power. The insertion and withdrawal forces and FIV while in the parked position may be a cause for wear; however, no significant wear, to date, has been observed in these components and is expected to be insignificant over the lifetime of the reactor vessel internals.

Loss of material due to wear in the core support shield assembly is a potential aging mechanism at the interface of the CSS top flange and the reactor vessel. Like the plenum rib pads, wear may occur due to installation and removal of the core support assembly.

In the 1978 - 1980 time frame, vent valve inspections revealed signs of wear on the locking devices. This resulted in a redesign of the locking devices (i.e., modified condition) to eliminate this problem. The modified vent valve locking assembly was installed only for those vent valves whose inspection revealed wearing of the locking device. As such, wear is a potential aging effect for the locking devices on the original vent valve assemblies.

In the lower internals, wear is a potential aging mechanism at the interface of the lower grid fuel assembly support pads and the fuel. Similar to the upper grid location, wear may be a result of fuel installation and removal plus flow-induced vibration and thermal effects. Wear of the guide blocks is also a potential aging mechanism since wear of the guide blocks against the guide lugs has been noticed during a 10-year inspection. See Section 3.5.3 for additional information.

The inner surfaces of the incore guide tubes and spiders could be subject to loss of material through wear when the incore monitors are installed and removed during outages. Potential for loss of material due to wear of the guide tubes exists at the location where the incore detector exits the guide tube assembly and enters the fuel assembly. Westinghouse has observed wear of the guide tubes due to FIV in this region since it is directly exposed to RCS flow. However, because of design differences between the B&W and Westinghouse designs and given the fact that the B&W-designed detectors are inserted and withdrawn once per fuel cycle, wear of the guide tubes and spiders is expected to be insignificant over the lifetime of the reactor internals.

Summary
In summary, the reactor vessel internals items that are subjected to loss of material that require programmatic management during the period of extended operation are:

1) Loss of material of the plenum rib pads due to wear.

2) Loss of material of the fuel assembly support pads on the upper grid assembly due to wear.

3) Loss of material of the top flange on the core support shield cylinder due to wear.

4) Loss of material of the fuel assembly support pads on the lower grid assembly due to wear.

5) Loss of material of the guide blocks on the lower grid assembly due to wear.

6) Loss of material of the locking devices on the original vent valve assembly.

3.4 Loss of Closure Integrity in the Reactor Vessel Internals

Stress Relaxation
Stress relaxation is caused by long term exposure of the internals materials to elevated temperatures or neutron irradiation. At temperatures well above the operating temperatures, the

thermal effect is predominant. However, even at low temperatures, the presence of neutron irradiation can result in stress relaxation [3].

Stress relaxation is considered an applicable aging mechanism for those component items whose maintaining a preload is important to the structural integrity function of the reactor vessel internals. These bolts include:

- CRGT flange to upper grid screws
- Core support shield to core barrel bolts
- Core barrel to thermal shield bolts
- Lower internals assembly to core barrel bolts
- Lower grid rib-to-shell forging screws
- Shell forging to flow distributor bolts
- Lower internals assembly to thermal shield bolts
- Baffle and former bolts

Summary
In summary, the reactor vessel internals items that are subjected to the loss of closure integrity that require programmatic management during the period of extended operation are:

1) Loss of closure integrity of the CRGT flange to upper grid screws due to stress relaxation.

2) Loss of closure integrity of the core support shield to core barrel bolts due to stress relaxation.

3) Loss of closure integrity of the core barrel to thermal shield bolts due to stress relaxation.

4) Loss of closure integrity of the lower internals assembly to core barrel bolts due to stress relaxation.

5) Loss of closure integrity of the lower grid rib-to-shell forging screws due to stress relaxation.

6) Loss of closure integrity of the shell forging to flow distributor bolts due to stress relaxation.

7) Loss of closure integrity of the lower internals assembly to thermal shield bolts due to stress relaxation.

8) Loss of closure integrity of the baffle and former bolts due to stress relaxation.

3.5 Reactor Vessel Internals Performance History

For the historical review of the reactor vessel internals, a review of the Nuclear Plant Reliability Data System (NPRDS) and Licensee Event Reports (LERs) from July 1974 through September 1994 (allowing a six-month delay in reporting to the data base) was performed to identify past incidents of aging effects applicable to the reactor vessel internals.

3.5.1 Cracking in Reactor Vessel Internals

NPRDS Data

A search of the NPRDS was conducted for problems associated with Reactor Vessel Internals at B&W Reactors in the United States from July 1, 1974 through the present. This review identified only one reported incidence. In 1988 a bolt holding the thermal shield to the lower core barrel at a non-B&W-designed plant was found to be severed. This was attributed to transgranular fracture.

Licensee Event Reports

No LERs were located that discussed instances of cracking in reactor vessel internals except for the bolting problems that are discussed in Section 3.5.4.

NRC Generic Letters, Bulletins and Information Notices

The following Information Notices provided licensees with information regarding cracking of stainless steel and/or nickel-base alloy components with relevance to the items within the scope of the reactor vessel internals report:

IN 82-14	TMI-1 Steam Generator/Reactor Coolant Chemistry/Corrosion Problem,
IN 83-49	Sampling and Prevention of Intrusion of Organic Chemicals into Reactor Coolant Systems,
IN 84-18	Stress Corrosion Cracking in Pressurized Water Reactors,
IN 90-68	Intergranular Stress Corrosion Cracking of Reactor Coolant Pump Bolts

In general, cracking due to SCC is precluded in the reactor vessel internals stainless steel items due to the strict operational control of RCS water chemistry. INs 82-14, 83-49 and 84-18 prove by exception how well the chemistry control is maintained by pointing out those few instances when the RCS chemistry has been known to have been contaminated.

Information Notice 82-14 alerted licensees to instances of sulfur compounds that were inadvertently added to the primary system during an extended outage, which resulted in degradation of steam generator tubes. Inadvertent addition of organic contaminants, such as glycol and cleaning solvents, to the RCS was discussed in IN 83-49. The introduction of corrodants to the RCS through contaminants in purchased boric acid and at the free surface of the spent fuel pool was discussed in IN 84-18. Reactor coolant water chemistry requirements include a check for sulfates, organics, and other contaminants that can cause SCC during all modes of operation, including shutdown and refueling.

Information Notice 90-68 provides information about SCC cracking in Alloy A-286 bolts used to hold the turning vanes to the RCP at a foreign plant. The IN includes a general discussion of the problems experienced with cracking of Alloy A-286 bolting materials, including the problems discussed in this report with the B&W reactor vessel internals bolting.

No NRC IE Bulletins provided licensees with information directly relevant to cracking of items within the scope of the reactor vessel internals report.

The following Generic Letter provided licensees with information regarding cracking of components within the scope of the reactor vessel internals report:

GL 91-17	Generic Safety Issue (GSI) 29, "Bolting Degradation or Failure in Nuclear Power Plants"

Generic Letter 91-17 addresses degradation of threaded fasteners and is discussed in Section 3.5.4.

Summary
In summary, the review confirmed that cracking is an applicable aging effect for the Alloy A-286 bolting within the scope of the reactor vessel internals report for the period of extended operation.

3.5.2 Reduction of Fracture Toughness in Reactor Vessel Internals

NPRDS, LERs, and NRC communications were reviewed to determine instances of reduction of fracture toughness. No reported instances of failures of components within the reactor vessel internals scope due to reduction of fracture toughness were reported.

3.5.3 Loss of Material in Reactor Vessel Internals

NPRDS Data
A search of the NPRDS was conducted for problems associated with reactor vessel internals at B&W reactors in the United States from July 1, 1974 through the present. This review identified no reported incidences of loss of material in reactor vessel internals.

Licensee Event Reports
A review of LERs identified the following events of direct relevance to the loss of material in reactor vessel internals:

Guide Blocks
During the 10 year ISI conducted after the sixth fuel cycle at a B&W-designed plant, visual inspection of the inside surface of the reactor vessel revealed that the twelve guide lugs showed the imprint of the mating guide blocks. The depths of the imprints were not measured but a close review of the videotapes of the inspection determined that they were small (on the order of a few mils). A possible cause for this condition is wear at the clamping surfaces causing a loss of preload on the internals and an increase in their vibratory motion. Therefore detailed measurements of the clamping area were taken and compared to as-built dimensions taken before plant operation. These inspections showed that there had been no wear of the clamping surfaces (within the accuracy of the measurements) during the first six fuel cycles. The contact marks on the guide lugs were probably due to normal internals motion during contact between the lugs and blocks. This contact can occur because of bowing of the internals due to minor temperature differences across their diameter.

NRC Generic Letters, Bulletins and Information Notices
No generic NRC communications were located with direct relevance to loss of material of reactor vessel internals.

Summary
In summary, the review noted that loss of material is an applicable aging effect for the guide blocks for the period of extended operation.

3.5.4 Loss of Mechanical Closure Integrity in Reactor Vessel Internals

NPRDS Data
A search of the NPRDS was conducted for problems associated with Reactor Vessel Internals at B&W Reactors in the United States from July 1, 1974 through the present. This review identified only one reported incidence as described in Section 3.5.1.

Past Availability and Plant Performance Committee Reports
The B&W Owners Availability Reports from 1984 to 1989, and the 1992 and 1993 Plant Performance Reports, indicate that there have been very few lost Effective Full Power Hours (EFPH) due to RV Internals problems. The only significant loss of EFPH during the above time period was approximately 250 to 350 hours per plant for 10 year ISI inspections of the reactor vessel and internals, and was part of a planned outage.

Licensee Event Reports
The following instances of loss of mechanical closure integrity in reactor vessel internals were reported in LERs:

Thermal Shield Bolts
In the early 1980's, inservice inspections at several B&W plants revealed that lower internals assembly-to-thermal shield bolts were missing. The majority (~80%) of the remaining bolts were loose, and several bolt locking cups were also missing. The bolts were fabricated from high-strength grade Alloy A-286 stainless steel. The failures were attributed to IGSCC at the

bolt-head-to-bolt-shank transition. The replacement bolts were redesigned to reduce the tensile stress level in the bolt, and this was accomplished by redesigning the shank region, peening the surface of the bolt, and reducing the preload used to install the bolts. The material of construction was also changed from Alloy A-286 stainless steel to Alloy X-750 [12].

Core Barrel Bolts
In 1983, ultrasonic inspections at two B&W-designed plants showed indications of cracking in a number of the core support shield to core barrel bolts. The results were verified when the bolt heads became separated from the bolt shanks when the locking clips were removed. The bolts were fabricated from Alloy A-286. Failures were attributed to IGSCC and were not detected by visual examinations. The cracked bolts were replaced by bolts made of the same material, but manufactured by machining rather than a hot-heading operation. In addition, the torque applied to the replacement bolts during installation was significantly reduced [12].

NRC Generic Letters, Bulletins and Information Notices
The only notices and bulletins located with direct relevance to loss of closure integrity concerned boric acid wastage of low alloy steel bolting, which is not directly relevant to the stainless steel and nickel-base alloy bolting used in reactor vessel internals.

The following Generic Letter provided licensees with information regarding loss of mechanical closure integrity that applies to the reactor vessel internals scope:

GL 91-17 Bolting Degradation or Failure in Nuclear Power Plants.

Generic Letter 91-17 provided licensees with information on GSI 29 which addressed degradation of all safety-related bolts, studs, embedments, machine cap screws, and other threaded fasteners. The NRC concluded that existing requirements, in combination with actions taken in response to industry initiatives, are sufficient to assure integrity of safety-related bolting.

Summary
In summary, the review confirmed that the loss of mechanical closure integrity is an applicable aging effect for the bolting within the scope of the reactor vessel internals report for the period of extended operation.

3.6 Summary of Aging Effects Applicable to the Reactor Vessel Internals

Based on the identification of applicable aging effects for the material, environment, and stress combinations, as discussed in Sections 3.1 through 3.4 and further substantiated by the historical review in Section 3.5, the aging effects applicable to the reactor vessel internals were identified.

Specific aging effects applicable to the plenum assembly include: (1) cracking and reduction of fracture toughness of the upper grid assembly base metal and welds; (2) loss of material of the fuel assembly support pads and the plenum rib pads; (3) loss of closure integrity of the CRGT flange to upper grid screws; and (4) reduction of fracture toughness of the CRGT assembly spacer castings.

Applicable aging effects for the core support shield assembly include: (1) loss of material of the core support shield cylinder top flange; (2) cracking, reduction of fracture toughness, and loss of closure integrity of the core support shield to core barrel bolts; (3) reduction of fracture toughness of the ONS-3 outlet nozzles; (4) reduction of fracture toughness of the internal vent valve bodies and retaining rings; (5) cracking of the Alloy 600 locking devices on the modified vent valve assembly; and (6) wear of the locking devices on the original vent valve assemblies.

Applicable aging effects for the core barrel assembly include: (1) cracking and reduction of fracture toughness of the core barrel assembly base metal and welds; (2) cracking, reduction of fracture toughness, and loss of closure integrity of the baffle-to-baffle bolts and the baffle-to-former bolts; (3) cracking, reduction of fracture toughness, and loss of closure integrity of the lower internals assembly to core barrel bolts; and (4) cracking and loss of closure integrity of the core barrel to thermal shield bolts.

Applicable aging effects for the lower internals assembly include: (1) cracking and reduction of fracture toughness of the lower grid assembly base metal and welds; (2) cracking and loss of closure integrity of the lower internals assembly to thermal shield bolts and the shell forging to flow distributor bolts; (3) loss of material of the fuel assembly support pads and the guide blocks; (4) loss of closure integrity of the lower grid rib to shell forging screws; and (5) reduction of fracture toughness of the incore guide tube spacer castings.

The NRC, in its Final Safety Evaluation noted that change in dimensions by void swelling must also be addressed. The GLRP in its response to the draft safety evaluation noted that components of the core barrel assembly (i.e., baffle bolts, former plates, and baffle plates) are considered susceptible items to dimensional change by void swelling during the period of extended operation. The aging effects of these components will be carried forward to Sections 4.0 and 5.0 of this report.

As discussed in Section 3.0, the GLRP delineates between those commitments required to validate the design of items within the scope of the reactor vessel internals report and commitments required to manage aging. Both may include technical elements that will be carried forward to the period of extended operation. No commitments associated with design validation of the reactor vessel internals were identified that will be carried forward to the period of extended operation.

Technical Specifications or the Final Safety Analysis Report (FSAR) (for plants using Standard Technical Specifications) require that all plants maintain chloride, fluoride, and oxygen concentration in the RCS within prescribed limits so that the RCS is protected against SCC. If chloride, fluoride, or oxygen limits are exceeded, corrective measures are required to reduce concentrations below the allowable limits. The reactor coolant chemistry requirements are required to ensure that the design basis of components within the scope of the reactor vessel internals report and must be continued in the period of extended operation.

A summary of applicable aging effects for the scope identified in Section 2.0 is provided in Table 3-2. The applicable aging effects, as determined by the evaluation in Section 3.1 through 3.5, will then be compared to the technical elements of existing aging management programs to determine if the applicable aging effects are adequately managed for the period of extended operation. This programmatic review is provided in Section 4.0.

Table 3-1 Estimated Fluence at Baffle/Former Bolt Locations

Item	Estimated Fluence (n/cm^2)	
	After 32 EFPY	After 48 EFPY
Top of Core		
Baffle-to-Former Bolts	1.72E + 21	2.58E + 21
Baffle-to-Baffle Bolts	6.09E + 20	9.14E + 20
Middle of Core		
Baffle-to-Former Bolts	3.33E + 22	4.99E + 22
Baffle-to-Baffle bolts	4.26E + 22	6.40E + 22
Bottom of Core		
Baffle-to-Former Bolts	6.85E + 21	1.03E + 22
Baffle-to-Baffle bolts	3.62E + 21	5.43E + 21

Table 3-2 Applicable Reactor Vessel Internals Aging Effects

Affected Parts	Aging Effect
PLENUM ASSEMBLY	
Plenum Cover Assembly	None
Plenum Cylinder	None
Upper Grid Assembly	
Upper Grid Assembly	Cracking of base metal and welds (IASCC)
	Reduction of Fracture Toughness of base metal and welds (Irradiation Embrittlement)
Fuel Assembly Support Pads	Loss of Material (Wear)
Plenum Rib Pads	Loss of Material (Wear)
Control Rod Guide Tube Assembly	
CRGT Flange to Upper Grid Screws	Loss of Closure Integrity (Stress Relaxation)
CRGT Assembly Spacer Castings	Reduction of Fracture Toughness (Thermal Embrittlement)
CORE SUPPORT SHIELD ASSEMBLY	
Core Support Shield Cylinder Top Flange	Loss of Material (Wear)
Core Support Shield to Core Barrel Bolts	Cracking (IASCC & SCC)
	Reduction of Fracture Toughness (Irradiation Embrittlement)
	Loss of Closure Integrity (Stress Relaxation)
ONS-3 CSS Outlet Nozzles	Reduction of Fracture Toughness (Thermal Embrittlement)
Vent Valve Assembly Bodies	Reduction of Fracture Toughness (Thermal Embrittlement)
Vent Valve Assembly Retaining Rings	Reduction of Fracture Toughness (Thermal Embrittlement)
Vent Valve Assy Modified Locking Devices	Cracking (SCC)
Vent Valve Assy Original Locking Devices	Loss of Material (Wear)

Table 3-2 Applicable Reactor Vessel Internals Aging Effects (continued)

Affected Parts	Aging Effect
CORE BARREL ASSEMBLY	
Core Barrel Assembly	Cracking of base metal and welds (IASCC)
	Reduction of Fracture Toughness of base metal and welds (Irradiation Embrittlement)
	Change in Dimension (Void Swelling)
Baffle-to-Baffle Bolts	Cracking (IASCC)
	Reduction of Fracture Toughness (Irradiation Embrittlement)
	Loss of Closure Integrity (Stress Relaxation)
	Change in Dimension (Void Swelling)
Baffle-to-Former Bolts	Cracking (IASCC)
	Reduction of Fracture Toughness (Irradiation Embrittlement)
	Loss of Closure Integrity (Stress Relaxation)
	Change in Dimension (Void Swelling)
Lower Internals Assy to Core Barrel Bolts	Cracking (IASCC & SCC)
	Reduction of Fracture Toughness (Irradiation Embrittlement)
	Loss of Closure Integrity (Stress Relaxation)
Core Barrel to Thermal Shield Bolts	Cracking (SCC)
	Loss of Closure Integrity (Stress Relaxation)

Table 3-2 Applicable Reactor Vessel Internals Aging Effects (continued)

Affected Parts	Aging Effect
LOWER INTERNALS ASSEMBLY	
Lower Grid Assembly	Cracking of base metal and welds (IASCC)
	Reduction of Fracture Toughness of base metal and welds (Irradiation Embrittlement)
Lower Internals Assy to Thermal Shield Bolts	Cracking (SCC)
	Loss of Closure Integrity (Stress Relaxation)
Fuel Assembly Support Pads	Loss of Material (Wear)
Guide Blocks	Loss of Material (Wear)
Shell Forging to Flow Distributor Bolts	Cracking (SCC)
	Loss of Closure Integrity (Stress Relaxation)
Lower Grid Rib to Shell Forging Screws	Loss of Closure Integrity (Stress Relaxation)
Incore Guide Tube Spider Castings	Reduction of Fracture Toughness (Thermal Embrittlement)

4. DEMONSTRATION THAT EFFECTS OF AGING ARE MANAGED

The aging management review is performed by demonstrating that the applicable aging effects, as identified in Section 3.0, can be managed by existing programs when continued into the period of extended operation. No further action is required for license renewal when the evaluation of an existing program demonstrates that it is adequate to manage the applicable aging effect such that corrective action may be taken prior to loss of the structural integrity of the reactor vessel internals such that their component functions are maintained.

Demonstration for the purposes of this license renewal technical evaluation is accomplished by establishing a clear relationship among:

1) The components under review,

2) The aging effects on these items caused by the material-environment-stress combinations which, if undetected, could result in a condition in which the reactor vessel internals could not perform its intended function(s) in the period of extended operation, and

3) The credited aging management programs whose actions serve to preserve the reactor vessel internals intended function(s) for the period of extended operation.

The components within the scope of the reactor vessel internals report were described in Section 2.0 and the aging effects that apply to those components were summarized in Section 3.0. The purpose of this section is to describe the existing programs that are credited for managing the applicable aging effects and to provide justification as to why the credited technical elements adequately manage aging for the period of extended operation.

The aging management programs primarily credited within this section fall under ASME B&PV Code, Section XI, Plant Technical Specifications and commitments to generic NRC communications. Some general background on these program groups credited for aging management is provided in the following sections.

ASME B&PV Code Section XI ISI
The regulatory basis for providing an inservice inspection/testing program is found in 10 CFR 50.55a(g), which specifically requires inservice inspection (ISI) and inservice testing (IST) per ASME B&PV Code, Section XI, and 10 CFR 50.36(c)(3), which provides general surveillance requirements. In addition, plant Technical Specifications specifically require both ISI and IST.

Plant-specific Technical Specifications may specify which edition of Section XI of the Code will be effective for the initial inspection period. As required by 10 CFR 50.55a, every 120 months the Inservice Inspection (ISI) Plan is reviewed and revised to meet the latest NRC-authorized edition of the ASME B&PV Code. This revision is submitted to the NRC for approval. At present, the approved references to Section XI in 10 CFR 50.55a include addenda through the 1988 Addenda and editions through the 1989 Edition. Mandatory Appendix VII (Qualification of Nondestructive Examination Personnel for Ultrasonic Examination) is required when referencing the 1989 Edition; however, mandatory Appendix VIII (Performance Demonstration for Ultrasonic Examination Systems) was first introduced in the 1989 Addenda.

The ASME B&PV Code, Section XI, requirements for inservice inspection of the reactor vessel internals items within scope of this report are shown in Table IWB 2500-1 of the 1989 [31] edition of ASME Section XI, including mandatory Appendices VII and VIII--Appendix VIII in accordance with 1989 Addenda. The applicable aging effects discussed in Chapter 3.0 are dispositioned in accordance with the requirements and acceptance standards for the Examination Categories specified in the 1989 Edition of ASME Section XI (including Appendices VII and VIII). It is the responsibility of the licensee to ensure that the examination requirements and acceptance standards associated with the Examination meet the requirements and acceptance standards of the 1989 Edition, including Appendices VII and VIII (1989 Addenda), of ASME B&PV Code, Section XI, when crediting this report for license renewal.

Items within the scope of this report may be installed in areas inaccessible for maintenance and inspection. Though a conscious effort was made during design and construction to make items requiring inservice inspection accessible, competing design requirements meant it was not always achievable to do so. For those limited instances where an inaccessible item requiring inspection does exist, ASME B&PV Code, Section XI, has provisions to handle such a situation. Under the philosophy of ASME B&PV Code, Section XI, programmatic oversight does not imply 100% direct coverage of all items within a system. Rather, provisions are made under the guidance of the code to programmatically deal with those inaccessible locations that will require indirect assurance of integrity. Indirect assurance comes in the form of approved code relief, use of statistical sampling methods, and use of indirect symptomatic evidence.

The only inspection of the reactor vessel internals within the scope of this report required by ASME Section XI is in Examination Category B-N-3, Item number B13.70 which requires a complete VT-3 visual examination of all accessible surfaces of the core support structure, which must be removed from the reactor vessel for examination. This examination is typically performed in the refueling transfer canal using remote video devices. Visual examination indications associated with the ISI inspection must be corrected if any of the relevant conditions of IWB-3520.2 are observed, as listed below:

a) structural distortion or displacement of parts to the extent that component function may be impaired;

b) loose, missing, cracked, or fractured parts, bolting, or fasteners;

c) foreign materials or accumulation of corrosion products that could interfere with control rod motion or could result in blockage of coolant flow through fuel;

d) corrosion or erosion that reduces the nominal section thickness by more than 5%;

e) wear of mating surfaces that may lead to loss of function; or

f) structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5%.

These relevant conditions must be corrected in accordance with IWB-3142 prior to continued service, with the following options available: (1) acceptance by supplemental examinations, such as additional surface or volumetric examination intended to further characterize the indication; (2) acceptance by corrective measures or repairs, such as repair welding; (3) acceptance by analytical evaluation, which may involve more frequent inspections of the item; and (4) acceptance by replacement of the item.

Technical Specifications
The aging management elements contained in the plant Technical Specifications include primary chemistry requirements which, as discussed in Section 3.0, are required to maintain the design basis of the components. (The primary chemistry requirements are found in the plant FSAR for those plants using the Standard Technical Specifications.) The Technical Specifications, or FSAR for plants using Standard Technical Specifications, require that all plants maintain chloride, fluoride, and oxygen concentration in the RCS within prescribed limits so that the RCS is protected against SCC. If chloride, fluoride, or oxygen limits are exceeded, corrective measures are required to reduce concentrations below the allowable limits. The reactor coolant chemistry requirements are required to ensure that the design basis of the component is maintained. This aging management measure provides a defense-in-depth strategy against aging effects that could lead to loss of the RCS intended functions supported by the reactor vessel internals.

Vent Valve Inspections
The aging management elements of the reactor internals vent valves are contained in plant technical specifications for ANO-1 and TMI-1 and contained in the Pump and Valve In-Service Test Program per Section XI for ONS 1, 2, and 3. Both the technical specifications and the In-Service Test Program require vent valve testing and inspections be performed each refueling outage. The aging management measure provided in these requirements includes a provision primarily to visually inspect the valve body and disc seating surfaces. However, the entire vent valve assembly is typically inspected including the locking devices. In both cases, any observed surface irregularities would be evaluated. In addition, vent valve operation is tested through manual actuation that the vent valve will be fully open for a force less than or equal to 400 lbs.

Commitments to NRC Generic Communications
As discussed in Section 3.0, the GLRP delineates between those commitments required to validate the design of components and commitments required to manage aging. Both may include technical elements that will be carried forward to the period of extended operation. No

commitments associated with design validation that will be carried forward to the period of extended operation were identified in Section 3.6, nor are there any such programs implemented as a result of commitments to NRC generic communications credited in Chapter 4.0 for managing applicable aging effects defined in Section 3.0.

The following sections discuss the aging management programs for the following aging effects: Cracking, Reduction of Fracture Toughness, Loss of Material, and Stress Relaxation. The NRC, in its Final Safety Evaluation noted that change in dimension (void swelling) must also be addressed. The GLRP in its response to the NRC Draft Safety Evaluation noted that the components of the core barrel assembly (i.e., baffle bolts, former plates, and baffle plates) are considered susceptible items. As such, the GLRP committed to continue to participate and follow industry activities and evaluate research data regarding void swelling. This commitment will be part of the RV Internals Aging Management Program discussed in Section 4.6.

4.1 Demonstration of Aging Management for Cracking

As discussed in Section 3.1, the reactor vessel internals items that are subjected to cracking that require programmatic management are as follows:

1) Cracking of the core barrel assembly base metal and welds due to IASCC.

2) Cracking of the baffle-to-baffle bolts due to IASCC.

3) Cracking of the baffle-to-former bolts due to IASCC.

4) Cracking of the upper grid assembly base metal and welds due to IASCC.

5) Cracking of the lower grid assembly base metal and welds due to IASCC.

6) Cracking of the core support shield to core barrel bolts due to IASCC and SCC.

7) Cracking of the lower internals assembly to core barrel bolts due to IASCC and SCC.

8) Cracking of the core barrel to thermal shield bolts due to SCC.

9) Cracking of the lower internal assembly to thermal shield bolts due to SCC.

10) Cracking of the shell forging to flow distributor bolts due to SCC.

11) Cracking of the Alloy 600 locking devices on the modified vent valve assembly due to SCC.

Examination Category B-N-3 of the ASME Section XI ISI program, Subsection IWB, provides requirements for the visual inspection (VT-3) for removable core support structures. These requirements define conditions in IWB-3520.2 which, if detected, must be corrected prior to continued service. One of these conditions include "loose, missing, cracked, or fractured parts, bolting, or fasteners". IWB-3142 provides options for correcting the relevant condition, such as (1) acceptance by supplemental surface and/or volumetric examination, in order to characterize the indication more accurately; (2) acceptance by analytical evaluation, which may involve more frequent examination of the item; or (3) acceptance by replacement of the item. Examination Category B-N-3 is a viable program to detect cracking of the locking devices on the modified vent valve assembly since this part is readily accessible. However, it is recognized that Examination Category B-N-3 may not be adequate to detect cracking for the remainder of the above component items because of accessibility concerns. As such, the GLRP has implemented a program to manage the effects of aging due to cracking of the reactor vessel internals. This aging management program is discussed in Section 4.6.

Cracking of the Alloy 600 locking devices on the modified vent valve assembly due to PWSCC is managed by Examination Category B-N-3. This program will be supplemented by technical specification requirements for ANO-1 and TMI-1 and the Pump and Valve In-Service Test Program for Oconee which require vent valve testing and a visual inspection of the vent valve each refueling outage.

The continuation of ASME B&PV Code Section XI programmatic controls along with the vent valve testing and inspection requirements and the Reactor Vessel Internals Aging Management Program will manage cracking of the reactor vessel internals items that could cause loss of the reactor vessel internals function(s) in the period of extended operation. Specific component items that are subjected to cracking and the program listings to manage this aging effect are listed in Table 4-1 on a component item basis.

4.2 Demonstration of Aging Management for the Reduction of Fracture Toughness

As discussed in Section 3.2, the reactor vessel internals items that are subjected to reduction of fracture toughness that require programmatic management are as follows:

1) Reduction of fracture toughness of the ONS-3 outlet nozzles due to thermal embrittlement.

2) Reduction of fracture toughness of the vent valve bodies and retaining rings due to thermal embrittlement.

3) Reduction of fracture toughness of the core barrel assembly base metal and welds due to irradiation embrittlement.

4) Reduction of fracture toughness of the baffle-to-baffle bolts due to irradiation embrittlement.

5) Reduction of fracture toughness of the baffle-to-former bolts due to irradiation embrittlement.

6) Reduction of fracture toughness of the core support shield to core barrel bolts due to irradiation embrittlement.

7) Reduction of fracture toughness of the lower internals assembly to core barrel bolts due to irradiation embrittlement.

8) Reduction of fracture toughness of the upper grid assembly base metal and welds due to irradiation embrittlement.

9) Reduction of fracture toughness of the lower grid assembly base metal and welds due to irradiation embrittlement.

10) Reduction of fracture toughness of the CRGT assembly spacer castings due to thermal embrittlement.

11) Reduction of fracture toughness of the incore guide tube spider castings due to thermal embrittlement.

Examination Category B-N-3 of the ASME Section XI ISI program, Subsection IWB, provides requirements for the visual inspection (VT-3) for removable core support structures. These requirements define conditions in IWB-3520.2 which, if detected, must be corrected prior to continued service. One of these conditions include "loose, missing, cracked, or fractured parts, bolting, or fasteners". IWB-3142 provides options for correcting the relevant condition, such as (1) acceptance by supplemental surface and/or volumetric examination, in order to characterize the indication more accurately; (2) acceptance by analytical evaluation, which may involve more frequent examination of the item; or (3) acceptance by replacement of the item. However, for aging management of component items susceptible to irradiation embrittlement, it is recognized that Examination Category B-N-3 may not be adequate to detect the reduction of fracture toughness in components. As such, the GLRP has implemented a program to manage the effects of aging due to the reduction of fracture toughness of the reactor vessel internals. This aging management program is discussed in Section 4.6.

For the aging management of the ONS-3 cast austenitic stainless steel CSS outlet nozzles due to thermal embrittlement, Examination Category B-N-3 will be supplemented by evaluation procedures for potential flaws specified in IWB-3640. These procedures formally apply to austenitic piping and fittings, however they also may be applied to CASS components as discussed in Section 4.2 of BAW-2243A [30] and EPRI-TR-106092 [35]. Reduction of fracture toughness of the vent valve bodies and retaining rings due to thermal embrittlement are managed by Examination Category B-N-3. Further, these components are managed by technical specification requirements for ANO-1 and TMI-1 and the Pump and Valve In-Service Test Program for the Oconee units, which require vent valve testing and a visual inspection of the internal vent valve components. However, for the vent valve bodies, these programs will be further supplemented by the evaluation procedures for potential flaws in cast austenitic stainless steels as mentioned above.

In response to the NRC draft safety evaluation regarding the aging management of CASS items, the GLRP agreed that a limited augmented inspection is appropriate to manage the aging effects of thermal embrittlement and its potential synergistic effect with neutron fluence for the period of extended operation. The details of the inspections (sample size, examination method, and acceptance criteria) would be determined by the RV Internals Aging Management Program which is discussed in Section 4.6.

The continuation of ASME B&PV Code Section XI programmatic controls along with the vent valve testing and inspection requirements and the Reactor Vessel Internals Aging Management Program will manage reduction of fracture toughness of the reactor vessel internals items that could cause loss of the reactor vessel internals function(s) in the period of extended operation. Specific component items that are subjected to reduction of fracture toughness and the program listings to manage this aging effect are listed in Table 4-1 on a component item basis.

4.3 Demonstration of Aging Management for the Loss of Material

As discussed in Section 3.3, the reactor vessel internals items that are subjected to loss of material that require programmatic management are as follows:

1) Loss of material of the plenum rib pads due to wear.

2) Loss of material of the fuel assembly support pads on the upper grid assembly due to wear.

3) Loss of material of the top flange on the core support shield cylinder due to wear.

4) Loss of material of the fuel assembly support pads on the lower grid assembly due to wear.

5) Loss of material of the guide blocks due to wear.

6) Loss of material of the locking devices on the original vent valve assemblies due to wear.

Loss of material of the above component items are managed by Examination Category B-N-3 of the ASME Section XI ISI program, Subsection IWB, which provides requirements for the visual inspection (VT-3) for removable core support structures. These requirements define conditions in IWB-3520.2 which, if detected, must be corrected prior to continued service. These conditions include "wear of mating surfaces that may lead to loss of function and corrosion or erosion that reduces the nominal section by more than 5%". IWB-3142 provides options for correcting the relevant condition, such as acceptance by corrective measures or repairs.

Loss of material of the locking devices on the original vent valve assemblies due to wear is managed by Examination Category B-N-3. This program will be supplemented by technical specification requirements for ANO-1 and TMI-1 and the Pump and Valve In-Service Test Program for Oconee which require vent valve testing and a visual inspection of the vent valve each refueling outage.

The continuation of ASME B&PV Code Section XI programmatic controls along with the vent valve testing and inspection requirements will manage loss of material of the reactor vessel internals items that could cause loss of the reactor vessel internals function(s) in the period of extended operation. Specific component items that are subjected to loss of material and the program listings to manage this aging effect are listed in Table 4-1 on a component item basis.

4.4 Demonstration of Aging Management for the Loss of Mechanical Closure Integrity

As discussed in Section 3.4, the reactor vessel internals items that are subjected to loss of mechanical closure integrity that require programmatic management are as follows:

1) Loss of closure integrity of the CRGT flange to upper grid screws due to stress relaxation.

2) Loss of closure integrity of the core support shield to core barrel bolts due to stress relaxation.

3) Loss of closure integrity of the core barrel to thermal shield bolts due to stress relaxation.

4) Loss of closure integrity of the lower internals assembly to core barrel bolts due to stress relaxation.

5) Loss of closure integrity of the lower grid rib-to-shell forging screws due to stress relaxation.

6) Loss of closure integrity of the shell forging to flow distributor bolts due to stress relaxation.

7) Loss of closure integrity of the lower internals assembly to thermal shield bolts due to stress relaxation.

8) Loss of closure integrity of the baffle and former bolts due to stress relaxation.

The VT-3 visual examination required by Examination Category B-N-3 of the ASME Section XI ISI Program, Subsection IWB, is specifically designed to determine the general mechanical and structural condition, including "structural distortion and displacements, loose or missing parts, wear of mating surfaces that may lead to the loss of integrity at bolted or welded connections. IWB-3142 provides options for correcting the relevant condition, such as (1) acceptance by supplemental surface and/or volumetric examination, in order to further characterize the condition; (2) acceptance by corrective measures (i.e. re-establishing the preload) or repairs; or (3) acceptance by replacement of the item. However it is recognized that Examination Category B-N-3 may not be adequate to detect the loss of mechanical closure integrity in components. As such, the GLRP has implemented a program to manage the effects of aging due to the loss of mechanical closure integrity of the reactor vessel internals. This aging management program is discussed in Section 4.6.

The continuation of ASME B&PV Code Section XI programmatic controls along with the Reactor Vessel Internals Aging Management Program will manage loss of mechanical closure integrity of the reactor vessel internals items that could cause loss of the reactor vessel internals function(s) in the period of extended operation. Specific component items that are subjected to loss of mechanical closure integrity and the program listings to manage this aging effect are listed in Table 4-1 on a component item basis.

4.5 Time Limited Aging Analyses Applicable to the Reactor Vessel Internals

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

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

2) consider the effects of aging;

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

4) were determined to be relevant by the licensee in making a safety determination;

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

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

TLAA applicable to the reactor vessel internals items within the scope of this report were identified by reviewing the following documentation: (1) plant-specific docketed correspondence files, plant-specific FSARs, and BAW topical reports, and (2) ASME B&PV Code Section XI inservice inspection requirements.

Applicable TLAA include: (1) FIV endurance limit assumptions as reported in BAW-10051 [2], (2) transient cycle count assumptions used implicitly in the design of the reactor vessel internals and used in the fatigue cumulative usage factor calculations for the replacement bolting items as reported in BAW-1843PA [12] and BAW-1789P [33], (3) the effect of neutron irradiation on reactor vessel internals material properties and deformation limits as reported in BAW-10008, Part 1, Rev. 1 [24], and (4) flaw growth acceptance in accordance with the ASME B&PV Code, Section XI, inservice inspection requirements. Flaw growth acceptance in accordance with Section XI is plant-specific and is not addressed in this report; all remaining TLAA are evaluated in this report.

Options for Evaluation
The analyses and calculations identified above as TLAA for the reactor vessel internals scope may be grouped under cracking (initiation and growth) and reduction of fracture toughness. For these identified TLAA it must be demonstrated that

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

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

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

4.5.1 Fatigue - Cracking (Initiation and Growth)

FIV Endurance Limit Assumptions
BAW-10051 calculated stress values for the redesigned reactor vessel internals and compared them to endurance limit (stress) values. These endurance limit values were based on an assumed value of 10^{12} cycles for 40 years of operation. Since the fatigue curves at the time of design only went up to 10^6 cycles, these curves were extrapolated to 10^{12} cycles. The methodology used in BAW-10051 was extended from 40 years to 60 years by multiplying the assumed endurance limit cycles by 1.5 and then conservatively using 10^{13} cycles to determine the endurance limit based on more recent ASME fatigue curves which extend now to 10^{11} cycles. The component item stress values in BAW-10051 were compared to the recalculated endurance limit values and shown acceptable. Hence, this TLAA is resolved per 10 CFR 54.21(c)(1)(ii).

Transient Cycle Count Assumptions
As noted in Section 2, the reactor vessel internals were designed and constructed prior to the development of ASME Code requirements for core support structures. As such, existing industry structural practice was used in the design of the internals structural members and the only specific fatigue analyses performed in the original design were those that addressed high cycle fatigue as noted earlier in this section. The internals designers did, however, consider the reactor coolant system functional design requirements when performing their structural design. Meeting these requirements in the original design meant that the reactor vessel internals were implicitly designed for low cycle fatigue based on the projected reactor coolant system design transients. In modifications following original design, plant-specific fatigue analyses were performed for the reactor vessel internals replacement bolts as presented in BAW-1843PA [12] and BAW-1789P [33]. These topical reports summarize fatigue analyses performed to the ASME code (Section III, Subsection NG) including both high cycle fatigue (FIV) and low cycle fatigue (design transients).

Design cyclic loadings and thermal conditions for the B&W-designed reactor coolant system Class 1 components are defined by the component design specifications. These design cyclic loadings (e.g., 240 heatups and cooldowns) were used implicitly in the original design of the reactor vessel internals and, later, explicitly to calculate the ability of the replacement bolting items to withstand cyclic operation without fatigue failure. The ability to withstand cyclic operation without fatigue failure is expressed in terms of calculations required by Section III, i.e., cumulative usage factors.

In accordance with Section 4.1.2.6 of the ANO-1 and TMI-1 FSARs, and Section 5.1.2.6 of the ONS FSAR, design cyclic loadings for the B&W-designed reactor coolant system Class 1 components are based on a service lifetime of 40 years. These cyclic loadings were used to generate cumulative usage factors for the B&W-designed Class 1 components. It has been demonstrated that the existing cumulative usage factors for the reactor vessel internals replacement bolting items within the scope of this report remain valid for the period of extended operation. Matrices detailing the fatigue usage factors of the reactor vessel internals replacement bolting items and the design transients used to calculate their fatigue usage factors (transient type and number of design cycles) were prepared using component design basis documentation. Design cyclic loadings are contained in each plant's FSAR or Technical Specifications. Each of the participating utilities monitors occurrences of design transients and are thereby managing the potential for cracking (initiation and growth) resulting from fatigue.

Based on the fatigue cumulative usage factor-design transient matrices, the controlling transients used to calculate fatigue cumulative usage factors for the reactor vessel internals replacement bolting items within the scope of this report have been identified [28]. For this set of controlling cyclic design transients, the number of transients accrued to date for each participating utility was compiled and a conservative projection was made to determine if the number of design cycles may be exceeded in the period of extended operation. In no instance was the extrapolation shown to exceed the number of design cycles prior to 60 years of operation. It is thereby demonstrated that the existing usage factor calculations due to the design transients remain valid for the period of extended operation in accordance with 10 CFR 54.21 (c)(1)(i). Plants must continue to monitor and track occurrences of design transients.

The total usage factor was recalculated for the replacement bolting items by considering both the existing design transient usage factors shown acceptable for 60 years, and by the recalculation of fatigue usage factors due to flow-induced vibration. The FIV calculated methodology presented in BAW-1843PA and BAW-1789P for the replacement bolting items was extended from 40 years to 60 years. Hence, the FIV usage factors are resolved per 10 CFR 54.21(c)(1)(ii). Therefore, the total usage factor for the replacement bolting items are shown acceptable for the period of extended operation in accordance with a combination of 10 CFR 54.21(c)(1)(i) and (ii).

Since the controlling design transients applicable for 40 years of operation remain valid for 60 years of operation for the reactor vessel and the replacement bolts in accordance with Reference 28, it follows that the remainder of the reactor vessel internals, implicitly designed for 40 years of operation, are acceptable for 60 years of operation given no changes in the boundary conditions from the original design. The only possible boundary condition change is the change in the material ductility discussed in Section 4.5.2.

4.5.2 Ductility - Reduction of Fracture Toughness

BAW-10008, Part 1, Rev. 1, documents the acceptability of the reactor vessel internals under LOCA and a combination of LOCA and seismic loadings. The effect of irradiation on the material properties and deformation limits for the internals is presented in Appendix E where it is concluded that at the end of 40 years, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits.

Existing literature was reviewed with regard to irradiation of the reactor vessel internals. Literature reviewed were: (1) BAW-2060, "Project Topical Report: Oconee Nuclear Power Station Unit 1 Reactor Internals Life Extension Project"; (2) EPRI TR-103838, "PWR Reactor

Pressure Vessel Internals License Renewal Industry Report; Revision 1"; and (3) Sixth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors. Two conclusions were drawn from the review:

1) Additional testing of the physical and mechanical property changes of irradiated material and continued surveillance of the reactor internals should be performed to provide reliable data on the irradiated properties of stainless steel.

2) ASME Section XI visual (VT-3) examination standards for category B-N-3 is a current and effective program for detection of cracks and repair/replacement for vessel internals components that are accessible by removal of the core and/or other internals.

However as noted in Section 4.2, Examination Category B-N-3 may not be adequate to detect reduction of fracture toughness in components. A program is being implemented to manage the effects of aging due to the reduction of fracture toughness of the reactor vessel internals. This aging management program is discussed in Section 4.6. Hence, this TLAA will be resolved on a plant-specific basis per 10 CFR 54.21 (c)(1)(iii) based on the results and conclusion of the program.

4.6 Reactor Vessel Internals Aging Management Program

Overview

A comprehensive aging management program will be developed to supplement the ASME Section XI ISI program for those aging effects on the reactor vessel internal component items identified in Sections 4.1, 4.2, and 4.4. Examination Category B-N-3 of the ASME Section XI program, Subsection IWB, provides for a visual VT-3 inspection of removable core support structures. This inspection program may not be adequate to detect aging effects for certain reactor vessel internals component items since their locations are not easily accessible using current technology. As such, this aging management program will be implemented to manage the aging effects for those items identified in Table 4-1 such that the reactor vessel internals can perform their component intended function(s) for the period of extended operation associated with license renewal.

Purpose

The purpose of the Reactor Vessel Internals Aging Management Program is to:

1) Continue the investigation of the potential aging effects that have been identified in this report for the reactor vessel internals; and

2) Establish appropriate monitoring and inspection programs that will continue to maintain the reactor vessel internals functional through the licensed life of the unit.

The program will be subject to periodic re-evaluation such that as new information becomes available, the program elements can change to allow the program to grow and develop in order to meet the purpose of the program.

Summary of Aging Effects
Table 4-1 indicates that for certain reactor vessel internals component items - aging effect combinations, the Section XI ISI program will be supplemented by a Reactor Vessel Internals Aging Management Program. These component items and their associated aging effects are summarized below for each reactor internals assembly.

Applicable aging effects for the plenum assembly include: (1) cracking due to IASCC and reduction of fracture toughness due to irradiation embrittlement of the upper grid assembly base metal and welds; (2) loss of closure integrity of the CRGT flange to upper grid screws due to stress relaxation; and (3) reduction of fracture toughness due to thermal embrittlement and potential synergistic effects with neutron fluence of the CRGT spacer castings.

Applicable aging effects for the core support assembly include: (1) cracking due to IASCC and SCC, reduction of fracture toughness due to irradiation embrittlement, and loss of closure integrity due to stress relaxation of the core support shield to core barrel bolts; and (2) reduction of fracture toughness due to thermal embrittlement and potential synergistic effects with neutron fluence of the vent valve bodies, vent valve retaining rings, and ONS-3 outlet nozzles.

Applicable aging effects for the core barrel assembly include: (1) cracking due to IASCC and reduction of fracture toughness due to irradiation embrittlement of the core barrel assembly base metal and welds; (2) cracking due to IASCC, reduction of fracture toughness due to irradiation embrittlement, and loss of closure integrity due to stress relaxation of the baffle-to-baffle and the baffle-to-former bolts; (3) cracking due to IASCC and SCC, reduction of fracture toughness due to irradiation embrittlement, and loss of closure integrity due to stress relaxation of the lower internals assembly to core barrel bolts; (4) cracking due to SCC and loss of closure integrity due to stress relaxation of the core barrel to thermal shield bolts; and (5) change in dimension due to void swelling of the baffle and former plates, baffle-to-baffle bolts, and baffle-to-former bolts.

Applicable aging effects for the lower internals assembly include: (1) cracking due to IASCC and reduction of fracture toughness due to irradiation embrittlement of the lower grid assembly base metal and welds; (2) cracking due to SCC and loss of closure integrity due to stress relaxation of the lower internals assembly to thermal shield bolts; (3) cracking due to SCC and loss of closure integrity due to stress relaxation of the shell forging to flow distributor bolts; (4) loss of closure integrity due to stress relaxation of the lower grid rib to shell forging screws; and (5) reduction of fracture toughness due to thermal embrittlement and potential synergistic effects with neutron fluence of the incore guide tube spider castings.

Reactor Vessel Internals Aging Management Program
This program represents a pro-active position to address potential aging degradation of the reactor vessel internals component items. The program will be divided to address each of the specific aging mechanisms as noted above. Each of these divisions will consist of various

elements to manage the aging mechanisms for the license renewal period. Elements of the program to address each aging mechanism are noted below:

- To address IASCC, the program elements are: (1) to perform fluence mapping in order to determine the component items with the highest susceptibility; (2) to evaluate the effects of bolt failure; and (3) develop appropriate inspection techniques for the high susceptible locations.

- To address SCC, the program elements are: (1) to incorporate the on-going activities of the Internals Bolting Surveillance Program. This program is designed to evaluate the effects of applied stresses, fabrication processes, and the PWR environment on various bolting materials that are used in the reactor vessel internals; and (2) establish appropriate monitoring and inspection techniques.

- To address thermal embrittlement and its potential synergistic effect with neutron fluence, the GLRP in response to the NRC draft safety evaluation agreed that a limited augmented inspection is appropriate. The program elements are to determine sample size, examination method, acceptance criteria, and the timing of the inspection.

- To address irradiation embrittlement, the program elements are: (1) to develop a better understanding of the material property changes; (2) to determine the component items with the highest susceptibility; and (3) establish appropriate test techniques. [Note: The GLRP in response to the NRC draft safety evaluation agreed that an augmented inspection for the accessible regions of the baffle plates is appropriate to manage cracking and reduction of fracture toughness. Details of the inspection such as sample size using risk-based methods, examination method, and acceptance criteria will be developed in the program.]

- To address stress relaxation, the program elements are: (1) to determine critical locations; and (2) establish appropriate monitoring and inspection techniques.

- To address void swelling, the GLRP in its response to the NRC draft safety evaluation committed to continue to participate in and follow industry activities and evaluate research data to determine whether change in dimension by void swelling could impact the intended function of the RV internals.

Execution of the Reactor Vessel Internals Aging Management Program will ensure that the reactor vessel internals component intended functions are maintained for the period of extended operation. The commitment to implement the aging management program and to notify the NRC staff regarding the status of the program activities on a regular basis will be part of the FSAR supplement (attached in Appendix A) and be included in the license renewal application.

Table 4-1 Programs That Manage Applicable Reactor Vessel Internals Aging Effects

Affected Parts	Aging Effect	Programs That Manage Applicable Aging Effects
PLENUM ASSEMBLY		
Upper Grid Assembly		
Upper Grid Assembly	Cracking of base metal and welds	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Reduction of Fracture Toughness of base metal and welds	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
Fuel Assembly Support Pads	Loss of Material	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval.
Plenum Rib Pads	Loss of Material	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval.
Control Rod Guide Tube Assembly		
CRGT Flange to Upper Grid Screws	Loss of Closure Integrity	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
CRGT Assembly Spacer Castings	Reduction of Fracture Toughness	RVI Aging Management Program See Section 4.6.

Table 4-1 Programs That Manage Applicable Reactor Vessel Internals Aging Effects (continued)

Affected Parts	Aging Effect	Programs That Manage Applicable Aging Effects
CORE SUPPORT SHIELD ASSEMBLY		
Core Support Shield Cylinder Top Flange	Loss of Material	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval.
Core Support Shield to Core Barrel Bolts	Cracking	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Reduction of Fracture Toughness	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Loss of Closure Integrity	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
ONS-3 CSS Outlet Nozzles	Reduction of Fracture Toughness	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval Supplemented by extension of the evaluation procedures in IWB-3640 to CASS items as specified in Section 4.2 of BAW-2243A [30]. RVI Aging Management Program See Section 4.6.

Table 4-1 Programs That Manage Applicable Reactor Vessel Internals Aging Effects (continued)

Affected Parts	Aging Effect	Programs That Manage Applicable Aging Effects
CORE SUPPORT SHIELD ASSEMBLY (continued)		
Vent Valve Bodies	Reduction of Fracture Toughness	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval. Supplemented by: • Plant Technical Specifications (ANO-1, TMI-1) *Vent valve testing and inspection requirements each refueling outage* • Pump and Valve In-Service Test Program (ONS-1,-2,-3) *Vent valve testing and inspection requirements each refueling outage* Further Supplemented by extension of the evaluation procedures in IWB-3640 to CASS items as specified in Section 4.2 of BAW-2243A [30]. RVI Aging Management Program See Section 4.6.
Vent Valve Retaining Rings	Reduction of Fracture Toughness	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval. Supplemented by: • Plant Technical Specifications (ANO-1, TMI-1) *Vent valve testing and inspection requirements each refueling outage* • Pump and Valve In-Service Test Program (ONS-1,-2,-3) *Vent valve testing and inspection requirements each refueling outage* RVI Aging Management Program See Section 4.6.

Table 4-1 Programs That Manage Applicable Reactor Vessel Internals Aging Effects (continued)

Affected Parts	Aging Effect	Programs That Manage Applicable Aging Effects
CORE SUPPORT SHIELD ASSEMBLY (continued)		
Vent Valve Locking Devices (Modified)	Cracking	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval. Supplemented by: • Plant Technical Specifications (ANO-1, TMI-1) *Vent valve testing and inspection requirements each refueling outage* • Pump and Valve In-Service Test Program (ONS-1,-2,-3) *Vent valve testing and inspection requirements each refueling outage*
Vent Valve Locking Devices (Original)	Loss of Material	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval. Supplemented by: • Plant Technical Specifications (ANO-1, TMI-1) *Vent valve testing and inspection requirements each refueling outage* • Pump and Valve In-Service Test Program (ONS-1,-2,-3) *Vent valve testing and inspection requirements each refueling outage*

Table 4-1 Programs That Manage Applicable Reactor Vessel Internals Aging Effects (continued)

Affected Parts	Aging Effect	Programs That Manage Applicable Aging Effects
CORE BARREL ASSEMBLY		
Core Barrel Assembly	Cracking of base metal and welds	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Reduction of Fracture Toughness of base metal and welds	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Dimensional Change	RVI Aging Management Program See Section 4.6.
Baffle-to-Baffle Bolts	Cracking	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Reduction of Fracture Toughness	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Loss of Closure Integrity	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Dimensional Change	RVI Aging Management Program See Section 4.6.

Table 4-1 Programs That Manage Applicable Reactor Vessel Internals Aging Effects (continued)

Affected Parts	Aging Effect	Programs That Manage Applicable Aging Effects
CORE BARREL ASSEMBLY (continued)		
Baffle-to-Former Bolts	Cracking	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Reduction of Fracture Toughness	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Loss of Closure Integrity	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Dimensional Change	RVI Aging Management Program See Section 4.6.

Table 4-1 Programs That Manage Applicable Reactor Vessel Internals Aging Effects (continued)

Affected Parts	Aging Effect	Programs That Manage Applicable Aging Effects
CORE BARREL ASSEMBLY (continued)		
Lower Internals Assembly to Core Barrel Bolts	Cracking	<u>ASME Section XI ISI Programs</u> *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval <u>RVI Aging Management Program</u> See Section 4.6.
	Reduction of Fracture Toughness	<u>ASME Section XI ISI Programs</u> *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval <u>RVI Aging Management Program</u> See Section 4.6.
	Loss of Closure Integrity	<u>ASME Section XI ISI Programs</u> *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval <u>RVI Aging Management Program</u> See Section 4.6.

Table 4-1 Programs That Manage Applicable Reactor Vessel Internals Aging Effects (continued)

Affected Parts	Aging Effect	Programs That Manage Applicable Aging Effects
CORE BARREL ASSEMBLY (continued)		
Core Barrel to Thermal Shield Bolts	Cracking	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Loss of Closure Integrity	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
LOWER INTERNALS ASSEMBLY		
Lower Grid Assembly	Cracking of base metal and welds	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Reduction of Fracture Toughness of base metal and welds	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.

Table 4-1 Programs That Manage Applicable Reactor Vessel Internals Aging Effects (continued)

Affected Parts	Aging Effect	Programs That Manage Applicable Aging Effects
LOWER INTERNALS ASSEMBLY (continued)		
Lower Internals Assembly to Thermal Shield Bolts	Cracking	<u>ASME Section XI ISI Programs</u> *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval <u>RVI Aging Management Program</u> See Section 4.6.
	Loss of Closure Integrity	<u>ASME Section XI ISI Programs</u> *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval <u>RVI Aging Management Program</u> See Section 4.6.
Fuel Assembly Support Pads	Loss of Material	<u>ASME Section XI ISI Programs</u> *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval.
Guide Blocks	Loss of Material	<u>ASME Section XI ISI Programs</u> *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval.

Table 4-1 Programs That Manage Applicable Reactor Vessel Internals Aging Effects (continued)

Affected Parts	Aging Effect	Programs That Manage Applicable Aging Effects
LOWER INTERNALS ASSEMBLY (continued)		
Shell Forging to Flow Distributor Bolts	Cracking	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
	Loss of Closure Integrity	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
Lower Grid Rib to Shell Forging Screws	Loss of Closure Integrity	ASME Section XI ISI Programs *Examination Category B-N-3, Removable Core Support Structures* VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6.
Incore Guide Tube Spider Castings	Reduction of Fracture Toughness	RVI Aging Management Program See Section 4.6.

5. SUMMARY AND CONCLUSIONS

The B&W Owners Group Generic License Renewal Program has performed a review of the reactor vessel internals in accordance with 10 CFR 54.21 (a)(3) for the following B&W operating plants: Arkansas Nuclear One Unit 1 (ANO-1), Oconee Nuclear Station Units, 1, 2, and 3 (ONS -1,2,3) and Three Mile Island Unit 1 (TMI-1). Components within the scope of this report include all major structural components that are located within but not integrally attached to the reactor vessel. The fuel assemblies, control rod assemblies, and incore instrumentation are not considered part of the reactor vessel internals and are not evaluated in this report.

The reactor vessel internals component intended functions are:

1) Provide support and orientation of the reactor core (i.e., the fuel assemblies).

2) Provide support, orientation, guidance, and protection of the control rod assemblies.

3) Provide a passageway for the distribution of the reactor coolant flow to the reactor core.

4) Provide a passageway for support, guidance, and protection for the incore instrumentation.

5) Provide a secondary core support for limiting the downward displacement of the core support structure in the event of a postulated failure of the core barrel.

Maintaining the structural integrity of the reactor vessel internals will assure that their component intended functions are maintained within the scope of license renewal in the period of extended operation. The purpose of this report is to demonstrate that the aging effects for the reactor vessel internals are adequately managed in accordance with the requirements of 10 CFR 54.21 (a)(3) and 10 CFR 54.21 (c)(1) for the period of extended operation associated with license renewal.

The approach for demonstrating the management of aging effects on the reactor vessel internals was to determine applicable aging effects by reviewing materials of construction, operating environment, operating stresses, industry experience, and NRC generic communications and then to determine how existing regulated programs and commitments manage those identified aging effects. Applicable aging effects that can challenge the reactor vessel internals intended function during the period of extended operation and the programs that are credited for managing applicable aging effects are summarized in Section 5.1.

Time limited aging analyses (TLAA) applicable to the scope of the reactor vessel internals include: (1) FIV endurance limit assumptions, (2) transient cycle count assumptions used in fatigue cumulative usage factor calculations, (3) the effect of neutron irradiation on reactor vessel internals material properties and deformation limits, and (4) flaw growth acceptance in accordance with the ASME B&PV Code, Section XI, inservice inspection requirements. Flaw growth acceptance in accordance with Section XI is plant-specific and is not addressed in this report; all remaining TLAA are evaluated in this report.

5.1 Demonstration of Aging Management

The reactor vessel internals items within the scope of this report are divided into four main component groupings:

1) Plenum Assembly
2) Core Support Shield Assembly
3) Core Barrel Assembly
4) Lower Internals Assembly

5.1.1 Plenum Assembly

Aging Effects
Specific aging effects applicable to the plenum assembly include: (1) cracking due to IASCC and reduction of fracture toughness due to irradiation embrittlement of the upper grid assembly base metal and welds; (2) loss of material of the fuel assembly support pads and the plenum rib pads due to wear; (3) loss of closure integrity of the CRGT flange to upper grid screws due to stress relaxation; and (4) reduction of fracture toughness due to thermal embrittlement and its potential synergistic effect with neutron fluence of the CRGT assembly spacer castings.

Aging Management Programs
Loss of material of the fuel assembly support pads and plenum rib pads are managed by the requirements and acceptance standards for ASME Code Section XI, Table IWB 2500-1, Examination Category B-N-3.

Cracking and reduction of fracture toughness of the upper grid assembly base metal and welds and loss of closure integrity of the CRGT flange to upper grid screws may not be adequately managed by Examination Category B-N-3. As such, the GLRP has implemented a program to manage these aging effects of these reactor vessel internals component items including the aging management of the CRGT assembly spacer castings due to thermal embrittlement. This aging management program is discussed in Section 4.6. Applicants crediting this report in a license renewal application must track the commitment to implement the aging management program or conclusions resulting from the program.

5.1.2 Core Support Shield Assembly

Aging Effects
Applicable aging effects for the core support shield assembly include: (1) loss of material of the core support shield cylinder top flange due to wear; (2) cracking due to IASCC and SCC, reduction of fracture toughness due to irradiation embrittlement, and loss of closure integrity due

to stress relaxation of the core support shield to core barrel bolts; (3) reduction of fracture toughness due to thermal embrittlement and its potential synergistic effect with neutron fluence of the ONS-3 outlet nozzles; (4) reduction of fracture toughness due to thermal embrittlement and its potential synergistic effect with neutron fluence of the internal vent valve bodies and retaining rings; (5) cracking of the Alloy 600 locking devices on the modified vent valve assembly; and (6) wear of the locking devices on the original vent valve assembly.

Aging Management Programs
Loss of material of the core support shield cylinder top flange is managed by the requirements and acceptance standards for ASME Code Section XI, Table IWB 2500-1, Examination Category B-N-3.

Reduction of fracture toughness of the ONS-3 outlet nozzles is managed by the requirements and acceptance standards for ASME Code Section XI, Table IWB 2500-1, Examination Category B-N-3 and supplemented by flaw evaluation procedures as discussed in Section 4.2.Reduction of fracture toughness, cracking, and wear of the above mentioned internal vent valve assembly parts are managed by the requirements and acceptance standards for ASME Section XI, Table IWB 2500-1, Examination Category B-N-3 and supplemented by vent valve testing and inspection requirements from the ANO-1 and TMI-1 technical specifications and the Oconee Pump and Valve In-Service Test Program. The vent valve bodies are further supplemented by flaw evaluation procedures as discussed in Section 4.2. These CASS items are further supplemented by the RV Internals Aging Management Program as discussed in Section 4.6.

Cracking, reduction of fracture toughness, and loss of closure integrity of the core support shield to core barrel bolts may not be adequately managed by Examination Category B-N-3. As such, the GLRP has implemented a program to manage these aging effects for those reactor vessel internals component items. This aging management program is discussed in Section 4.6. Applicants crediting this report in a license renewal application must track the commitment to implement the aging management program or conclusions resulting from the program.

5.1.3 Core Barrel Assembly

Aging Effects
Applicable aging effects for the core barrel assembly include: (1) cracking due to IASCC and reduction of fracture toughness due to irradiation embrittlement of the core barrel assembly base metal and welds; (2) cracking due to IASCC, reduction of fracture toughness due to irradiation embrittlement, and loss of closure integrity due to stress relaxation of the baffle-to-baffle bolts and the baffle-to-former bolts; (3) cracking due to IASCC and SCC, reduction of fracture toughness due to irradiation embrittlement, and loss of closure integrity due to stress relaxation of the lower internals assembly to core barrel bolts; (4) cracking due to SCC and loss of closure integrity due to stress relaxation of the core barrel to thermal shield bolts; and (5) change in dimension due to void swelling of the baffle and former plates, baffle-to-baffle bolts, and baffle-to-former bolts.

Aging Management Programs
Cracking, reduction of fracture toughness, loss of closure integrity, and dimensional change for the above reactor vessel internals component items may not be adequately managed by Examination Category B-N-3. As such, the GLRP has implemented a program to manage these aging effects for those reactor vessel internals component items. This aging management program is discussed in Section 4.6. Applicants crediting this report in a license renewal application must track the commitment to implement the aging management program or conclusions resulting from the program.

5.1.4 Lower Internals Assembly

Aging Effects
Applicable aging effects for the lower internals assembly include: (1) cracking due to IASCC and reduction of fracture toughness due to irradiation embrittlement of the lower grid assembly base metal and welds; (2) cracking due to SCC and loss of closure integrity due to stress relaxation of the lower internals assembly to thermal shield bolts and the shell forging to flow distributor bolts; (3) loss of material due to wear of the fuel assembly support pads and the guide blocks; (4) loss of closure integrity due to stress relaxation of the lower grid rib to shell forging screws; and (5) reduction of fracture toughness due to thermal embrittlement and its potential synergistic effect with neutron fluence of the incore guide tube spider castings.

Aging Management Programs
Loss of material of the fuel assembly support pads and guide blocks are managed by the requirements and acceptance standards for ASME Code Section XI, Table IWB 2500-1, Examination Category B-N-3.

Cracking, reduction of fracture toughness, and loss of closure integrity for the above reactor vessel internals component items may not be adequately managed by Examination Category B-N-3. As such, the GLRP has implemented a program to manage these aging effects for those reactor vessel internals component items. This aging management program is discussed in Section 4.6. Applicants crediting this report in a license renewal application must track the commitment to implement the aging management program or conclusions resulting from the program.

5.1.5 Time Limited Aging Analyses

Time limited aging analyses (TLAA) applicable to the scope of the reactor vessel internals include: (1) FIV endurance limit assumptions, (2) transient cycle count assumptions used in fatigue cumulative usage factor calculations, (3) the effect of neutron irradiation on reactor vessel internals material properties and deformation limits, and (4) flaw growth acceptance in accordance with the ASME B&PV Code, Section XI, inservice inspection requirements. Flaw growth acceptance in accordance with Section XI is plant-specific and is not addressed in this report; all remaining TLAA are evaluated in this report.

Evaluations

Fatigue - FIV Endurance Limit Assumptions

The FIV endurance limit stress values were based on an assumed value of 10^{12} cycles for 40 years of operation. The methodology was extended to 60 years by utilizing the more recent ASME fatigue curves and assuming 1.5 times more allowable cycles. The component stress values from BAW-10051 were compared to the recalculated endurance limit stress values and shown acceptable. Therefore, this TLAA is resolved per 10 CFR 54.21(c)(1)(ii).

Fatigue - Transient Cycle Count Assumptions

Since the original internals design implicitly considered the reactor coolant system transient cycle assumptions, validation of these assumptions for 60 years of operation will assure the original design intent of the internals is maintained. Since, the design transients applicable for 40 years of operation remain valid for 60 years of operation with no increase in the number of transients anticipated, fatigue of the reactor vessel internals, implicit in the original design, is acceptable for 60 years of operation in accordance with 10 CFR 54.21(c)(1)(i).

The fatigue usage factor calculation for the replacement bolts consists of two components: fatigue due to design transients and fatigue due to FIV. It was demonstrated that the fatigue design transients applicable for 40 years of operation remain valid for 60 years of operation in accordance with 10 CFR 54.21(c)(1)(i). It was also demonstrated that the FIV usage factors when compared with the design transient usage factors are acceptable by extending the FIV methodology from 40 years to 60 years in accordance with 10 CFR 54.21 (c)(1)(ii). Therefore, this TLAA is resolved per a combination of 10 CFR 54.21 (c)(1)(i) and (ii). Plants must continue to monitor and track occurrences of the design transients.

Ductility (Reduction of Fracture Toughness)

The effects of irradiation on the material properties and deformation limits was shown acceptable under LOCA and combined LOCA and seismic loadings for 40 years of operation. With the additional neutron irradiation for the period of extended operation, reduction of fracture toughness is a concern. The existing Section XI Examination Category B-N-3 may not be adequate to detect reduction of fracture toughness in components. The GLRP has implemented a program to manage the effects of aging due to the reduction of fracture toughness of the reactor vessel internals. This aging management program is discussed in Section 4.6. Hence, this TLAA will be resolved on a plant-specific basis per 10 CFR 54.21 (c)(1)(iii) based on the results and conclusion of the program.

5.2 Conclusions

The foregoing evaluations demonstrate that the effects of aging on the components within the scope of this report will be adequately managed so that the reactor vessel internals component function will be adequately maintained consistent with the CLB for the period of extended operation. These evaluations meet the requirements of 10 CFR 54.21(a)(3) and 10 CFR 54.21 (c)(1).

Based on the evaluations that have been performed, actions have been identified and either 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 components that are within the scope of this report. By implementing these actions, there is, and will continue to be, reasonable assurance that licensed plant operations will continue to be conducted in accordance with the CLB for the components within the scope of this report.

It is the intention of the utilities of the B&W Owners Group that upon completion of the renewal applicant action items as listed in the NRC final safety evaluation, referencing this topical report in plant-specific license renewal applications, and summarizing in an FSAR supplement the aging management programs and the TLAA evaluations contained in this topical report will constitute an acceptable basis for the NRC finding required by 10 CFR 54.29 for the scope of components addressed in this report.

6. CERTIFICATION

This report is an accurate description of the evaluation of aging effects management of the reactor vessel internals of the participants in the B&W Owners Group Generic License Renewal Program.

3/8/00

Frank M. Gregory Date
Supervisory Engineer

3/8/00

Mark A. Rinckel Date
GLRP Project Engineer

This report was reviewed and found to be an accurate description of the work reported.

3/23/00

Gregory D. Robison Date
Project Engineer, Duke Power Company

3/21/00

David J. Masiero Date
Project Engineer, GPU Nuclear Corporation

3/8/00

Robert W. Clark Date
Entergy Operations, Inc.

This report is approved for release.

3/30/00

William R. Gray Date
GLRP Program Director, FTI

7. REFERENCES

1. Title 10, Chapter 1, Code of Federal Regulations - Energy, Part 54 - Requirements for Renewal of Operating Licenses for Nuclear Power Plants, US NRC, as modified in Federal Register/ Vol. 60, No. 88/ Monday, May 8, 1995/ Final Rule, 10 CFR Part 54.

2. E.O. Hooker, "Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibrations," BAW-10051, Babcock & Wilcox[1], Lynchburg, VA, September 1972.

3. "Pressurized Water Reactor Vessel Internals License Renewal Industry Report," Revision 1, EPRI TR-103838, Electric Power Research Institute, Palo Alto, CA, July 1994.

4. V.N. Shah, A.G. Ware, and A.M. Porter, Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Cracking, NUREG/CR-6245, EGG-2715.

5. K.H. Luk, "Pressurized-Water Reactor Internals Aging Degradation Study," NUREG/CR-6048, Oak Ridge National Laboratory, September 1993.

6. M. Kodama, S. Nishimura, J. Morisawa, S. Suzuki, S. Shima, M. Yamamoto, "Effects of Fluence and Dissolved Oxygen on IASCC in Austenitic Stainless Steels," Fifth International Symposium on Environmental Degradation of Material in Nuclear Power Systems - Water Reactors, August 1991, p. 948.

7. G. Vames and G. Robison, PWR reactor Coolant System License Renewal Industry Report; Revision 1, NUMARC-90-07-1, May 1992.

8. E. Howell and L.H. Vaughan, "Corrosion of Reactor Metal in Boric Acid, RDE-1086, Babcock & Wilcox, Alliance, Ohio, August 1960.

9. DELETED

10. Andresen, P.L., et al., "State of Knowledge of Radiation Effects on Environmental Cracking in Light Water Reactor Core Materials," Proceedings of the Fourth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Jekyll Island, Georgia, August 6-10, 1989, pp. 1-83 - 1-120.

11. "EdF Replaces Control Rod Guide Tubes," Nuclear Engineering International, Nov. 1982, p.2.

12. "Evaluation of Internals Bolting Concerns in 177 FA Plants," BAW-1843PA, Babcock & Wilcox[1], Lynchburg, VA, January 1986.

---

[1] Available from Framatome Technologies, Lynchburg, Va.

13. Kenik, E.A., et al., "Irradiation-Assisted Stress Corrosion Cracking," Journal of Nuclear Metals, Volume 211, 1994, pp. 52-59.

14. J.F. Copeland, Draft Commentary - ASME Code Section III, RP1757-61, Structural Integrity Associates, Inc., San Jose, California, October 1986.

15. H.H. Uhlig, Corrosion and Corrosion Control, Wiley and Sons, Inc., c. 1963.

16. Yankee Atomic Electric Company to United States Atomic Energy Commission, "Evaluation of Yankee Vessel Cladding Penetrations," License No. DPR-3 (Docket No. 50-29), October 15, 1965.

17. DELETED

18. DELETED

19. Scott, P., "A Review of Irradiation Assisted Stress Corrosion Cracking," Journal of Nuclear Materials, Volume 211, 1994, pp. 101-122.

20. R. Maccary, Nondestructive Examination Acceptance Standards, EPRI NP-1406-SR, Electric Power Research Institute, Pala Alto, CA, May 1980.

21. DELETED

22. D.E. Thoren and R.J. Harris, "Prototype Vibration Measurements Program for Reactor Internals - 177 Fuel Assembly Plant," BAW-10038, Babcock & Wilcox[1], Lynchburg, VA, September 1972.

23. D.E. Thoren and H.J. Fortune, "Prototype Vibration Measurement Results for B&W's 177 Fuel Assembly, Two-Loop Plant," BAW-10039, Babcock & Wilcox[1], Lynchburg, VA, April 1973.

24. "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake," BAW-10008, Part 1, Rev. 1, Babcock & Wilcox[1], Lynchburg VA, June 1970

25. DELETED

26. ASME Boiler and Pressure Vessel Code, 1965 Edition with Addenda through and including the 1967 Summer Addenda.

27. DELETED

28. R. Winks, "Fatigue Trackable Components and Transients," Proprietary, 51-1234566-02, Framatome Technologies, Inc., Lynchburg, VA, June 1996.

29. DELETED

30. G. Robison, E. Grubbs, M. Rinckel, and R. Starkey, "Demonstration of the Management of Aging Effects for the Reactor Coolant Piping System," BAW-2243A, Framatome Technologies, Inc., June 1996.

31. ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1989 Edition, American Society of Mechanical Engineers.

32. "BAW 177-FA Owners Group - Effects of Asymmetric LOCA Loadings - Phase II Analysis," BAW-1621, Babcock & Wilcox[1], Lynchburg VA, July 1980.

33. "Evaluation of Internals Bolting Concerns in 177 FA Plants," BAW-1789P, Babcock and Wilcox[1], Lynchburg, VA, August 1984.

34. DELETED

35. R. Nickell and M.A. Rinckel, "Evaluation of Thermal Aging Embrittlement for Cast Austenitic Stainless Steel Components", EPRI-TR-106092, Research Project 2643-33, Final Report, March 1996.

[1] Available from Framatome Technologies, Lynchburg, Va.

APPENDIX A

FSAR Supplement Summary Description

Reactor Vessel Internals Aging Management Program

1) [Utility/Applicant] shall establish and maintain a Reactor Vessel Internals Aging Management Program that includes, but is not limited to, the following activities:

   (a) Continue the investigation of the potential aging effects that have been identified in BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals." The scope of the investigation shall include, but not be limited to, the development of key program elements to address the following aging mechanisms; IASCC, SCC, irradiation embrittlement, thermal embrittlement, void swelling, and stress relaxation.

   (b) Prior to [insert current date of current operating license], establish appropriate monitoring and inspection programs that will provide additional assurance that the Reactor Vessel Internals will remain functional through the licensed life of [plant].

2) [Utility/Applicant] shall provide the NRC with either annual reports or periodic updates (after the completion of significant milestones) on the status of the above program activities, commencing within one year of the issuance of a renewed license.

APPENDIX B

Appendix B contains the NRC Request for Additional Information dated December 2, 1998

December 2, 1998

Mr. David J. Firth
Program Director
Generic License Renewal Program
The B&W Owners Group
1700 Rockville Pike, Suite 525
Rockville, MD 20852

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE BABCOCK & WILCOX OWNER'S GROUP GENERIC LICENSE RENEWAL PROGRAM TOPICAL REPORT ENTITLED, "DEMONSTRATION OF THE MANAGEMENT OF AGING EFFECTS FOR THE REACTOR VESSEL INTERNALS," BAW-2248, JULY 1997

Dear Mr. Firth:

By letter dated July 29, 1997, the Babcock & Wilcox Owners Group (BWOG) submitted Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," July 1997, requesting the U.S. Nuclear Regulatory Commission staff's review and issuance of a safety evaluation report.

Based on the review of the information submitted, the staff has identified, in the enclosure, areas where additional information is needed to complete the review.

Please provide a schedule for the submittal of your response within 30 days of the receipt of this letter. Additionally, the staff is willing to meet with the BWOG before you submit your response to clarify the staff's request for additional information.

Sincerely,

/Signed/

Raj K. Anand, Project Manager
License Renewal Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Project No. 683

Enclosure: As stated

cc w/encl: See next page

Project No. 683

Mr. Rick Edwards
Framatone Technologies
1700 Rockville Pike
Suite 525
Rockville, Maryland 20852

Mr. Michael Laggart
Manager, Corporate Licensing
GPU Nuclear Corporation
One Upper Pond Road
Parsippany, New Jersey 07054

Chairman
Board of County Commissioners
of Dauphin County
Dauphin County Courthouse
Harrisburg, Pennsylvania 17120

Mr. J. W. Hampton
Nuclear Generation Vice President
Duke Energy Corporation
Oconee Nuclear Station
MC: ONO 1VP
P. O. Box 1439
Seneca, South Carolina 29670

Mr. John R. McGaha, Vice President
Operations Support Licensing
Entergy Operations, Inc.
P. O. Box 31995
Jacksonville, Mississippi 39286

William R. McCollum, Jr
Duke Energy Corporation
P.O. Box 1439
Seneca, SC 29679

Mr. R. L. Gill
GLRP Licensing Coordinator
C/O Duke Energy Corporation
EC-12R
P. O. Box 1006
Charlotte, North Carolina 28201-1006

Gregory D. Robison
Duke Energy Corporation
Mail Stop EC-12R
P.O. Box 1006
Charlotte, North Carolina 28201-1006

Babcock & Wilcox Owners Group
Generic License Renewal Program

Mr. James J. Fisicaro
Director, Licensing
Entergy Operations, Inc.
Route 3, Box 137G
Russelville, Arkansas 72801

Earnest L. Blake, Jr., Esq
Shaw, Pitman, Potts
and Trowbridge
2300 N. Street, N.W.
Washington, DC 20037

M. S. Tuckman
Duke Energy Corporation
Mail Stop EC-07H
P.O. Box 1006
Charlotte, North Carolina 28201-1006

B. Gutherman, Manager
Florida Power Corporation, (GA2A)
Crystal River Energy Complex
15760 W. Power Street
Crystal River, Florida 34428-6708

Mr. William Dornsife, Acting Director
Bureau of Radiation Protection
Pennsylvania Department of
Environmental Resources
P. O. Box 2063
Harrisburg, Pennsylvania 17120

Chairman
Board of Supervisors
of Londonderry Township
R.D. #1 Geyes Church Road
Middletown, Pennsylvania 17057

Mr. J. E. Burchfield
Compliance Manager
Duke Energy Corporation
Oconee Nuclear Site
P. O. Box 1439
Seneca, South Carolina 29679

Douglas J. Walters
Nuclear Energy Institute
1776 I Street, NW
Suite 400
Washington, DC 20006-3708
DJW@NEI.ORG

REQUEST FOR ADDITIONAL INFORMATION

THE BABCOCK & WILCOX OWNER'S GROUP GENERIC LICENSE RENEWAL PROGRAM TOPICAL REPORT ENTITLED, " DEMONSTRATION OF THE MANAGEMENT OF AGING EFFECTS FOR THE REACTOR VESSEL INTERNALS," BAW-2248, JULY 1997

1) Section 1.4 of BAW-2248 identifies the functions of the reactor vessel internals (RVI). It does not include, "provide shielding for the RPV [reactor pressure vessel]." Do the intended functions of the reactor vessel internals "provide shielding for the reactor pressure vessel"?

2) BAW-2248 addresses certain applicable aging effects for specific reactor vessel internals components. Describe, in summary form, the bases for concluding that the following aging effects were not significant for the specific components: stress corrosion cracking (SCC), and irradiation-assisted stress corrosion cracking (IASCC) of the plenum cover and plenum cylinder; SCC, IASCC, wear, and thermal embrittlement of the rod control assembly (CRA) guide tubes; SCC and IASCC of the CRA guide tube bolts; SCC of the upper grid assembly; SCC and IASCC of the upper grid rib section, upper grid assembly bolts, and the upper internals fuel guide pads; SCC, IASCC, and neutron irradiation embrittlement of the core support shield and the core support shield flange; IASCC and neutron irradiation embrittlement of the vent valve assemblies; SCC of the core barrel assembly; SCC, IASCC, creep, and neutron irradiation embrittlement of the baffle and former plates; SCC, and stress relaxation of the baffle-former bolts; SCC of the lower grid top rib section, the lower grid bottom rib weldment, and the lower grid assembly support posts; SCC, IASCC, and neutron irradiation embrittlement of the lower internals fuel guide pads, the lower grid assembly bolts, and the lower grid assembly guide blocks.

3) Section 54.21(a)(3) of 10 CFR states that the integrated plant assessment must " demonstrate that the effects of aging will be adequately managed...for the period of extended operation." Section 4.6 of BAW-2248 describes a proposed Reactor Vessel Internals Aging Management Program (RVIAMP) that is intended to meet 10 CFR 54.21(a)(3), for the affected part and aging effect combinations listed in Table 4-1 of BAW-2248. For those items listed in Table 4-1 susceptible to a reduction of fracture toughness aging mechanism, submit a fracture mechanics analysis to determine the critical flaw size during normal operation and during emergency and faulted conditions. Identify the inspection procedure and the capability of the inspection to detect flaws smaller in size than that of the critical flaw.

   The fracture toughness of austenitic stainless steel can become degraded with high levels of neutron irradiation; for example, fluences greater than 1×10^{20} n/cm^2 (E > 1MeV). Fracture toughness data for irradiated stainless steels at such high fluences are not plentiful. Two sources available in the public literature are References 1 and 2.

   The data in Reference 1 are for the initiation fracture toughness (i.e., at the initiation of crack growth), defined by:

   J_{Ic} is defined as the J-integral value at the initiation of crack growth and E is the Young's modulus for the material. For Type 304 stainless steel plate irradiated to a fluence of ~5×10^{20} n/cm^2 (E > 1MeV) at ~280 °C and tested at 288 °C, the lowest reported value in Reference 1 of J_{Ic} (~75 in.-lb/in.^{2}) corresponds to a K_{Jc} of ~50 ksi$\sqrt{in}$.

From Reference 2, J-integral resistance or J-R curve data are reported for two samples fabricated from core shroud material removed from an overseas boiling water reactor (BWR) (see figure attached). The fluence for these samples is reported in Reference 2 as 8×10^{20} n/cm^2.

Reconciliation of the J_{Ic} from Reference 1 with the J-R curve trends from Reference 2 (through scaling of the J levels in the J-R curves) can provide one estimate of the fracture toughness of highly irradiated austenitic stainless steel.

Provide any other fracture toughness data used in this evaluation.

4) Aging effects of many reactor vessel internal components will be managed by the Reactor Vessel Internals Aging Management Program. The program elements are discussed in Section 4.6 of BAW-2248. Provide a plan and schedule for completing all the elements of the program.

5) Table 4-1 of BAW-2248 indicates that management of reduction of fracture toughness in vent valve bodies and vent valve retaining rings will be accomplished principally by the American Society of Mechanical Engineers (ASME) Section Inservice Inspection (ISI) Program. This is in contrast to the treatment of other RVI components subject to loss of fracture toughness, for which the RVIAMP is set forth to manage the aging effects. Why are the vent valve components treated differently, and should they be included in the scope of the RVIAMP?

6) Table 4-1 of BAW-2248 indicates that the vent valve retaining ring, vent valve bodies, and the locking devices on the modified vent valve assembly do not require a supplemental aging management program. Aging effects will be managed during the renewal term using ASME Boiling and Pressure Vessel Code (Code) inspection methods. Since functions of these components are affected by either a reduction of fracture toughness or stress corrosion cracking, will ASME Code VT-3 visual examination be adequate for discovering cracks that could lead to failure of the component? What examination methodology is required? Are the surfaces of the components accessible for detecting cracks that could lead to failure of the components?

7) Page 3-5 of BAW-2248 states that "the ONS-1 CRGT assembly sectors required straightening after the first hot functional test (FHT)." Provide a summary of the evaluation indicating that the cracking mechanisms (SCC and IASCC) are not plausible in this case. If these guide tube sectors were to be degraded by a cracking mechanism,could such cracking impede the ability of the reactor vessel internals to perform its function to "provide support, orientation, guidance, and protection of the control rod assemblies"?

8) Examination Category B-N-3 of ASME Section XI requires a VT-3 visual examination of " accessible surfaces" of removable core support structures. For assemblies and parts determined to be susceptible to no aging mechanisms, and hence to require no additional aging management, the VT-3 examination provides one measure of assurance of the structural integrity of the part. Which components not susceptible to an aging mechanism (and hence, no additional aging management) will receive a VT-3 examination that can serve as a sampling of nonsusceptible components?

9) Section 3.3 of BAW-2248 indicates that crevice corrosion is not expected to be a concern, unless the internals are exposed to a series of long outages that have stagnation and high impurity levels. What impurity levels and how much cumulative outage time are required before crevice corrosion becomes a concern? What components could be affected by crevice corrosion? How could crevice corrosion be prevented if there were a long outage?

10) Section 3.3 of BAW-2248 indicates that wear is not a concern for the modified vent valve locking devices. Explain the difference between the original design and the modified design that eliminated the concern for wear of the vent valve locking device. To what criteria are the modified vent valve locking devices being inspected to ensure that they are not subject to wear? Summarize the results of these inspections; include the number and frequency of inspection. Will these inspections be continued into the license renewal term?

11) Section 3.3 of BAW- 2248 indicates that Westinghouse has observed wear of incore guide tubes caused by flow induced vibration in regions directly exposed to reactor coolant system (RCS) flow. Wear of Babcox & Wilcox (B&W)-designed guide tube and spiders are not a concern; however, because of differences between the Westinghouse and B&W guide tube design and because the B&W-designed detectors are inserted and withdrawn once per fuel cycle. Explain the difference in design and operation of the Westinghouse and B&W incore guide tubes that indicates wear is not a concern for the B&W design. Are there any limits on the number of insertions and withdrawals of the incore monitors that could lead to a concern about wear of the guide tubes?

12) During its interaction meetings with the staff (referenced in the Section 4.3.11 of the Oconee Nuclear Station License Renewal - Technical Information New Programs and Activities Report, June 1998), the B&W Owners Group (B&WOG) described current and ongoing reactor internals baffle bolt activities that included preparation for possible augmented baffle bolt inspection during the next 10-year ISI interval at Oconee 1 (2003 at the earliest). Describe baffle bolt inspections that will be conducted prior to the start of the extended license renewal period and indicate how these actions provide the basis for assuring the baffle bolt monitoring and inspection techniques that are planned during the period of extended operation are appropriate.

13) Describe the program that will be implemented as outlined in Section 4.6 of BAW-2248 with regard to the aging management of the reactor internals baffle bolts. Describe the overall inspection program, including aspects such as, intervals, monitoring, and inspection techniques.

14) Describe the replacement bolts and redesigned RVI that are referred to in the fatigue analysis discussed in BAW-2248 Section 4.5.1. Are the replacement bolts and redesigned RVI identified in the fatigue analysis related to the issue of the A-286 bolt cracking discovered in B&W RVI? Are the baffle bolts discussed in the BAW-2248 report included in the fatigue analysis? If not, what is the basis for not including the baffle bolts in the fatigue analysis? If the baffle bolts are included in the analysis, describe how baffle bolt cracking is taken into account and identify the analysis report.

REFERENCES

1. Loss, F. J., and Gray, Jr., R. A., "J-Integral Characterization of Irradiated Stainless Steels," NRL Report 7565, Naval Research Laboratory, Washington, D. C., April 25, 1973.

2. EPRI TR-107079, "BWR Vessel and Internals Project, BWR Core Shroud Inspection and Flaw Evaluation Guideline, Revision 2 (BWRVIP-01)," October 1996, pp. 4-13.

Attachment: Figure

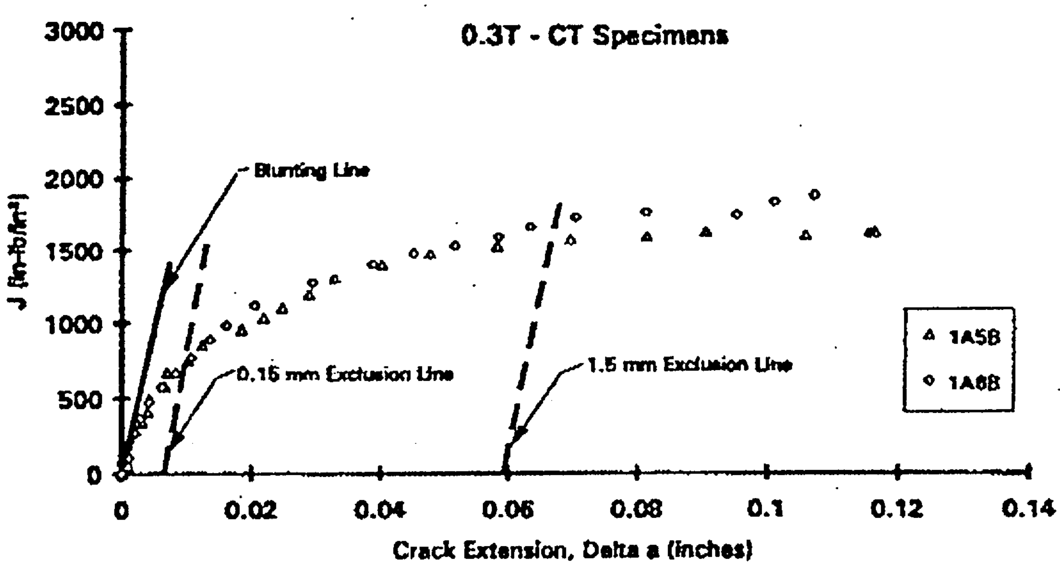


Figure J-R curve data for irradiated stainless steel for a fluence of 8×10^{20} n/cm^2 (Ref. 2).

APPENDIX C

Appendix C contains the GLRP responses to the NRC Request for Additional Information dated February 18, 1999

Duke Energy Corporation | Oconee 1, 2, 3
Entergy Operations, Inc. | ANO-1
Florida Power Corporation | Crystal River

The B&W Owners Group

GPU Nuclear, Inc. | TMI-1
Toledo Edison Company | Davis-Bessee
Framatome Technologies, Inc

Working Together to Economically Provide Reliable and Safe Electrical Power

Suite 525 • 1700 Rockville Pike • Rockville, MD 20852 • (301) 230-2100

February 18, 1999
OG-1744

Project No. 683

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. David B. Matthews, Deputy Director
Division of Reactor Program Management

Subject: B&WOG Generic License Renewal Program Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (RAIs 1 through 14 from December 4, 1998"

References:
1) Letter from Raj K. Anand to David J. Firth, December 2, 1998, entitled "Request for Additional Information Regarding The Babcock & Wilcox Owners Group Generic License Renewal Program Topical Report Entitled, Demonstration of The Management Of Aging Effects for The Reactor Vessel Internals, BAW-2248, July 1997."
2) NRC Meeting Summary dated May 6, 1998, entitled "Summary of Meeting on April 23, 1998, Between the U.S. Nuclear Regulatory Commission and B&WOG Representatives to Discuss the Status of the B&WOG Generic License Renewal Program, Project Number 683."

Gentlemen:

Reference 1 contains a Request for Additional Information (RAI) on the subject document. Each RAI, the B&WOG response, and the recommended change to the appropriate section of BAW-2248 are included in the attached table.

The technical elements of the B&WOG reactor vessel internals aging management program were presented to the NRC on April 23, 1998 [Reference 2]. Since the time of that meeting the industry has initiated a project to address generic materials issues and the scope of the B&WOG RV internals aging management program has changed. This is reflected in our attached RAI response.

The PWR Material Reliability Project (MRP) was established during the second quarter of 1998 to proactively address and resolve, on a consistent industry-wide basis, existing and emerging PWR material-related issues. This group is directed by utilities and managed by EPRI. There is a close coordination with, and direct participation by the major US NSSS vendors, their associated Owners Groups, and NEI. Based on a prescribed issue selection criteria (per MRP charter) an Issue Task Group (ITG) was formed to manage the emerging reactor pressure vessel (RPV) internals material issues. The MRP Integration and Implementation Group (IIG) will provide management oversight to all ITGs and will report to the same executive group as the Steam Generator Management Project (SGMP).

The RPV Internals ITG has developed generic programs that address applicable aging effects such as cracking, reduction of fracture toughness and loss of preload. The mechanisms presently being considered are irradiation assisted stress corrosion cracking (IASCC), stress corrosion cracking (SCC), irradiation embrittlement (IE), swelling, and stress relaxation. One such program is a predictive model for IASCC of baffle bolts. In addition, the ongoing EPRI Joint Baffle Bolt (JoBB) Program established in 1996 has been incorporated into the RPV Internals ITG to provide plant inspection and research test data.

The Owner Groups will utilize data and information from the ITG programs to perform design-specific analyses, i.e., safety evaluations. The B&WOG will provide support to the ITG by providing plant- and design-specific data, such as fluence and stress profiles. The utilities will be responsible for using the tools provided by both the ITG and Owners Groups to determine the necessary steps (e.g., inspections, operability determinations, and replacements) to manage the applicable aging effects.

We anxiously look forward to receiving the draft safety evaluation on BAW-2248 and are prepared to work with the Staff to resolve any additional items that may arise.

Please call me at 804/832-2783 if you need any additional information.

Sincerely,

[redacted]

William R Gray
Project Manager
B&W Owners Group Services

WRG\mcl

Attachment 1: RAI Response Table

c: w/Attachment

R. K. Anand	- US NRC/NRR
R. L. Gill	- Duke Energy Corporation
G. D. Robison	- Duke Energy Corporation
W. F. Brady	- Duke Energy Corporation
W. H. Mackay	- Entergy Operations, Inc.
R. W. Clark	- Entergy Operations, Inc.
D. J. Masiero	- GPU Nuclear, Inc.
G. L. Lehmann	- GPU Nuclear, Inc.
M. A. Rinckel	- Framatome Technologies
R. N. Edwards	- Framatome Technologies

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
1.	Section 1.4 of BAW-2248 identifies the functions of the reactor vessel internals (RVI). It does not include, "provide shielding for the RPV [reactor pressure vessel]." Do the intended functions of the reactor vessel internals "provide shielding for the reactor pressure vessel"?	The function entitled "provide shielding for the reactor pressure vessel" is not within the scope of BAW-2248. Each license renewal applicant that references BAW-2248 must determine if the function entitled "provide shielding for the reactor pressure vessel" is a reactor vessel internals intended function. If not an intended function, the license renewal applicant should provide justification for that conclusion. Should a license renewal applicant determine that the function entitled "provide shielding for the reactor pressure vessel" is an intended function of the reactor vessel internals, the items that support this function must be identified and reviewed in accordance with 54.21(a)(3). For example, the thermal shield was eliminated from the scope of aging management review in BAW-2248 since it did not support an intended function. BAW-2248 will be revised to state that the thermal shield is not within the scope of the report.	Open	<u>Section 2.0, Page 2-1, Line 17</u> Delete the line beginning with "The thermal shield...of this report." and replace with the following two lines. The thermal shield is not within the scope of this report (see the discussion in section 2.5.1). The thermocouple guide tube assemblies do not perform intended functions as defined in 10 CFR 54 [1] and are, therefore, not within the scope of this report. <u>Section 2.5.1, Page 2-18, First Paragraph, Line 6</u> Delete the first paragraph and replace with the following text. The thermal shield and its restraining devices are described here for the benefit of the reviewers and are not within the scope of this report. Failures of the thermal shield bolts, which are documented in Section 3.6, resulted in the B&WOG reactor internals bolting integrity program. As such, the thermal shield bolts are subject to aging management review and will be addressed in this report.	
2.	BAW-2248 addresses certain applicable aging effects for specific reactor vessel internals components. Describe, in summary form, the bases for concluding that the following aging effects were not significant for the specific components: stress corrosion cracking (SCC), and irradiation-assisted stress corrosion cracking (IASCC) of the plenum cover and plenum cylinder; SCC, IASCC, wear, and thermal embrittlement of the rod control assembly (CRA) guide tubes; SCC and IASCC of the CRA guide tube bolts; SCC of the upper grid assembly; SCC and IASCC of the upper grid rib section, upper grid assembly	The cited reactor vessel internals items from the NRC RAI#2 are listed along with each relevant aging effect (and associated mechanism) in Table 1.	Open	<u>Section 3.4, page 3-11 (line 5), add the following:</u> Baffle and former bolts <u>Section 3.4, page 3-11, (line 29), add the following:</u> 8) Loss of closure integrity of the baffle and former bolts due to stress relaxation. <u>Section 4.4, page 4-7, add the following:</u> 8) Loss of closure integrity of the baffle and former bolts due to stress relaxation. <u>Table 4-1, page 4-17</u>, under "Baffle-to-Baffle Bolts", add "stress relaxation" in the column heading "Aging effect" as	

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
	bolts, and the upper internals fuel guide pads; SCC, IASCC, and neutron irradiation embrittlement of the core support shield and the core support shield flange; IASCC and neutron irradiation embrittlement of the vent valve assemblies; SCC of the core barrel assembly; SCC, IASCC, creep, and neutron irradiation embrittlement of the baffle and former plates; SCC, and stress relaxation of the baffle-former bolts; SCC of the lower grid top rib section, the lower grid bottom rib weldment, and the lower grid assembly support posts; SCC, IASCC, and neutron irradiation embrittlement of the lower internals fuel guide pads, the lower grid assembly bolts, and the lower grid assembly guide blocks.			a separate effect, identified with the identical Management Program as for Cracking and Reduction of Fracture Toughness. Table 4-1, page 4-18, under "Baffle-to-Former Bolts", add "stress relaxation" in the column heading "Aging effect" as a separate effect, identified with the identical Management Program as for Cracking and Reduction of Fracture Toughness	
3.	Section 54.21(a)(3) of 10 CFR states that the integrated plant assessment must "demonstrate that the effects of aging will be adequately managed...for the period of extended operation." Section 4.6 of BAW-2248 describes a proposed Reactor Vessel Internals Aging Management Program (RVIAMP) that is intended to meet 10 CFR 54.21(a)(3), for the affected part and aging effect combinations listed in Table 4-1 of BAW-2248. For those items listed in Table 4-1 susceptible to a reduction of fracture toughness aging mechanism, submit a fracture mechanics analysis to determine the critical flaw size during normal operation and during emergency and faulted conditions. Identify the inspection procedure and the capability of the inspection to detect flaws smaller in size than that of the critical flaw.	Austenitic stainless steel core support assembly weld and plate materials were shown to have adequate resistance to unstable flaw growth, even after sustained exposure to neutron irradiation. The effect of sustained exposure to neutron irradiation is a reduction in the fracture toughness of the austenitic stainless steel materials. This potential reduction in fracture toughness has been evaluated by determining the critical flaw size and by demonstrating resistance to unstable flaw growth for representative PWR internals geometries under nominal, design-basis, and bounding loads. The elastic and elastic-plastic fracture mechanics calculations, plus the net (uncracked) cross section limit load estimates, were based on stress-strain data and elastic-plastic crack growth resistance curves that accounted for the effects of prolonged neutron irradiation (i.e., fluence of 8E20 n/cm2 E> 1 MeV per	Open		

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
	The fracture toughness of austenitic stainless steel can become degraded with high levels of neutron irradiation; for example, fluences greater than 1 x 10^{20} n/cm^2 (E > 1MeV). Fracture toughness data for irradiated stainless steels at such high fluences are not plentiful. Two sources available in the public literature are References 1 and 2. The data in Reference 1 are for the initiation fracture toughness (i.e., at the initiation of crack growth), defined by: $$K_{Jc} = \sqrt{J_{Ic} x E}$$ J_{Ic} is defined as the J-integral value at the initiation of crack growth and E is the Young's modulus for the material. For Type 304 stainless steel plate irradiated to a fluence of ~5 x 10^{20} n/cm^2 (E > 1MeV) at ~280 °C and tested at 288 °C, the lowest reported value in Reference 1 of J_{Ic} (~75 in.-lb/in.^{2}) corresponds to a K_{Jc} of ~50 ksi in. From Reference 2, J-integral resistance or J-R curve data are reported for two samples fabricated from core shroud material removed from an overseas boiling water reactor (BWR) (see figure attached). The fluence for these samples is reported in Reference 2 as 8 x 10^{20} n/cm^2. Reconciliation of the J_{Ic} from Reference 1 with the J-R curve trends from Reference 2 (through scaling of the J levels in the J-R curves) can provide one estimate of the fracture toughness of highly irradiated austenitic stainless steel. Provide any other fracture toughness	EPRI TR-107079). Five simplified geometries representative of PWR core support structures were examined: (1) an axially-loaded rectangular parallelepiped with a semi-elliptical horizontal surface flaw representing an edge-cracked core support columnar structure; (2) an axially-loaded rectangular parallelepiped with a quarter-elliptical horizontal surface flaw representing a corner-cracked core support columnar structure; (3) an internally-pressurized torus, with a longitudinal semi-elliptical surface flaw; (4) an axially-loaded torus, with a semi-elliptical surface flaw in the horizontal plane; and, (5) an axially-loaded cylinder with a circumferential through-wall flaw. The third and fourth geometries represent edge-cracked plate or shell core support structures with two different postulated flaw orientations and loading perpendicular to the plane of the flaw. The fifth geometry represents a shell core support structure with a circumferential through-wall flaw and loading perpendicular to the flaw. Flaw tolerance calculations were carried out for the representative core support assembly geometries in four steps. *First* postulated flaws were assumed. *Second*, negligible flaw growth was assumed prior to application of nominal, design-basis, and bounding loads. *Third*, applied J-integrals were calculated from the combination of the tensile stresses and the postulated flaws, using linear elastic fracture mechanics (LEFM) solutions obtained from literature and a conversion to elastic-plastic crack driving force valid for localized plasticity at the crack tip. As a check, approximate limit loads were calculated for the net (uncracked) section, in order to determine the validity of the applied J-integrals that were derived			

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
	data used in this evaluation.	from LEFM expressions. *Finally*, the evaluation procedure specified in ASME Section XI, Article K-4000, K-4220, was used to demonstrate flaw stability. Specifically, Paragraph K-4220 provides a flaw stability criterion that limits the elastic-plastic crack driving force to less than the material elastic-plastic resistance at a crack extension of 0.1 inches. The material elastic-plastic resistance at a crack extension of 0.1 inches is approximately 1,500 in-lb/in^2 for stainless steel specimens irradiated to 8.0E20 n/cm^2 in accordance with EPRI TR-107079 [BWRVIP-01], Revision 2, October 1996. The critical flaw size for marginally-stable crack growth for the columnar core support structure had a length extending completely along one edge of the column and half of the distance across the wall, with a flaw area covering 40 % of the column cross section. The critical flaw size for marginally-stable crack growth for the plate/shell core support structure had a length extending over 60 % through the vertical dimension (depth) of the torus and over 40 % through the torus thickness, with a flaw area exceeding 30 % of the torus cross-sectional area. Because of yield strength increases caused by prolonged neutron irradiation, the net (uncracked) ligament stresses are in the elastic range even for the critical flaw sizes reported above. The critical flaw size for a circumferenetial through-wall flaw in the shell support structure (i.e., core barrel assembly) is approximately 15 inches in length. Smaller through-wall flaws in the core barrel would be stable under design-basis and bounding loads.			

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
		These analyses show that the critical flaw sizes are so large that current ASME Code Section XI, Subsection IWB periodic visual (VT-3) inservice inspections of accessible surface areas, in accordance with Examination Category B-N-3, continue to be adequate to manage the effects of reduced fracture toughness for these reactor vessel internals welds and base metal. The evaluation presented above does not cover high-strength bolting applications. It is recognized that the issue of reduction of fracture toughness is considered an emerging issue that is being addressed prior to the license renewal period. Irradiation embrittlement (IE) is a task which is currently being evaluated as part of the reactor vessel internals issues task group (please see the cover letter). The reactor vessel internals ITG will assist the B&WOG, and other PWR Owners and Owners Groups, by performing work that is generically applicable to all of the participating PWRs. If it is determined that IE is a significant aging mechanism for bolting materials based on characterization and collection of material property, the B&WOG will use the data to perform evaluations of the B&W-designed internals to determine if additional activities or inspections are needed during or prior to the period of extended operation.			
4.	Aging effects of many reactor vessel internal components will be managed by the Reactor Vessel Internals Aging Management Program. The program elements are discussed in Section 4.6 of BAW-2248. Provide a plan and schedule for completing all the elements of the program.	The program that was developed to address the technical elements listed in Section 4.6 of BAW-2248 was presented to the NRC on April 23, 1998 [NRC Meeting Summary dated May 6, 1998, entitled "Summary of Meeting on April 23, 1998, Between the U.S. Nuclear Regulatory Commission and B&WOG Representatives to Discuss the Status of the B&WOG Generic License	Open	None	

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
		Renewal Program, Project Number 683."]. Since the time of that meeting, the industry has formed a reactor vessel internals issue task group (ITG) under the guidance of the industry sponsored Materials Reliability Program. The industry activities will supplement the B&WOG program as discussed in the cover letter.			
5.	Table 4-1 of BAW-2248 indicates that management of reduction of fracture toughness in vent valve bodies and vent valve retaining rings will be accomplished principally by the American Society of Mechanical Engineers (ASME) Section Inservice Inspection (ISI) Program. This is in contrast to the treatment of other RVI components subject to loss of fracture toughness, for which the RVIAMP is set forth to manage the aging effects. Why are the vent valve components treated differently, and should they be included in the scope of the RVIAMP?	The vent valve bodies are fabricated from cast austenitic stainless steel. As such, the aging management programs that are applicable for reduction of fracture toughness for the vent valve bodies are the ASME Section XI ISI Program, the Plant Technical Specifications for ANO-1 and TMI-1, and the Pump and Valve In-Service Test Program for the Oconee units. These programs will be further supplemented by the extension of the evaluation procedures in IWB-3640 to CASS items as specified in the GLRP RCS Piping Topical Report (BAW-2243A). The vent valve retaining rings are fabricated from precipitation-hardening stainless steel materials and are potentially susceptible to reduction of fracture toughness by thermal embrittlement. However, the retaining rings are in compression and reduction of fracture toughness will not compromise the function of the retaining rings. BAW-2248 lists the ASME Section XI ISI Program as an applicable program to manage this aging effect. The vent valves are tested and inspected each refueling outage through the Plant Technical Specifications for ANO-1 and TMI-1 and the Pump and Valve In-Service Test Program for the Oconee units. This supplemental program of vent valve testing and inspection will be added for the retaining rings. This supplemental program	Open	<u>p4-6, replace lines 18 – 22 with the following</u>: Reduction of fracture toughness of the vent valve bodies and retaining rings due to thermal embrittlement are managed by Examination Category B-N-3. Further, these components are managed by technical specification requirements for ANO-1 and TMI-1 and the Pump and Valve In-Service Test Program for the Oconee units, which require vent valve testing and a visual inspection of the internal vent valve components. However, for the vent valve bodies, these programs will be further supplemented by the evaluation procedures for potential flaws in cast austenitic stainless steel as mentioned above. <u>p4-6, modify line 25 with the following</u>: ... vent valve testing and inspection ... <u>p4-16, under programs for the vent valve bodies add</u>: Further supplemented by extension of the evaluation procedures in IWB-3640 to CASS items as specified in Section 4.2 of BAW-2243A[30]. <u>p4-16, under programs for the vent valve retaining rings add:</u> Supplemented by: • <u>Plant Technical Specifications (ANO-1, TMI-1)</u> *Vent valve testing and inspection requirements each refueling outage* • <u>Pump and Valve In-Service Test Program (ONS-1,-2,-3)</u> *Vent valve testing and inspection requirements each refueling outage*	

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
		along with the ASME Section XI ISI program manages the aging effects of the retaining rings for the period of extended operation.			
6.	Table 4-1 of BAW-2248 indicates that the vent valve retaining ring, vent valve bodies, and the locking devices on the modified vent valve assembly do not require a supplemental aging management program. Aging effects will be managed during the renewal term using ASME Boiling and Pressure Vessel Code (Code) inspection methods. Since functions of these components are affected by either a reduction of fracture toughness or stress corrosion cracking, will ASME Code VT-3 visual examination be adequate for discovering cracks that could lead to failure of the component? What examination methodology is required? Are the surfaces of the components accessible for detecting cracks that could lead to failure of the components?	Aging management of the vent valve bodies and retaining rings are addressed in RAI #5. Management of the vent valve locking devices is supplemented by technical specification requirements for ANO-1 and TMI-1 and the Pump and Valve In-Service Test Program for the Oconee units. These specifications and test programs include a provision to perform testing and a visual inspection of the internal vent valve. This inspection includes the valve body, disc seating surfaces and the locking devices. Table 4-1 will be modified to include this supplemental inspection of the vent valve locking devices. Cracking by primary water stress corrosion cracking is considered a plausible aging effect for the parts of the internal vent valve modified locking devices that are fabricated from Alloy 600 materials. These parts are located at the connection of the jackscrew to the retaining ring. In addition, there is redundancy in the design since there are two connections (one on each side of the retaining ring). The LRTF believes that the visual inspections required in accordance with Examination Category B-N-3 as supplemented by testing and inspection that the vent valves receive each outage will manage the aging effects of the locking devices for the period of extended operation.	Open	<u>p4-3, modify the last sentence on lines 37 - 39 with the following:</u> The aging management measure provided in these requirements includes a provision primarily to visually inspect the valve body and disc seating surfaces. However, the entire vent valve assembly is typically inspected including the locking devices. In both cases, any observed surface irregularities would be evaluated. In addition, vent valve operation is tested through manual actuation that the vent valve will be fully open for a force less than or equal to 400 lbs. <u>p4-5, add following paragraph at line 3</u>: Cracking of the Alloy 600 locking devices on the modified vent valve locking devices by PWSCC is managed by Examination Category B-N-3. This program will be supplemented by technical specification requirements for ANO-1 and TMI-1 and the Pump and Valve In-Service Test Program for ONS-1, 2, and 3 which require vent valve testing and a visual inspection of the vent valve each refueling outage. <u>p4-5, line 4 modify sentence as follows</u>: ... programmatic controls along with the vent valve testing and inspection requirements and the Reactor Vessel ... <u>p4-7, add following paragraph at line 12</u>: Loss of material of the locking devices on the original vent valve assemblies due to wear is managed by Examination Category B-N-3. This program will be supplemented by technical specification requirements for ANO-1 and TMI-1 and the Pump and Valve In-Service Test Program for ONS-1, 2, and 3 which require vent valve testing and a visual inspection of the vent valve each refueling outage. <u>p4-7, line 13 modify sentence as follows</u>: ... programmatic controls along with the vent valve testing and inspection requirements will manage ...	

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
				p4-16, under programs for the vent valve locking devices (modified) add: Supplemented by: • Plant Technical Specifications (ANO-1, TMI-1) *Vent valve testing and inspection requirements each refueling outage* • Pump and Valve In-Service Test Program (ONS-1,-2,-3) *Vent valve testing and inspection requirements each refueling outage* p4-16, under programs for the vent valve locking devices (original) add: Supplemented by: • Plant Technical Specifications (ANO-1, TMI-1) *Vent valve testing and inspection requirements each refueling outage* • Pump and Valve In-Service Test Program (ONS-1,-2,-3) *Vent valve testing and inspection requirements each refueling outage* p5-3, lines 17 and 18, modify sentences as follows: ... B-N-3 and supplemented by vent valve testing and inspection ...	
7.	Page 3-5 of BAW-2248 states that "the ONS-1 CRGT assembly sectors required straightening after the first hot functional test (FHT)." Provide a summary of the evaluation indicating that the cracking mechanisms (SCC and IASCC) are not plausible in this case. If these guide tube sectors were to be degraded by a cracking mechanism, could such cracking impede the ability of the reactor vessel internals to perform its function to "provide support, orientation, guidance, and protection of the control rod assemblies"?	Although straightening may have introduced residual stresses due to cold work from the straightening process, the reactor coolant environment is not conducive to SCC since halogens are controlled to less than 150 ppb and oxygen is controlled to less than 5 ppb during operation. Further, the 48 EFPY fluence at the guide tube location is below the extrapolated PWR threshold for IASCC of 1×10^{21} n/cm^2	Open	Page 3-5: Delete the last three sentences of the third paragraph, which starts at line 23 with "This cold working may have ...", and which continues through line 27. Replace with: Although straightening may have introduced residual stresses due to cold work from the straightening process, the reactor coolant environment is not conducive to SCC since halogens are controlled to less than 150 ppb and oxygen is controlled to less than 5 ppb during operation. Further, the 48 EFPY fluence at the guide tube location is below the extrapolated PWR threshold for IASCC of 1×10^{21} n/cm^2. Therefore, cracking by either SCC or IASCC is not an applicable aging effect for the CRGT assembly sectors.	

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
8.	Examination Category B-N-3 of ASME Section XI requires a VT-3 visual examination of "accessible surfaces" of removable core support structures. For assemblies and parts determined to be susceptible to no aging mechanisms, and hence to require no additional aging management, the VT-3 examination provides one measure of assurance of the structural integrity of the part. Which components not susceptible to an aging mechanism (and hence, no additional aging management) will receive a VT-3 examination that can serve as a sampling of nonsusceptible components?	The components with no applicable aging effects include the Plenum Cover Assembly and Plenum Cylinder. These items receive visual inspection (VT-3) in accordance with ASME Section XI, Examination Category B-N-3, each inspection interval. However, inspections are not required if no applicable aging effects are identified.	Open	None.	
9.	Section 3.3 of BAW-2248 indicates that crevice corrosion is not expected to be a concern, unless the internals are exposed to a series of long outages that have stagnation and high impurity levels. What impurity levels and how much cumulative outage time are required before crevice corrosion becomes a concern? What components could be affected by crevice corrosion? How could crevice corrosion be prevented if there were a long outage?	Crevice corrosion is considered a severe form of pitting. As with pitting, crevice corrosion will not occur unless dissolved oxygen levels are above 100 ppb and halogens are above 150 ppb. In the presence of an oxygenated environment that contains impurites, the rate at which loss of material by pitting and crevice corrosion occurs is dependent on temperature: the higher the temperature the higher the rate of corrosion. During shutdown, aerated primary coolant can have dissolved oxygen contents of approximately 8 ppm when the reactor vessel head is removed for refueling; however, temperatures are low and impurities are controlled during refueling (per the EPRI Water Chemistry Manual) at levels that will preclude crevice or pitting corrosion. During extended outages the plant may remain in cold shutdown with the RCS in the filled and pressurized condition with halogen and dissolved oxygen controlled to less than 150 ppb and 100 ppb, respectively. Degradation of reactor vessel internals by	Open	<u>Section 3.3, Page 3-9, Line 7</u> Delete the following three sentences. "During shutdown conditions, primary water oxygen and impurity concentrations increase. Thus, crevice corrosion effects may be expected to be more significant during shutdown conditions. Unless the internals are exposed to a series of long outages where stagnancy and high impurity levels remain, crevice corrosion is not expected to be a concern." <u>Replace with the following text</u>: During shutdown, aerated primary coolant can have dissolved oxygen contents of approximately 8 ppm when the reactor vessel head is removed for refueling; however, impurities are controlled during both refueling and extended outages at levels that will preclude crevice or pitting corrosion.	

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
		crevice or pitting corrosion has not been observed as a result of extended outages at Three Mile Island Unit 1, which experienced an extended shutdown period of approximately 5 years. Therefore, crevice corrosion is not a significant degradation mechanism during extended outages owing to the plant-specific Water Chemistry Programs.			
10.	Section 3.3 of BAW-2248 indicates that wear is not a concern for the modified vent valve locking devices. Explain the difference between the original design and the modified design that eliminated the concern for wear of the vent valve locking device. To what criteria are the modified vent valve locking devices being inspected to ensure that they are not subject to wear? Summarize the results of these inspections; include the number and frequency of inspection. Will these inspections be continued into the license renewal term?	Each of the RVI vent valves is held in place by segments (retaining rings) which expand into grooves in the core support shield assembly. The segments are expanded by two jackscrews per vent valve. The jackscrews are torqued and then a locking device is utilized. The original locking devise was a "pop-up" cup that engaged a spline on the jackscrew. The device was spring-loaded and could be depressed while turning the jackscrew. The splines were then aligned and the cup released to pop-up and engage the spline on the jackscrew. Several of the original locking cups were found to be damaged under service conditions. In particular for valves located on either side of the outlet nozzles, the cups were found to be worn. The spring force was not sufficient to stabilize the cup in the high cross flow velocity near the outlet nozzles. The cups were vibrating and rubbing against the jackscrew that was causing the cup to wear severely. The correction action for this defect was to remove the original locking cup assembly and install a crimp cup assembly which utilized a crimp cup welded to a base block. This arrangement was much stiffer and not subject to flow-induced vibration. The vent valves with the new locking devices, just like the vent valves with the old locking devices, are inspected and tested at each refueling outage per technical	Open	None	

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
		specifications requirements for ANO-1 and TMI-1 and the Pump and Valve In-Service Test Program for ONS-1, 2, and 3. (See response to RAI #6). Besides the wear problem in the 1978-1980 time frame, inspections of the vent valves have not found any significant problems. The technical specification and test programs currently in place will be continued into the period of extended operation.			
11.	Section 3.3 of BAW-2248 indicates that Westinghouse has observed wear of incore guide tubes caused by flow induced vibration in regions directly exposed to reactor coolant system (RCS) flow. Wear of Babcock & Wilcox (B&W)-designed guide tube and spiders are not a concern; however, because of differences between the Westinghouse and B&W guide tube design and because the B&W-designed detectors are inserted and withdrawn once per fuel cycle. Explain the difference in design and operation of the Westinghouse and B&W incore guide tubes that indicates wear is not a concern for the B&W design. Are there any limits on the number of insertions and withdrawals of the incore monitors that could lead to a concern about wear of the guide tubes?	The design and operation of the Westinghouse and B&W incore guide tubes are described below. Westinghouse Design As described in NRC Information Notice 87-44, the Westinghouse plants that utilize bottom mounted instrumentation use retractable incore neutron detector cables that reside within retractable thimble tubes. The thimble tubes normally extend from a transfer device (located above the seal table), through the seal table, through the bottom of the reactor vessel, through reactor internals guide tubes, and through selected fuel assembly guide tubes. The thimble tubes are supported and surrounded by high pressure conduit (i.e., tubing) that extends from the seal table to the bottom of the reactor vessel. From the bottom of the reactor vessel the thimble tubes are directed through bottom mounted instrumentation nozzles (which are attached to the bottom of the reactor vessel), through bottom mounted instrumentation columns (which are part of the lower RV internals), and through the fuel assembly guide tubes. The thimble tubes are guided and supported in all regions of travel with the exception of the section between the lower core plate and the entrance to the fuel assembly guide tubes.	Open	None.	

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
		The thimble tubes are sealed at the leading (reactor end) but are open at the transfer device, which is located above the seal table, to allow insertion of an incore neutron detector cable. Mechanical high-pressure seals, located at the seal table, are used to seal the area between the thimble tube and the high-pressure conduit. Therefore, degradation of the thimble tube in the unguided and unsupported section may lead to RCS leakage that would bypass the mechanical high-pressure seal resulting in leakage at or near the transfer device. <u>Westinghouse Operation</u> The retractable Westinghouse incore detectors, which are mounted on a spool, are inserted into the thimble tubes periodically during power operation to obtain flux measurements. The incore detectors are subsequently withdrawn once the flux measurements are obtained. As reported in NRC Information Notice 87-44, loss of material by flow-induced vibration has been observed at the location where the thimble tubes are unguided and unsupported. That is, the region between the lower core plate and the entrance to the fuel assembly guide tube. Degradation of the thimble tube in the unguided and unsupported section may lead to RCS leakage that would bypass the mechanical high-pressure seal resulting in leakage at or near the transfer device. <u>B&W Design</u> The B&W-designed bottom mounted incore instrumentation is significantly different than the Westinghouse design in that a thimble tube is not used and the neutron detector cable is the only retractable item in the design. Specifically, the incore neutron			

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
		detector cables extend from the incore monitoring system (IMS) tank (analogous to seal table), through the incore monitoring system piping (analogous to pressure conduit used in the Westinghouse design), through the bottom of the reactor vessel, through the lower reactor internals instrument guide tubes, and through the fuel assembly instrument tubes. The incore neutron detector cables are supported in all regions of travel with the exception of the short section of cable that extends from the instrument guide tubes to the fuel assembly instrument tubes. The neutron detector cables have been reinforced with a vibratory wear sleeve that extends from just below the entrance to the fuel assembly instrument tube to the IMS nozzles in the bottom of the reactor vessel. The incore neutron detector cables in the B&W design are exposed to RCS fluid and the pressure boundary is defined by the IMS nozzles attached to the lower head of the reactor vessel, IMS piping, and bolted connections within the IMS tank (see BAW-2243A, Figure 2-16). The bolted connections in the IMS tank consist of the neutron detector cabling and the closure housing assemblies, which include a loading ring, nut ring, hydrostatic test plug, and studs associated with each housing assembly. <u>B&W Operation</u> The incore neutron detector cables reside within the fuel assembly guide tubes during power operation and are retracted only during refueling when a fuel assembly is replaced. Prior to removing a fuel assembly the incore detector is retracted just below the bottom of the fuel assembly and reinserted once the fuel assembly is replaced. Incore			

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
		detectors that are inoperable may be replaced with new detectors. At present, there are no limits on the number of insertions and withdrawals of incore monitoring cable that could cause a concern regarding wear of the internal guide tubes. Wear of the incore monitoring system piping due to insertions and withdrawals was determined to be insignificant for the period of extended operation as documented in BAW-2243A, Section 3.12, Page 3-10. In addition, damage to the incore neutron detector cables in the unguided and unsupported region between the instrument guide tubes and fuel assembly guide tubes would not lead to RCS leakage. However, damage in the unsupported and unguided region is unlikely owing to the vibratory wear sleeve that surrounds the incore neutron detector cable.			
12.	During its interaction meetings with the staff (referenced in the Section 4.3.11 of the Oconee Nuclear Station License Renewal - Technical Information New Programs and Activities Report, June 1998), the B&W Owners Group (B&WOG) described current and ongoing reactor internals baffle bolt activities that included preparation for possible augmented baffle bolt inspection during the next 10-year ISI interval at Oconee 1 (2003 at the earliest). Describe baffle bolt inspections that will be conducted prior to the start of the extended license renewal period and indicate how these actions provide the basis for assuring the baffle bolt monitoring and inspection techniques that are planned during the period of extended operation are appropriate.	Current baffle bolt activities include interaction with the EPRI MRP program as described in the cover letter The ITG on reactor internals is currently addressing the issues of cracking, reduction of fracture toughness, and loss of preload as it relates to baffle bolts and associated materials. The data and information acquired from these various ITG program activities will be used to determine the necessary steps in managing the issue of baffle bolt age-related degradation, including future inspections plans. These plans are expected to be outlined on a plant-specific basis, possibly beginning with the inspection at Oconee Unit 1 during their 4th ISI inspection interval (i.e., 2003-2013).	Open	None.	

NRC RAI	NRC Comment	LRTF Response	Status	Proposed Revision	Remarks
13.	Describe the program that will be implemented as outlined in Section 4.6 of BAW-2248 with regard to the aging management of the reactor internals baffle bolts. Describe the overall inspection program, including aspects such as, intervals, monitoring, and inspection techniques.	As provided in the response to RAI # 12, baffle bolt inspection details such as inspection technique, inspection intervals, and monitoring are to be established following the completion of activities of the ITG on reactor internals. The results of ongoing industry inspection activities and analyses will provide the necessary information needed to establish inspection frequency and monitoring details of baffle bolts at the B&W operating plants.	Open	None.	
14.	Describe the replacement bolts and redesigned RVI that are referred to in the fatigue analysis discussed in BAW-2248 Section 4.5.1. Are the replacement bolts and redesigned RVI identified in the fatigue analysis related to the issue of the A-286 bolt cracking discovered in B&W RVI? Are the baffle bolts discussed in the BAW-2248 report included in the fatigue analysis? If not, what is the basis for not including the baffle bolts in the fatigue analysis? If the baffle bolts are included in the analysis, describe how baffle bolt cracking is taken into account and identify the analysis report	The replacement bolts are the core barrel bolts and the thermal shield bolts. These bolts were replaced due to the Alloy A-286 cracking that occurred in the B&W-designed plants in the early 1980's. These bolts are discussed in more detail in Section 3.5.4 of BAW-2248. The fatigue analysis only considered the replacement bolting since they were being designed to ASME Section III, Subsection NG rules. The baffle bolts, along with the other original components, were designed prior to the development of ASME Code requirements for core support structures. See Section 4.5.1 for more details.	Open	None.	

Table 1—Aging Effects/Mechanisms for RV Internals Items		
Reactor Internals Items	**NRC RAI#2 Aging Effect (Degradation Mechanism)** IE– irradiation embrittlement TE– thermal embrittlement SR– stress relaxation	**Bases for concluding that the aging effects are not significant for the specific items**
Plenum cover	Cracking (SCC, IASCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb, and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC (see section 3.1, page 3-4, lines 10-21, and page 3-5, lines 29-40): The fluence through the period of extended operation for the plenum cover has been estimated to be below the PWR IASCC fluence threshold of $1 - 2 \times 10^{21}$ n/cm^2 (>1MeV)
Plenum cylinder	Cracking (SCC, IASCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb, and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC (see section 3.1, page 3-4, lines 10-21, and page 3-5, lines 29-40): The fluence through the period of extended operation for the plenum cylinder has been estimated to be below the PWR IASCC fluence threshold of $1 - 2 \times 10^{21}$ n/cm^2 (>1MeV)
(CRA) guide tubes	Cracking (SCC, IASCC) Loss of Material (wear)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC (see section 3.1, page 3-4, lines 10-21, and page 3-5, lines 29-40): The fluence through the period of extended operation for the (CRA) guide tubes has been estimated to be below the PWR IASCC fluence threshold of $1 - 2 \times 10^{2}$ n/cm^2 (>1MeV) Loss of Material due to Wear: As stated in section 3.3, page 3-9, line 28-30, no significant wear, to date, has been observed in these components and is expected to be insignificant over the lifetime of the reactor vessel internals. Therefore loss of mated by wear is not considered a significant aging effect for the CRA guide tubes. Loss of Fracture toughness due to thermal embrittlement: (see page 3.7, lines 15-31): Thermal aging is not an active aging

Table 1—Aging Effects/Mechanisms for RV Internals Items		
Reactor Internals Items	**NRC RAI#2 Aging Effect (Degradation Mechanism)** IE– irradiation embrittlement TE– thermal embrittlement SR– stress relaxation	**Bases for concluding that the aging effects are not significant for the specific items**
	Loss of Fracture Toughness (TE)	mechanism in the CRGT because all components except the CRGT spacer casting are fabricated with wrought austenitic stainless steel. Wrought austenitic stainless steels do not exhibit thermal aging. The CRGT spacer casting is fabricated with low carbon CF-3M that demonstrates only limited thermal aging, and is a non-load bearing component.
CRA guide tube bolts	Cracking (SCC, IASCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb, and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC (see section 3.1, page 3-4, lines 10-21, and page 3-5, lines 29-40): The fluence through the period of extended operation for the CRA guide tube bolts has been estimated to be below the PWR IASCC fluence threshold of 1 – 2 $x10^{21}$ n/cm^2 (>1MeV)
Upper grid assembly	Cracking (SCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb, and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC.
Upper grid rib section	Cracking (SCC, IASCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb, and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC: The upper grid rib section is defined to be part of the upper grid assembly (see section 2.1.3, page 2-6, and table 2-1) and is therefore included in the aging management plan during the period of extended operation as outlined on page 3-7, line 7.
Upper grid assembly bolts	Cracking (SCC, IASCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb, and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC: The upper grid assembly bolts are defined to be part of the upper grid assembly (see section 2.1.3, page 2-6, and table 2-1) and is therefore included in the aging management plan during the period of extended operation as outlined on page 3-7, line 7.

Table 1—Aging Effects/Mechanisms for RV Internals Items		
Reactor Internals Items	**NRC RAI#2** **Aging Effect** **(Degradation Mechanism)** IE– irradiation embrittlement TE– thermal embrittlement SR– stress relaxation	**Bases for concluding that the aging effects are not significant for the specific items**
Upper internals fuel guide pads	Cracking (SCC, IASCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb, and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC: The upper internals fuel guide pads are defined to be part of the upper grid assembly (see section 2.1.3, page 2-6, and table 2-1) and is therefore included in the aging management plan during the period of extended operation as outlined on page 3-7, line 7.
Core support shield	Cracking (SCC, IASCC) Loss of Fracture Toughness (IE)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5pp and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC (see section 3.1, page 3-4, lines 10-21, and page 3-5, lines 29-40): The fluence through the period of extended operation for the core support shield has been estimated to be below the PWR IASCC fluence threshold of $1 - 2 \times 10^{2}$ n/cm^2 (>1MeV) (see section 3.2, page 3-7, lines 33-45): Irradiation embrittlement becomes noticeable at a fluence of approximately 5×10^{20} n/cm^2 (E>1 MeV). The Core support shield is not anticipated to reach the threshold of 5×10^{20} n/cm^2 during the period of extended operation. The expected changes in mechanical properties as a result of neutron exposure are insignificant and therefore reduction of fracture toughness by irradiation embrittlement is not an applicable aging effect.
Core support flange	Cracking (SCC, IASCC) Loss of Fracture Toughness (IE)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5pp and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC (see section 3.1, page 3-4, lines 10-21, and page 3-5, lines 29-40): The fluence through the period of extended operation for the core support flange has been estimated to be below the PWR IASCC fluence threshold of $1 - 2 \times 10^{2}$ n/cm^2 (>1MeV) (see section 3.2, page 3-7, lines 33-45): Irradiation embrittlement becomes noticeable at a fluence of approximately 5×10^{20} n/cm^2 (E>1 MeV). The Core support flange is not anticipated to reach the threshold of 5×10^{20} n/cm^2 during the period of extended operation. The expected changes in mechanical properties as a result of neutron exposure are insignificant and therefore reduction of fracture toughness by irradiation embrittlement is not an applicable aging effect.

Table 1—Aging Effects/Mechanisms for RV Internals Items		
Reactor Internals Items	**NRC RAI#2 Aging Effect (Degradation Mechanism)** IE– irradiation embrittlement TE– thermal embrittlement SR– stress relaxation	**Bases for concluding that the aging effects are not significant for the specific items**
Vent valve assemblies	Cracking (IASCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC (see section 3.1, page 3-4, lines 10-21, and page 3-5, lines 29-40): The fluence through the period of extended operation for the plenum cover has been estimated to be below the PWR IASCC fluence threshold of $1 - 2 \times 10^{21}$ n/cm^2 (>1MeV)
	Loss of Fracture Toughness (IE)	(see section 3.2, page 3-7, lines 33-45): Irradiation embrittlement becomes noticeable at a fluence of approximately 5×10^{20} n/cm^2 (E>1 MeV). The Vent valve assembly is not anticipated to reach the threshold of 5×10^{20} n/cm^2 during the period of extended operation. The expected changes in mechanical properties as a result of neutron exposure are insignificant and therefore reduction of fracture toughness by irradiation embrittlement is not an applicable aging effect
Core barrel assembly	Cracking (SCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb, and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC.
Baffle and former plates	Cracking (SCC, IASCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC: The baffle and former plates are defined as part of the core barrel assembly (see section 2.3, page 2-11, lines 27-29) and are included in the aging management plan described in Section 4.6.
	Loss of Fracture Toughness (IE)	The baffle and former plates are defined as part of the core barrel assembly (see section 2.3, page 2-11, lines 27-29) and are included in the aging management plan described in Section 4.6.
	Loss of Closure Integrity (creep)	See section 3.0, page 3–1, lines 41-45: Thermal creep has been ruled out as an aging mechanism for reactor internal components because the temperature of the internals, including the baffle and former plates, is far below the threshold for creep of austenitic stainless steels. It has been stated that the only plausible time-dependent deformation process, which may occur during the period of extended operation, would be stress relaxation of pre-stressed bolting.

Table 1—Aging Effects/Mechanisms for RV Internals Items

Reactor Internals Items	NRC RAI#2 Aging Effect (Degradation Mechanism) IE– irradiation embrittlement TE– thermal embrittlement SR– stress relaxation	Bases for concluding that the aging effects are not significant for the specific items
Baffle-former bolts	Cracking (SCC) Loss of Closure Integrity (SR)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. The baffle bolts will be included in the aging management for stress relaxation. **<u>PROPOSED CHANGES</u>**: <u>Section 3.4, page 3-11 (line 5), add the following</u>: Baffle and former bolts <u>Section 3.4, page 3-11, (line 29), add the following</u>: 8) Loss of closure integrity of the baffle and former bolts due to stress relaxation. <u>Section 4.4, page 4-7, add the following</u>: 8) Loss of closure integrity of the baffle and former bolts due to stress relaxation. <u>Table 4-1, page 4-17</u>, under "Baffle-to-Baffle Bolts", add "stress relaxation" in the column heading "Aging effect" as a separate effect, identified with the identical Management Program as for Cracking and Reduction of Fracture Toughness. <u>Table 4-1, page 4-18</u>, under "Baffle-to-Former Bolts", add "stress relaxation" in the column heading "Aging effect" as a separate effect, identified with the identical Management Program as for Cracking and Reduction of Fracture Toughness
Lower grid top rib section	Cracking (SCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb, and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC.
Lower grid bottom rib weldment	Cracking (SCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb, and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC.

Table 1—Aging Effects/Mechanisms for RV Internals Items

Reactor Internals Items	NRC RAI#2 Aging Effect (Degradation Mechanism) IE– irradiation embrittlement TE– thermal embrittlement SR– stress relaxation	Bases for concluding that the aging effects are not significant for the specific items
Lower grid assembly support posts	Cracking (SCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb, and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC.
Lower internals fuel guide pads	Cracking (SCC, IASCC) Loss of fracture toughness (IE)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5ppb and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC: The lower internals fuel guide pads are defined to be part of the lower grid assembly (see section 2.4.1, page 2-13 through 2-15, and table 2-1) and is therefore included in the aging management plan during the period of extended operation as outlined on page 3-7, line 8. The lower internals fuel guide pads are defined to be part of the lower grid assembly (see section 2.4.1, page 2-13 through 2-15,1 and table 2-1) and is therefore included in the aging management plan to guard against the loss of fracture toughness during the period of extended operation as outlined on page 3-7, line 8.
Lower grid assembly bolts	Cracking (SCC, IASCC) Loss of fracture toughness (IE)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5ppb and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC: lower grid assembly bolts are defined to be part of the lower grid assembly (see section 2.4.1, page 2-13 through 2-15, and table 2-1) and is therefore included in the aging management plan during the period of extended operation as outlined on page 3-7, line 8. The lower grid assembly bolts are defined to be part of the lower grid assembly (see section 2.4.1, page 2-13 through 2-15, and table 2-1) and is therefore included in the aging management plan to guard against the loss of fracture toughness during the period of extended operation as outlined on page 3-7, line 8.
Lower grid assembly guide blocks	Cracking (SCC, IASCC)	Cracking due to SCC (see section 3.1, page 3-4, lines 30-43): Reactor coolant dissolved oxygen is controlled to less than 5 ppb, and halides are controlled to < 150 ppb, thereby eliminating the environmental conditions which promotes SCC. Cracking due to IASCC: The lower grid assembly guide blocks are defined to be part of the lower grid assembly (see section

Table 1—Aging Effects/Mechanisms for RV Internals Items		
Reactor Internals Items	**NRC RAI#2** **Aging Effect** **(Degradation Mechanism)** IE– irradiation embrittlement TE– thermal embrittlement SR– stress relaxation	**Bases for concluding that the aging effects are not significant for the specific items**
	Loss of fracture toughness (IE)	2.4.1, page 2-13 through 2-15, and table 2-1) and is therefore included in the aging management plan during the period of extended operation as outlined on page 3-7, line 8. The lower grid assembly guide blocks are defined to be part of the lower grid assembly (see section 2.4.1, page 2-13 through 2-15, and table 2-1) and is therefore included in the aging management plan to guard against the loss of fracture toughness during the period of extended operation as outlined on page 3-7, line 8.

APPENDIX D

Appendix D contains the NRC Draft Safety Evaluation
dated May 26, 1999

May 26, 1999

Mr. William R. Gray
Project Manager
Framatome Technologies, Incorporated
P. O. Box 10935
Lynchburg, VA 24506-0935

SUBJECT: DRAFT SAFETY EVALUATION CONCERNING THE BABCOCK & WILCOX OWNERS' GROUP (B&WOG) GENERIC LICENSE RENEWAL PROGRAM TOPICAL REPORT ENTITLED, "DEMONSTRATION OF THE MANAGEMENT OF AGING EFFECTS FOR THE REACTOR VESSEL INTERNALS," BAW-2248, JULY 1997

Dear Mr. Gray:

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

Resolution of the open items in the DSE and satisfactory completion of the identified applicant action items will allow the staff to find that a B&WOG GLRP member plant that references the report in a license renewal application has satisfied the requirements of 10 CFR 54.21(a)(3) and (c)(1) for the reactor vessel internals within the scope of BAW-2248.

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,

*/**Signed**/.*

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

Project No. 683

Enclosure: DSE

cc w/encl: See next page

Project No. 683

Mr. Rick Edwards
Framatone Technologies
1700 Rockville Pike
Suite 525
Rockville, Maryland 20852

Mr. Michael Laggart
Manager, Corporate Licensing
GPU Nuclear Corporation
One Upper Pond Road
Parsippany, New Jersey 07054

Chairman
Board of County Commissioners
of Dauphin County
Dauphin County Courthouse
Harrisburg, Pennsylvania 17120

Mr. J. W. Hampton
Nuclear Generation Vice President
Duke Energy Corporation
Oconee Nuclear Station
MC: ONO 1VP
P. O. Box 1439
Seneca, South Carolina 29670

Mr. John R. McGaha, Vice President
Operations Support Licensing
Entergy Operations, Inc.
P. O. Box 31995
Jacksonville, Mississippi 39286

Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303

Robert L. Gill, Jr.
Duke Energy Corporation
Mail Stop EC-12R
P. O. Box 1006
Charlotte, North Carolina 28201-1006
RLGILL@DUKE-ENERGY.COM

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000

Babcock & Wilcox Owners Group
Generic License Renewal Program
Arlington, Texas 76011

Mr. James J. Fisicaro
Director, Licensing
Entergy Operations, Inc.
Route 3, Box 137G
Russelville, Arkansas 72801

Earnest L. Blake, Jr., Esq
Shaw, Pitman, Potts
and Trowbridge
2300 N. Street, N.W.
Washington, DC 20037

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

B. Gutherman, Manager
Florida Power Corporation, (GA2A)
Crystal River Energy Complex
15760 W. Power Street
Crystal River, Florida 34428-6708

Mr. William Dornsife, Acting Director
Bureau of Radiation Protection
Pennsylvania Department of
Environmental Resources
P. O. Box 2063
Harrisburg, Pennsylvania 17120

Chairman
Board of Supervisors
of Londonderry Township
R.D. #1 Geyes Church Road
Middletown, Pennsylvania 17057

Mr. J. E. Burchfield
Compliance Manager
Duke Energy Corporation
Oconee Nuclear Site
P. O. Box 1439
Seneca, South Carolina 29679

Douglas J. Walters
Nuclear Energy Institute

1776 I Street, NW
Suite 400
Washington, DC 20006-3708
DJW@NEI.ORG

DRAFT SAFETY EVALUATION
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
CONCERNING
"DEMONSTRATION OF THE MANAGEMENT OF AGING EFFECTS FOR THE REACTOR VESSEL INTERNALS"

BABCOCK & WILCOX OWNERS GROUP REPORT NO. BAW-2248
PROJECT NO. 683

1.0 INTRODUCTION

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

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

1.1 Babcock & Wilcox Owners Group Topical Report

By letter dated July 29, 1997, the Babcock & Wilcox Owners Group (B&WOG) Generic License Renewal Program (GLRP) submitted topical report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (Reference 2), for staff review and approval. The purpose of the topical report is to provide a technical evaluation of the effects of aging of the reactor vessel internals and demonstrate that the aging effects within the scope of the report are adequately managed for the period of extended operation associated with license renewal. The topical report provides 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.

ATTACHMENT

1.2 Conduct of Staff Review

The staff reviewed the B&WOG topical report to determine whether it satisfied the requirements set forth in 10 CFR 54.21(a)(3) and (c)(1). The staff issued requests for additional information (RAIs) after completing the initial review. The B&WOG responded to the staff's RAIs. Requests for additional information, meeting summaries, and other correspondence are listed in Appendix A.

2.0 SUMMARY OF TOPICAL REPORT

The B&WOG topical report, BAW-2248, contains a technical evaluation of aging effects related to B&W reactor vessel internals components, and was provided to the staff to demonstrate that B&WOG member plant owners can adequately manage these effects of aging during the period of extended operation. This evaluation applies to the following B&WOG GLRP member plants:

* Arkansas Nuclear One, Unit 1 (ANO-1)
* Oconee Nuclear Station, Units 1, 2, and 3 (ONS-1,-2,-3)
* Three Mile Island, Unit 1 (TMI-1)

The topical report also contains evaluations of TLAAs, as defined in 10 CFR 54.3, for the reactor vessel internals. However, the topical report indicates that the TLAA of flaw growth acceptance prescribed in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI inservice inspection (ISI) program (Reference 3) is plant-specific, is not within the scope of the report, and will be resolved on a plant-specific basis.

In addition, the report indicates that inservice inspection programs identified in the ASME Code Section XI may need to be supplemented because certain components are not easily accessible using current technology.

2.1 <u>Components and Intended Functions</u>

2.1.1 <u>Intended Functions</u>

In Section 1.0 of the topical report, the following five intended functions for the Reactor Vessel Internals (RVI), and system components were identified based on the requirements of 10 CFR 54.4(a):

- Provide support and orientation of the reactor core (i.e., the fuel assemblies).
- Provide support, orientation, guidance and protection of the control rod assemblies.
- Provide a passageway for the distribution of the reactor coolant flow to the reactor core.
- Provide a passageway for support, guidance, and protection for the incore instrumentation.
- Provide a secondary core support for limiting the downward displacement of the core support structure in the event of a postulated failure of the core barrel.

The portions of the RVI that were identified in the topical report (BAW-2248) as not within the scope of this topical report, are discussed in Section 2.5 of the topical report. These items are: thermal shield, thermal shield upper restraint assemblies, and upper thermocouple guide tube assemblies. The staff believes that there is an additional intended function of the RVI as discussed in Section 3.1 of the safety evaluation.

2.1.2 Components

As described in the topical report, the RVI scope consists of two major structural sub-assemblies that are located within, but not integrally attached to (i.e., not welded to) the reactor pressure vessel (RPV). These major sub-assemblies are the plenum assembly (PA) and the core support assembly (CSA). For the purpose of defining materials, fasteners, construction, and assembly, the CSA can be further divided into three principal sub-assemblies; the core support shield assembly (CSS), the core barrel assembly (CBA), and the lower internals assembly (LIA). The mechanical fasteners (bolting) joining these sub-assemblies and associated items are within the scope of this topical report. The welds within the scope of the reactor vessel internals report include the major structural welds that form or join the major sub-assembly cylinders and flanges and minor structural welds joining parts such as lifting lugs, support pipes and tubes to the major sub-assemblies. There are no pressure-retaining or pressure boundary welds within the scope of this topical report.

The control rod assemblies (CRA), fuel assemblies (FA), and the incore monitors (IMS) are not considered part of the RVI and are not covered in the topical report. The thermal shield and upper thermocouple guide tube assemblies are RVI items; however, it is concluded in the topical report that they do not perform intended functions as defined in 10 CFR Part 54 and are, therefore, not within the scope of the topical report. The staff disagrees with the conclusion that the thermal shield is not within the scope of Part 54 (see Section 3.1of this safety evaluation). Portions of the internal vent valve assemblies are active components that do not require an aging management review under 10 CFR 54.21. The surveillance specimen holder tube assemblies (SSHT) are not part of the RVI for the plants included in the report. As such, the SSHT assemblies are not within the scope of the topical report. Physical and functional descriptions of the individual items within each of the four principle sub-assemblies are presented in Sections 2.1 through 2.4 of the topical report.

2.2 Effects of Aging

Section 3.0 of the topical report discusses the aging effects applicable to the reactor vessel internals described above for the period of extended operation for the participating B&W plants. The topical report states that the following effects of aging could result in adverse impact or loss of any of the reactor vessel internals intended functions :

* cracking (initiation and growth)
* loss of material
* reduction of fracture toughness
* loss of mechanical closure integrity (for bolted connections)

Table 3-2 of the topical report provides a detailed list of the subassemblies in each of the reactor vessel internal assemblies, and identifies the aging effect applicable to each subassembly, as determined by the B&WOG's evaluations. These evaluations included a review of industry operating experience to identify past incidents of aging effects applicable to the reactor vessel internals. This review is discussed in Section 3.5 of the topical report.

The following is a summary of Table 3-2 of the topical report:

Major RVI Assemblies	Applicable Aging Effects
Plenum Assembly	Cracking Loss of material Reduction of fracture toughness Loss of closure integrity
Core Support Shield Assembly	Cracking Loss of material Reduction of fracture toughness Loss of closure integrity
Core Barrel Assembly	Cracking Reduction of fracture toughness Loss of closure integrity
Lower Internals Assembly	Cracking Loss of material Reduction of fracture toughness Loss of closure integrity

2.3 Aging Management Programs

Section 4.0 of the topical report discusses the B&WOG bases for demonstrating that the applicable aging effects identified in Section 3.0 of the topical report can be managed by existing programs at ANO-1, ONS-1,-2,-3, and TMI-1 during the periods of extended operation of those plants. Table 4-1 in the topical report provides a detailed summary of the existing programs that manage aging effects that are applicable to each subassembly of the four major reactor vessel internals assemblies identified above. These programs are the following:

* ASME B&PV Code, Section XI, Inservice Inspection Program
* Reactor Vessel Internals Aging Management Program (RVIAMP)
* Plant Technical Specifications for Vent Valve Bodies in Core Support Shield Assemblies in ANO-1, and TMI-1
* Pump and Valve In-Service Test Programs for Vent Valve Bodies in Core Support Shield Assemblies of ONS-1,-2,-3

The topical report proposes that the RVIAMP supplement the ASME Section XI ISI program, since the report concludes that the inspection program required by Examination Category B-N-3 of the ASME Section XI program, subsection IWB, may not be adequate to detect aging effects for certain reactor vessel internal components. The RVIAMP addresses the specific aging effects of SCC, IASCC, (neutron) irradiation embrittlement and stress relaxation.

2.4 Time-Limited Aging Analyses

Section 4.5 of the topical report identifies the following TLAAs that are applicable to the reactor vessel internals, and presents the B&WOG's proposed aging management programs for each TLAA:

* Fatigue - Cracking (Initiation and Growth)
* Ductility - Reduction of Fracture Toughness

However, the topical report indicates that the TLAA of flaw growth acceptance in accordance with the ASME Section XI ISI program (Reference 3) is plant-specific, is not within the scope of the report, and will be resolved on a plant-specific basis.

In addition, the report indicates that inservice inspection programs identified in the ASME Code Section XI may need to be supplemented because certain components are not easily accessible using current technology.

3.0 STAFF EVALUATION

The staff reviewed the topical report and additional information submitted by the B&WOG to determine if they demonstrated that the effects of aging of the reactor vessel internal components covered by the report will be adequately managed so that there is reasonable assurance that the components will perform their intended functions consistent with the CLB for the period of extended operation, in accordance with 10 CFR 54.21(a)(3). This is the last step in the IPA described in 10 CFR 54.21(a).

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

3.1 Intended Functions

The staff reviewed Sections 1.0 and 2.0 of the subject topical report to determine whether there is reasonable assurance that the RVI components and supporting structures subject to AMR have been identified in accordance with the requirements of 10 CFR 54.21(a)(1). This was accomplished as described below.

As part of the evaluation, the staff determined whether the applicant has properly identified the systems, structures, and components within the scope of license renewal, pursuant to 10 CFR 54.4. The staff reviewed portions of the Updated Final Safety Analysis Reports (UFSARs) on the RVI for the applicable operating plants, and compared the information in the UFSARs with the information in the report to identify any portions of the RVI that the report did not identify as within the scope of license renewal. The staff then reviewed the structures and components not identified as within the scope of Part 54, and as described below, and requested the B&WOG to provide additional information and/or clarifications for a selected number of structures and components to verify that they do not have any intended functions as delineated in 10 CFR 54.4(a). The staff also reviewed the UFSARs for any safety-related system functions that were not identified as intended functions in the report to verify that structures and components having intended functions were not omitted from consideration.

After completing the initial review, by letter dated December 2, 1998, the staff issued RAIs regarding the RVI, and by letter dated February 18, 1999, the B&WOG provided responses to those RAIs. Section 1.4 of the topical report identifies the intended functions of the RVI. It does not include, "provide shielding for the RPV" as one of the intended functions. As a result of this omission of an RVI intended function, the components that support this intended function, namely, the thermal shield and the thermal shield upper restraint assemblies were omitted from the scope of license renewal, and were not identified as requiring AMR. NRC RAI # 1 pointed out this omission, and requested clarification. In response, the B&WOG agreed that the function "provide shielding for the RPV" was not included within the scope of the report. Therefore, the B&WOG recommended in its response that each license renewal applicant that references BAW-2248 must determine if the function "provide shielding for the RPV" is an RVI intended function. If not an intended function, the license renewal applicant should provide justification for that conclusion. Should a license renewal applicant determine that the function "provide shielding for the RPV" is an intended function of the RVI, then the items that support this intended function, such as the thermal shield and the thermal shield upper restraint assemblies, must be identified and reviewed in accordance with 10 CFR 54.21(a)(3). This is Renewal Applicant Action Item 3. The B&WOG also indicated in its response that BAW-2248 will be revised to provide this clarification. Based on the supporting information in the UFSARs, and the B&WOG's response to the staff's request for additional information, the staff has found no additional omissions in the report and, therefore, concludes that there is reasonable assurance that the report adequately identified those portions of the RVI and its associated (supporting) structures and components that fall within the scope of license renewal in accordance with 10 CFR 54.4.

As discussed above, the staff has reviewed the information provided in Sections 1.0 and 2.0 of the subject topical report (BAW-2248) and the additional information provided by the B&WOG in response to the staff's RAIs. Based on that review, the staff concluded that, except for the omission as discussed in the above paragraph, there is reasonable assurance that the topical report has appropriately identified the portions of the RVI and the associated structures and components thereof, that are within the scope of license renewal and are subject to AMR in accordance with the requirements of 10 CFR Part 54.

3.2 Effects of Aging

As discussed in Section 2.2 above, the effects of aging evaluated in BAW-2248 included reduction of fracture toughness, cracking, loss of material, and loss of mechanical integrity (for bolted connections). The B&WOG reviewed these aging effects for their applicability to the RVI assemblies within the scope of the report. The B&WOG reviewed RV internals service history of cracking of weld locations (due to mechanical failure, fatigue and other causes), loss of material (external wall thinning), and loss of closure integrity (wear and erosion). The B&WOG findings about these effects were incorporated into the aging management program.

3.2.1 Cracking

Stress Corrosion Cracking (SCC) and Irradiation-Assisted Stress Corrosion Cracking (IASCC)

The topical report identifies cracking as a potential aging effect due to either SCC or IASCC.

SCC results from the synergistic action of a susceptible material subjected to tensile stresses in a corrosive environment, which is specific to the material. The material may be inherently susceptible, or can become sensitized during fabrication. The tensile stresses can be due to the operational loading or residual fabrication stresses. The environmental parameters considered to be critical in SCC are the dissolved oxygen, halide and sulfide contents in the coolant. IASCC is a mechanism in which the presence of the neutron irradiation can make the material more susceptible to SCC.

For SCC, the report uses reactor coolant chemistry control, in particular dissolved oxygen less than 5 ppb and halides less than 150 ppb, as the basis for generally ruling out SCC as potentially significant. The staff believes that RVI components will not be susceptible to SCC with coolant containing these dissolved elements because at these low values the coolant corrosive environment is sufficiently low to preclude SCC. For RVI bolting applications, the topical report indicates that SCC is a potential aging effect due to the potential for occluded environment conditions in the crevice area typically associated with bolting. The specific applications cited as potentially susceptible to SCC are: core support shield to core barrel bolts, lower internal assembly to core barrel and thermal shield bolts, core barrel to thermal shield bolts, shell forging to flow distributor bolts, and Alloy 600 locking devices on the modified vent valve assembly.

For IASCC, the report uses a neutron fluence threshold of 1-2 x 10^{21} n/cm^2 (E>1 MeV) to determine susceptibility to IASCC. The NRC staff has reservations concerning this threshold fluence approach, and has proposed an aging management program which obviates the need for a threshold fluence consideration. RVI components determined to be susceptible to IASCC are: core barrel assembly base metal and welds, baffle to baffle and former bolts, core support shield to core barrel bolts, lower internals assembly to core barrel bolts, and upper and lower grid assembly base metals and welds. Of these components, baffle-former and baffle-baffle bolts are expected to be the first to exhibit indication of IASCC because they are nearest the core and have cracked in PWR plants.

Baffle Former Bolt Cracking

The technical evaluation in the BAW-2248 report, Section 3.5 included a review of the historical performance of the reactor vessel internals (RVI) to identify and assess past incidents of aging effects applicable to RVI. The assessment included a review of information in the nuclear plant reliability data system (NPRDS), Licensee Event Reports (LERs) and NRC Generic Letters (GL), Information Notices (IN) and Bulletins. In the BAW-2248 report, B&WOG identified NRC IN 91-05 as providing information regarding cracking in Alloy A-286 bolts used in reactor coolant pumps and the B&W-designed RVI. However, the RVI historical performance review does not include the aging effects applicable to RVI baffle bolting described in the more recently issued NRC IN 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants," issued on March 25, 1998.

3.2.2 Loss of Material

Loss of Material Due to Wear

In Section 3.3 of the topical report, B&WOG identifies the RVI items that are subject to loss of material due to wear. The items identified include the fuel assembly support pads on the upper and lower grid assemblies, the plenum rib pads, the guide blocks on the lower grid, the top flange on the core support cylinder, and the locking devices on the original vent valve assembly. Wear occurs as a result of movement at mating surfaces that may result from flow-induced vibration during plant operation and differential thermal expansion and contraction movements during plant heat-up, cool-down and changes in power operating cycles. The resulting relative movement between the interfacing and mating surfaces causes wear. The severity of the wear depends upon the frequency, duration of the motion and the loads imposed on the affected surfaces. The RVI items identified as subject to wear are typical of the RVI construction items found in locations of structural interfaces and mating surfaces that experience relative motion during plant operation. The identified RVI items subject to wear require programmatic aging management.

Loss of Material Due to Corrosion

The topical report cites three possible mechanisms for loss of material due to corrosion. These mechanisms are: (1) erosion and erosion-corrosion, (2) uniform attack/general corrosion, and, (3) pitting and crevice corrosion.

Erosion and erosion-corrosion are not considered to be applicable since all of the RVI components are fabricated from stainless steel or nickel-base alloys, and these materials have been found to be resistant to erosion and erosion-corrosion.

Uniform attack/general corrosion are not considered to be applicable since all of the RVI components are fabricated from stainless steel or nickel-base alloys, and these materials have been found to be resistant to general corrosion due to protective passivation layers which mitigate the susceptibility of these materials.

Pitting and crevice corrosion are not considered to be applicable due to the low oxygen levels in the reactor coolant as a result of the water chemistry controls.

Based on the RVI materials and reactor coolant environment, the staff concludes that loss of material due to corrosion is not considered to be an applicable aging effect for any of the RVI components.

3.2.3 Reduction of Fracture Toughness

The topical report identifies reduction of fracture toughness in RVI components as an applicable aging effect due to either thermal embrittlement or neutron irradiation embrittlement. Thermal embrittlement can occur in cast austenitic stainless steel (CASS) and martensitic stainless steel parts exposed to high temperatures typical of reactor operating conditions. Neutron irradiation embrittlement occurs in all steels due to exposure to high neutron flux conditions typical of many RVI components. Both of these mechanisms result in increased hardness and tensile strength, along with reduced ductility, impact strength, and fracture toughness of the material. For RVI components fabricated from CASS and hence subject to thermal embrittlement, concurrent exposure to high neutron fluence levels can result in a synergistic effect wherein the service-degraded fracture toughness is reduced from the levels predicted independently for either of the mechanisms. Therefore, components determined to be subject to thermal embrittlement require an additional consideration of the neutron fluence of the component to determine the full range of degradation mechanisms applicable for the component.

Thermal embrittlement was determined to be applicable to internal vent valve bodies, vent valve retaining rings, control rod guide tube (CRGT) and incore guide tube spiders, and the core support shield outlet nozzles at Oconee Nuclear Station Unit 3. All of these components are fabricated from CASS, except for the vent valve retaining rings, which are composed of precipitation-hardened stainless steel.

Determination of RVI components subject to neutron irradiation embrittlement was handled in the topical report using a fluence threshold to screen-out components with a neutron fluence level below 5×10^{20} n/cm^2 (E>1 MeV). Components found to be subject to neutron irradiation embrittlement include the upper grid assembly, core support shield to core barrel bolts, core barrel assembly, baffle-baffle and baffle-former bolts, lower internal assembly to core barrel bolts, and the lower grid assembly. The NRC staff has reservations concerning this threshold fluence approach: however, the proposed aging management program would obviate the need for a threshold fluence consideration (See Section 3.3 of this safety evaluation).

3.2.4 Loss of Closure Integrity for Bolted Closures

In Section 3.4 of the topical report, the B&WOG indicates that bolting stress relaxation is considered an applicable aging mechanism for those components where maintaining a preload is important to the structural integrity function(s) of the RVI. These RVI bolts include: the control rod guide tube (CRGT) to upper grid fasteners; the core support shield to core barrel bolts; the core barrel to thermal shield bolts; lower internals assembly to core barrel bolts; the lower grid rib-to-shell fasteners; the shell forging to flow distributor bolts; and the lower internals assembly to thermal shield bolts.

3.2.5 Change of Dimension

Section 3.1 of the topical report dismisses change of dimension of the RVI components due to void swelling as a significant aging effect due to the lack of evidence of void swelling under PWR conditions. However, EPRI TR-107521 (Reference 4) cites several sources with conflicting results. One source predicts swelling as great as 14% for PWR baffle-former assemblies over a 40-year plant lifetime, whereas results from another source indicate that swelling would be less than 3% for the most highly irradiated sections of the internals at 60 years. The issue of concern to the staff is the impact of change of dimension due to void swelling on the ability of the RVI to perform their intended function. The B&WOG should identify; (1) How much of a change in dimension would be required before the internals would not be able to meet their intended function; (2) What ongoing programs, if any will evaluate the impact of the void swelling on the intended function of the internals; and (3) When these programs will provide data to determine whether void swelling could impact the intended function of the internals.

Should it be determined that change of dimension by void swelling can impede the ability of the RVI to perform their intended functions, then an appropriate aging management program would be required to assure that the need for corrective actions can be properly identified. The determination of the need for an aging management program for changes in dimension is Topical Report Open Item 1.

3.2.6 Summary

With the exception of changes in dimension due to void swelling, the staff agrees with the B&WOG identification of applicable RV component aging effects that are subject to aging management as a condition of license renewal. The staff finds that the B&WOG should address the aging effect of change of dimension due to the aging mechanism of void swelling, as described in Topical Report Open Item 1.

3.3 Aging Management Programs

As described in Section 2.3, the aging management programs discussed by the B&WOG include the RVIAMP, ASME Section XI requirements, plant technical specifications, plant-specific test programs and licensee commitments in response to NRC generic communications. Applicants for license renewal will be responsible for describing any such commitments and identifying the appropriate regulatory control.

The principal change to aging management programs is the addition of the RVIAMP. The generic UFSAR supplement provided as Appendix A to the topical report BAW-2248 states that an applicant will continue to investigate the potential aging effects identified in BAW-2248, through the RVIAMP, and will establish appropriate monitoring and inspection programs prior to the expiration of the current license. In addition, an applicant would be required to provide annual written status reports to the NRC on the RVIAMP beginning one year after issuance of the renewed license.

3.3.1 Cracking

Stress Corrosion Cracking (SCC) and Irradiation-Assisted Stress Corrosion Cracking (IASCC)

Management of SCC and IASCC is achieved through two means. The first means for management of cracking is through the existing inservice inspection (ISI) program, which requires visual VT-3 examination in accordance with Examination Category B-N-3 of Section XI of the ASME Code. However, the topical report indicated that this visual examination may not be adequate to detect cracking for all of the susceptible RVI components due to accessibility concerns, except for the modified vent valve assembly. This concern is addressed through the second means to manage cracking, which is a planned Reactor Vessel Internals Aging Management Program (RVIAMP). The purposes of this program are to continue the investigation of the potential aging effects that have been identified in the topical report for the RVI, and to establish appropriate monitoring and inspection programs that will continue to maintain the RVI in a functional state during the period of extended operation. Further, in response to NRC RAI #3, the B&WOG indicated that an industry group, the PWR Materials Reliability Project (MRP) is addressing neutron embrittlement of RVI components under the auspices of an RPV Internals Issue Task Group. The RAI response stated that the results of this MRP activity will be incorporated in the development of the RVIAMP.

The NRC staff proposed to the B&WOG a modified approach to manage cracking of RVI components. In particular, a two-pronged approach was proposed. The two pronged approach is to perform inspection and to perform tests and analysis of irradiated material. The inspection part of the approach is a supplemental (enhanced VT-1) examination of the components believed to be the limiting components for cracking, considering both the susceptibility of the component to the aging mechanism as well as the material properties (in particular the fracture toughness) and the operating stresses on the component. These examinations would be included as part of the 10-year ISI program. This supplemental examination (enhanced VT-1) applies to all RVI components except for bolting. Aging management for the limiting RVI bolting, baffle former bolting, is described below.

Since the examination addresses the limiting components, plant-specific neutron fluence evaluations are not necessary. Initial consideration by the B&WOG indicated that the limiting components with respect to highest neutron fluence were the baffle plates and baffle former bolts. The B&WOG must identify the limiting components and modify the ISI program for cracking in the topical report accordingly. This is Topical Report Open Item 2.

The second part of this approach would determine the need for continuing the supplemental examination. Should data or evaluations from the MRP or any other industry research activity indicate that the supplemental (enhanced VT-1) examinations can be modified or possibly eliminated, each applicant would be required to provide plant-specific justification to demonstrate the basis for the modification or elimination.

Baffle Former Bolt Cracking

In BAW-2248, Section 3.1, and Table 3-2, the B&WOG identifies baffle former bolting as a component subject to aging effects. Section 4.1 of the report describes the demonstration of aging management for cracking and identifies baffle former bolts as RVI items that are subject to programmatic aging management, and which require Category B-N-3 examination in accordance with the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, ISI Program, Subsection IWB. Subsection IWB provides requirements for the visual inspection of removable core support structures. Further, the B&WOG indicates that visual inspections may not be adequate to detect bolting cracking that is inaccessible. Therefore, the B&WOG implemented a program to manage the effects of aging due to cracking as described in Section 4.6 of the topical report.

In BAW-2248, Section 4.6, the B&WOG outline generalized activities of the implemented program. One of the defined purposes of the aging management program, according to the report, is to establish appropriate monitoring, inspection techniques and inspection programs that will continue to maintain the RVI functional through the extended life of the plant.

During meetings with the staff subsequent to the submittal of the BAW-2248, the B&WOG described current and ongoing baffle bolt activities that included preparation for a possible augmented baffle bolt inspection during the next 10-year interval at a B&W lead plant.

By letter dated December 2, 1998 (Reference 5), NRC forwarded requests for additional information (RAI) with regard to baffle bolt cracking, NRC RAI #12 and RAI #13.

NRC RAI #12 requested the B&WOG to describe the baffle bolt inspections that will be conducted prior to the start of the period of extended operation and indicate how these actions providea basis for assuring that the baffle bolt monitoring and inspection techniques that are planned for implementation during the period of extended operation are appropriate.

NRC RAI #13 requested B&WOG to describe the program that will be implemented as outlined in Section 4.6 of BAW-2248 with regard to the aging management of reactor internals baffle bolts, and to describe the overall inspection program, including aspects such as intervals, monitoring and inspection techniques.

By letter dated February 18, 1999, (Reference 6), the B&WOG provided a response to the RAIs. The response to NRC RAI #4 indicates that the technical elements of the B&WOG RVI aging management program were presented during a meeting with the NRC on April 23, 1998. Since that meeting, the industry has initiated a project to address generic materials issues and the scope of the B&WOG RVI aging management program has changed. The PWR MRP was established during the second quarter of 1998 to address and resolve existing and emerging PWR materials issues. An Issues Task Group (ITG) was formed to manage emerging RVI materials issues. The ITG on reactor vessel internals is currently addressing the issue of cracking, reduction of fracture toughness and loss of preload related to baffle bolts and associated materials. The data and information acquired from these various ITG activities will be used to determine the necessary steps in managing the effects of aging on baffle bolts, including future inspection plans. These plans are expected to be outlined on a plant specific basis, possibly beginning with the inspection at Oconee Unit 1 during its fourth inservice inspection (ISI) interval.

Each renewal applicant will be responsible for using the tools provided by both the ITG and owners groups to determine the necessary steps (e.g., inspections, operability determinations, and replacements) to manage the applicable baffle bolt aging effects. Therefore, the requested information of the B&WOG in RAI #12 and RAI #13, with regard to the management of the effects of aging on baffle bolts is converted to a renewal applicant action item based on this transfer of responsibility from the B&WOG to individual applicant. This is Renewal Applicant Action Item 4.

3.3.2 Loss of Material Due to Wear

The B&WOG proposes to continue the ASME B&PV Code Section XI program to manage the loss of material of RVI items due to wear that could cause loss of the RVI function(s) during the period of extended operation. The RVI items are managed by Examination Category B-N-3 of the ASME B&PV Code, Section XI, Subsection IWB, which provides requirements for the visual inspection (VT-3) for removable RVI structures. These requirements define conditions in IWB-3520.2 which, if detected, must be corrected prior to continued service. These conditions include wear of mating surfaces that may lead to loss of function. Because of our experience with inspections performed in accordance with ASME B&PV Code, Section XI, Subsection IWB, the staff concludes that VT-3 inspections are capable of detecting the loss of material due to wear.

3.3.3 Reduction of Fracture Toughness

Thermal Embrittlement

The topical report stated that thermal embrittlement of all RVI components is managed by visual inspection (VT-3) in accordance with Examination Category B-N-3 of the ASME Section XI inservice inspection program, Subsection IWB. Aging management for vent valve bodies and retaining rings is also accomplished through vent valve testing and (visual) inspection requirements (at each refueling outage) in accordance with plant technical specifications at ANO-1 and TMI-1, and the Pump and Valve In-Service Test Program at ONS-1, -2, and -3. NRC staff does not agree that loss of fracture toughness can be managed through VT-3 inspection, and instead has proposed an alternative management methodology. VT-3 inspection may not be capable of detecting fine cracks that could lead to failure of thermally embrittled components.

The vent valve retaining rings would be subject to supplemental (enhanced VT-1) examination. This examination could be modified or eliminated, provided that the applicant can demonstrate through data (including microstructural considerations) and evaluation that loss of fracture toughness by thermal embrittlement and/or neutron irradiation embrittlement is not significant for the vent valve retaining rings. Such a demonstration could follow the same frame work as that proposed below for CASS RVI components.

The RVI components fabricated from CASS are potentially subject to a synergistic loss of fracture toughness due to the combination of thermal and neutron irradiation embrittlement. This enhanced loss of fracture toughness is not accounted for by current CASS screening criteria either in BAW-2243A nor in revisions to EPRI TR-106092 (Reference 7). To account for this synergistic loss of fracture toughness, a modified approach for CASS RVI components is proposed. This modified approach consists of either a supplemental (enhanced VT-1) examination of the affected components as part of the applicant's 10-year ISI program during the license renewal term, or a component-specific evaluation to determine the susceptibility to loss of fracture toughness. The proposed evaluation will first examine the neutron fluence of the component. If the neutron fluence is greater than 1×10^{17} n/cm^2 (E > 1 MeV), a mechanical loading assessment would be conducted for the component. This assessment will determine the maximum tensile loading on the component during ASME Code Level A, B, C and D conditions. If the loading is compressive or low enough to preclude fracture of the component, then the component would not require supplemental inspection. Failure to meet this criterion would require continued use of the supplemental (enhanced VT-1) inspection. If the neutron fluence is less than 1×10^{17} n/cm^2 (E > 1 MeV), an assessment would be made to determine if the affected component(s) are bounded by the screening criteria in EPRI TR-106092 (Reference 7), modified as described below. In order to demonstrate that the screening criteria in EPRI TR-106092 (Reference 7) are applicable to RVI components, a flaw tolerance evaluation specific to the reactor vessel internals would be performed. If the screening criteria are not satisfied, then a supplemental (enhanced VT-1) inspection will be performed on the component.

The CASS components should be evaluated to the criteria in EPRI TR-106092 with the following additional criteria:

- Statically cast components with a molybdenum content meeting the requirements of SA-351 Grades CF3 and CF8 and with a delta ferrite content less than 10 percent will not need supplemental examination.

- Ferrite levels will be calculated using Hull's equivalent factors or a method producing an equivalent level of accuracy (±6 percent deviation between measured and calculated values).

- Cast stainless components containing Niobium are subject to supplemental examination.

- Flaws in CASS with ferrite levels less than 25 percent and no niobium may be evaluated using ASME Code IWB-3640 procedures.

- Flaws in CASS with ferrite levels exceeding 25 percent or containing niobium must be evaluated using ASME Code IWB-3640 procedures. If Flaws are discovered in such components, fracture toughness data must be provided on a case-by-case basis.

Components that have delta ferrite levels below the screening criteria have adequate fracture toughness and do not require supplemental inspection. Components that have delta ferrite levels exceeding the screening criteria may not have adequate fracture toughness, as a result of thermal embrittlement, and do require supplemental inspection.

This proposed program was discussed with the B&WOG. It needs to be incorporated into the topical report. This is Topical Report Open Item 3.

The B&WOG methodology for estimating neutron fluence is contained in topical report BAW-2241. This methodology was approved by the staff; however, it is only applicable for calculating neutron fluence in the radial direction between the core edge to the reactor vessel cavity. Hence, the methodology can not be applied to RVI components above and below the core.

To determine whether CASS components are above or below the threshold value of 1×10^{17} n/cm^2, as discussed in Section 3.3.3 of the safety evaluation, the B&WOG must provide estimates of the neutron fluence of each CASS component at the expiration of the license renewal term, identify the method of determining the neutron fluence and provide justification for applicability of the method to components above or below the core. This is Topical Report Open Item 4.

Neutron Embrittlement

For non-CASS RVI components, the topical report stated that management of neutron embrittlement would be addressed through a planned RVIAMP. The purposes of this program are to continue the investigation of the potential aging effects that have been identified in the topical report for the RVI, and to establish appropriate monitoring and inspection programs that will continue to maintain the RVI functional through the period of extended operationt. Further, in response to an RAI, the B&WOG indicated that an industry group, the PWR MRP, is addressing neutron embrittlement of RVI components under the auspices of an RPV Internals Issue Task Group. The RAI response stated that the results of this MRP activity will be incorporated in the development of the RVIAMP.

The NRC staff proposed to the B&WOG a modified approach to manage neutron embrittlement of RVI components. In particular, a two-pronged approach was proposed. The two pronged approach is to perform inspections and to perform tests and analysis of irradiated material. The inspection part of the approach is supplemental (enhanced VT-1) examination of the components believed to be the limiting components for neutron irradiation embrittlement, considering both the susceptibility of the component to the aging mechanism as well as the material properties (in particular the fracture toughness) and the operating stresses on the component. These examinations would be included as part of the 10-year ISI program. Since the examination addresses the limiting components, plant-specific neutron fluence evaluations are not necessary. Initial consideration by the B&WOG indicated that the limiting components with respect to highest neutron fluence were the baffle plates and baffle former bolts. The B&WOG must identify the limiting components and incorporate this program into the topical report. This is Topical Report Open Item 5. Note that the supplemental (enhanced VT-1) examination does not apply to bolting due to accessibility limitation. Section 3.3.1 of the safety evaluation describes aging management for baffle former bolting.

The second part of this approach would be to determine the need for continuing the supplemental examination. Should data or evaluations from the MRP or any other industry research activity indicate that the supplemental (enhanced VT-1) examinations can be modified or possibly eliminated, each applicant would be required to provide plant-specific justification to demonstrate the basis for the modification or elimination of examinations.

3.3.4 Loss of Closure Integrity for Bolted Closures

In Section 3.4 of the topical report, the B&WOG indicates that bolting stress relaxation is considered an applicable aging mechanism for those components where maintaining a preload is important to the structural integrity function(s) of the RVI. These RVI bolts include: the control rod guide tube (CRGT) to upper grid fasteners; the core support shield to core barrel bolts; the core barrel to thermal shield bolts; lower internals assembly to core barrel bolts; the lower grid rib-to-shell fasteners; the shell forging to flow distributor bolts; and the lower internals assembly to thermal shield bolts.

In Section 4.4 of the topical report, the B&WOG indicates that the required programmatic management for these bolts is the VT-3 visual examination required by Examination Category B-N-3 of the ASME B&PV Code, Section XI ISI Program, Subsection IWB. The VT-3 visual examination is specifically designed to determine the general mechanical and structural conditions, including structural distortion and displacements, loose or missing parts, and wear of mating surfaces that may lead to the loss of integrity at bolted connections. IWB-3142 provides options for correcting the relevant condition(s), such as: (1) acceptance by supplemental surface and/or volumetric examination, in order to further characterize the condition;
(2) acceptance by corrective measures (i.e., re-establishing the preload) or repairs; or (3) acceptance by replacement of the item. However, the B&WOG indicates that it recognizes that visual examination may not be adequate to detect the loss of mechanical closure integrity of the RVI, and the GLRP has implemented a program to manage these aging effects as discussed in Section 4.6 of the topical report.

In Section 4.6 of the topical report, the B&WOG indicates a comprehensive aging management program will be developed to supplement the programmatic management for stress relaxation of RVI bolting contained in Section 4.4. The supplemented AMP will be implemented such that the RVI can perform their component intended function(s) for the period of extended operation. The B&WOG indicates the purpose of the program is to: (1) continue the investigation of the aging effects; and (2) establish the appropriate monitoring and inspection programs to continue to maintain the RVI functions. The B&WOG indicates the elements of the program that will be implemented to address stress relaxation of RVI bolting are: (1) to determine critical locations; and (2) establish appropriate monitoring and inspection techniques.

In the topical report, B&WOG indicates that the commitments to implement this AMP and to notify the NRC staff regarding the status of the program activities on a regular basis will be part of the updated FSAR supplement and will be included in any plant-specific license renewal application.

Since the B&WOG has proposed a comprehensive program to manage stress relaxation of RVI components, the staff considers the B&WOG proposal acceptable subject to applicant's fulfilment of the commitments included in the report. This is Renewal Applicant Action Item 1.

3.3.5 Change of Dimension

Should the evaluation described in Topical Report Open Item 1 demonstrate the need for an AMP, then the topical report would require revision to identify the elements of the AMP. This is addressed as part (c) of Topical Report Open Item 1.

3.4 Time-Limited Aging Analyses

Time-limited aging analyses are defined in 10 CFR 54.3 as those licensee calculations and analyses that:

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

(2) consider the effects of aging;

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

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

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

(6) are contained or incorporated by reference in the current licensing basis.

Paragraph 54.21(c)(1) requires the applicant to demonstrate that:

(i) the analyses remain valid for the period of extended operation;

(ii) the analyses have been projected to the end of the period of extended operation; or

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

In Section 4.5 of the topical report, the B&WOG identified the TLAA applicable to the RVI items within the scope of the report based on reviewing plant-specific docketed correspondence files, plant-specific FSARs, BAW topical reports, and the ASME B&PV Code, Section XI ISI requirements. The topical report identifies four TLAAs appplicable to the RVI:

- flow-induced vibration calculations and measurements to verify during hot function testing that the flow induced vibration stresses are below the endurance limit

- fatigue cumulative usage factor calculations, which rely on transient cycle count assumptions used for the design of the RVI

- calculation to demonstrate the deformation limits in the ASME Code are not violated as a result of neutron irradiation of RVI materials

- flaw growth calculations performed in accordance with the ASME B&PV Code, Section XI ISI requirements

The first two of these TLAAs are lumped together in the topical report under the heading " Fatigue - Cracking (Initiation and Growth)," and the third is referred to in the topical report as " Ductility - Reduction of Fracture Toughness." The flaw growth acceptance TLAA is identified in the topical report as requiring a plant-specific evaluation, and as such is not evaluated within the topical report. This is Renewal Applicant Action Item 5.

All of the TLAA analyses documents were identified in the topical report as required by 10 CFR 54.21(c)(1).

3.4.1 Fatigue - Cracking (Initiation and Growth)

The flow-induced vibration endurance limit assumptions were based on 10^{12} cycles for 40 years. The analysis was extended for the license renewal period by conservatively increasing the number of cycles to 10^{13}, and then determining the endurance limit using the latest ASME fatigue curves. The component stress values were found to be less than the endurance limit, rendering the evaluation acceptable, in conformance with the requirements of 10 CFR 54.21(c)(1).

In the topical report, B&WOG indicates that the design cyclic loadings and thermal conditions used for the analyses are defined in the component design specifications and that flow-induced vibration input used was obtained from hot functional testing data contained in the listed analyses documents. The ability to withstand cyclic loading without fatigue failure was evaluated using a cumulative usage factor methodology. For each utility the number of transients accrued to date was conservatively extrapolated, and in all cases it was found that the number of design cycles would not be exceeded in the period of extended operation. The B&WOG reported that each of the participating utilities monitors occurrences of design transients and is thus managing the potential for cracking resulting from fatigue. The topical report indicates that the plants must continue to monitor and track occurrences of design transients.

3.4.2 Ductility - Reduction of Fracture Toughness

Section 4.5.2 of BAW-2248 describes a TLAA related to the acceptability of the reactor vessel internals under loss-of-coolant-accident (LOCA) and seismic loading. The topical report states that Appendix E to BAW-10008, Part 1, Rev. 1, concludes "that at the end of 40 years, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits." The topical report indicates that this TLAA will be resolved on a plant-specific basis per 10 CFR 54.21 (c)(1)(iii) based on the results and conclusion of the planned RVIAMP. As described in the topical report, the planned RVIAMP program will provide the data necessary to resolve this TLAA.

Therefore, this item should be addressed as a renewal applicant action item on a plant-specific basis pending the results of the RVIAMP. This is Renewal Applicant Action Item 6.

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 topical report provides an acceptable demonstration that aging effects on reactor vessel internal components within the scope of this topical report will be adequately managed for the GLRP member plants, with the exception of the noted renewal applicant action items and topical report open items, so that there is reasonable assurance that the reactor vessel internal components will perform their intended functions in accordance with the CLB during the period of extended operation. The staff also concludes that, upon completion of the renewal applicant action items set forth in Section 4.1 below, the B&WOG topical report provides an acceptable evaluation of time-limited aging analyses for the reactor vessel internals for the GLRP member plants for the period of extended operation.

Any B&WOG GLRP member plant may reference this topical report in a license renewal application to satisfy the requirements of (1) 10 CFR 54.21(a)(3) for demonstrating that the effects of aging on the reactor vessel internal components within the scope of this topical report will be adequately managed and (2) 10 CFR 54.21(c)(1) for demonstrating that appropriate findings be made regarding evaluation of time-limited aging analyses for the reactor vessel internals for the period of extended operation. The staff also concludes that, upon completion of the renewal applicant action items set forth in Section 4.1 below, referencing this topical report in a license renewal application and summarizing in an FSAR supplement the aging management programs and the TLAA evaluations set forth in the topical report will provide the staff with sufficient information to make the necessary findings required by 10 CFR 54.29(a)(1) and (a)(2) for components within the scope of this topical report.

4.1 <u>Renewal Applicant Action Items</u>

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

(1) The license renewal applicant is to verify that the critical parameters for the plant are bounded by the topical report. Further, the renewal applicant is to commit to programs described as necessary in the topical report to manage the effects of aging during the period of extended operation on the functionality of the reactor vessel components. Applicants for license renewal will be responsible for describing any such commitments and identifying the appropriate regulatory control. Any deviations from the aging management programs within this topical report described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel internal components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

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

(3) License renewal applicants must identify whether the intended function of the RVI is to provide shielding for the RPV. If not an intended function, the license renewal applicant should provide justification for that conclusion. Should a license renewal applicant determine that the RVI's intended function is to provide shielding for the RPV, then the items that support this intended function, such as, the thermal shield and the thermal

shield upper restraint assemblies, must be identified and reviewed in accordance with 10 CFR 54.21(a)(3).

(4) According to B&WOG, one of its objectives in BAW-2248 states, "It is intended that NRC review and approval of this report will allow that no further review of the matters described herein will be needed when the report is incorporated by reference in a plant specific renewal license application." The license renewal applicant must address matters not described in the report, such as the baffle former bolt cracking issues addressed in Section 3.3.1 of this SE pertaining to References 5 and 6, with regard to the industry ITG project, initiated after April 23, 1998, to address generic RVI materials issues. The BWOG indicates this industry effort resulted in subsequent changes in the BWOG RVI aging management program. The ITG is currently addressing the issues of cracking of baffle bolts. The BWOG indicates that the changes in the aging management program now requires the applicants to be responsible for using the industry ITG project developed information to determine the necessary steps (e.g., inspection, operability determinations, and replacements) in managing the effects of aging on baffle bolts.

(5) If flaws have been detected in the reactor vessel internals, a TLAA plant-specific evaluation must be performed to determine whether the flaw growth is acceptable in accordance with the ASME B&PV Code, Section XI, inservice inspection requirements at the expiration of the renewed license.

(6) Plant-specific analysis is required to demonstrate that, under loss-of-coolant-accident (LOCA) and seismic loading and with irradiation accumulated at the expiration of the renewal license, the internals have adequate ductility to absorb local strain at the regions of maximum stress intensity and will meet deformation limits. The data to demonstrate that the internals will meet the deformation limits at the expiration of the renewed license will be developed from the RVAMP.

4.2 Topical Report Open Items

(1) The B&WOG should identify:

(a) How much of a change in dimension would be required before the internals would not be able to meet their intended function,

(b) What ongoing programs, if any will evaluate the impact of the void swelling on the intended function of the internals; and

(c) When these programs will provide data to determine whether void swelling could impact the intended function of the internals.

(2) The B&WOG must modify their aging management program as described in Section 3.3.1 of the safety evaluation for managing the effects of cracking (SCC and IASCC). One acceptable option is the program described in Section 3.3.1. In addition, the B&WOG must identify the limiting components (excluding RVI bolting).

(3) The B&WOG must modify their aging management program as described in Section 3.3.3 of the safety evaluation, under "Thermal Embrittlement" for managing the

effects of thermal embrittlement, and possibly synergistic effects with neutron embrittlement, on the fracture toughness of cast austenitic stainless steel (CASS) RVI components. One acceptable option is the program described in Section 3.3.3 under " Thermal Embrittlement."

(4) To determine whether CASS components are above or below the threshold neutron fluence value of 1 x 10^{17}n/cm^2, as discussed in Section 3.3.3, the B&WOG must provide estimates of the neutron fluence of each CASS component at the expiration of the license renewal term, identify the method of determining the neutron fluence and provide justification for applicablity of the method to components above or below the core.

(5) The B&WOG must modify their aging management program as described in Section 3.3.3 under "Neutron Embrittlement" for managing the effects of neutron embrittlement on the fracture toughness of RVI components. One acceptable option is the program described in Section 3.3.3 of the safety evaluation, under "Neutron Embrittlement." In addition. The B&WOG must identify the limiting components (excluding RVI bolting).

5.0 REFERENCES

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Federal Register, Vol. 60, No. 88, pp. 22461-22495, May 8, 1995.

2. BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," Babcock & Wilcox Owners Group, July 1997.

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

4. EPRI Technical Report TR-107521, "Generic License Renewal Technical Issues Summary," Electric Power Research Institute, April 1998.

5. Letter from Raj K. Anand, NRC, to David J. Firth, December 2, 1998, "Request for Additional Information Regarding the Babcock & Wilcox Owners Group Generic License Renewal Program Topical Report Entitled Demonstration of the Management of Aging Effects for the Reactor Vessel Internals, BAW-2248, July 1997."

6. Letter from William R. Gray to David B. Mathews, NRC, dated February 18, 1999, "B&WOG Generic License Renewal Program Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (RAIs 1 through 14 from December 4, 1998)."

7. EPRI Technical Report TR-106092, "Evaluation of Thermal Aging Embrittlement for Cast Austenitic Stainless Steel Components in LWR Reactor Coolant Systems," Electric Power Research Institute, September 1997.

APPENDIX A

LIST OF CORRESPONDENCE

1. Letter from David J. Firth (B&WOG) to Marylee Slosson (NRC), July 29,1997, transmitting B&WOG Generic License Renewal Program Topical Report BAW-2248, " Demonstration of the Management of Aging Effects for the Reactor Vessel Internals,"

July, 1997.

2. Letter from Raj K. Anand (NRC) to David J. Firth (B&WOG), December 2, 1998, " Request for Additional Information Regarding the Babcock & Wilcox Owners Group Generic Licensee Renewal Program Topical Report Entitled Demonstration of the Management of Aging Effects for the Reactor Vessel Internals, BAW-2248, July 1997."

3 Letter from William R. Gray (B&WOG) to David B. Mathews (NRC), February 18, 1999, " B&WOG Generic License Renewal Program Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (RAIs 1 through 14 from December 4 1998)."

4 NRC Meeting Summary dated May 6, 1998, entitled " Summary of Meeting on April 23, 1998, between the U.S. NRC staff and B&WOG representatives to discuss the status of the B&WOG Generic License Renewal Program."

Table of Contents

APPENDIX E

Appendix E contains the GLRP responses to the NRC Draft Safety Evaluation Open Items dated June 29, 1999 and August 16, 1999

Duke Energy Corporation | Oconee 1, 2, 3
Entergy Operations, Inc. | ANO-1
Florida Power Corporation | Crystal River

The B&W Owners Group

GPU Nuclear, Inc. | TMI-1
Toledo Edison Company | Davis-Besse
Framatome Technologies, Inc

Working Together to Economically Provide Reliable and Safe Electrical Power

Suite 525 • 1700 Rockville Pike • Rockville, MD 20852 • (301) 230-2100

June 29, 1999
OG-1762

Project No. 683

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. David B. Matthews, Deputy Director
Division of Reactor Program Management

Subject: B&WOG Generic License Renewal Program Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" - Response to DSE Topical Report Open Items

Reference: Letter from Raj K. Anand to William R. Gray, May 26, 1999, entitled "Draft Safety Evaluation Concerning the Babcock & Wilcox Owners Group Generic License renewal Program Topical Report Entitled, Demonstration of the Management of Aging Effects for the Reactor Vessel Internals, BAW-2248, July 1997"

Gentlemen:

The Reference contains the Draft Safety Evaluation on the subject document. Each Topical Report Open Item from Section 4.2 of the DSE, the B&WOG response, and the recommended change to the appropriate section of BAW-2248 are included in the attached table.

In addition, the B&WOG suggests that the following clarifications be made to the NRC safety evaluation report.

Section 3.3.1 Cracking
The B&WOG understands through discussions with the staff that a supplemental (enhanced VT-1) examination is analogous to the visual examination that is performed on BWR core shroud welds. However, the complete description of an "enhanced VT-1" examination does not appear to be available in a document that is available to the public. The B&WOG recommends that the staff make the definition of "enhanced VT-1" available to the public.

Section 3.3.3 Reduction of Fracture Toughness
With regard to the additional screening criteria for CASS components, it is the understanding of the B&WOG that the term "meeting" should be replaced with "exceeding" in the first item. The text should read as follows.

- Statically cast components with a molybdenum content exceeding the requirements of SA-351 Grades CF3 and CF8 and with a delta ferrite content less than 10 percent will not need supplemental examination.

We look forward to receiving the final safety evaluation on BAW-2248 and are prepared to work expeditiously with the Staff to resolve any additional items that may arise.

Please call me at 804/832-2783 if you need any additional information.

Sincerely,

W R Gray

William R Gray
Project Manager
B&W Owners Group Services

WRG/mcl
Attachment: RAI Response Table

c: with Attachment

R. K. Anand	- US NRC/NRR
R. L. Gill	- Duke Energy Corporation
G. D. Robison	- Duke Energy Corporation
W. F. Brady	- Duke Energy Corporation
D. E. Whitaker	- Duke Energy Corporation
W. H. Mackay	- Entergy Operations, Inc.
R. W. Clark	- Entergy Operations, Inc.
D. F. Spond	- Entergy Operations, Inc.
D. J. Masiero	- GPU Nuclear, Inc.
G. L. Lehmann	- GPU Nuclear, Inc.
Z. B. Fu	- GPU Nuclear, Inc.
M. A. Rinckel	- Framatome Technologies
R. N. Edwards	- Framatome Technologies
S. Fyfitch	- Framatome Technologies
F. M. Gregory	- Framatome Technologies

NRC Open Item No.	Open Item	B&WOG Response	Proposed revision to BAW-2248	Remarks
DSE-1	The B&WOG should identify: (a) How much of a change in dimension would be required before the internals would not be able to meet their intended function, (b) What ongoing programs, if any will evaluate the impact of the void swelling on the intended function of the internals; and (c) When these programs will provide data to determine whether void swelling could impact the intended function of the internals.	Dimensional changes that may be caused by void swelling have not been evaluated as part of the design of the reactor vessel internals. Therefore, the B&WOG cannot specify the dimensional changes caused by void swelling that would lead to loss of the reactor vessel internals intended functions. The B&WOG and industry programs that are characterizing void swelling are reported below. As stated in the B&WOG cover letter of the RAI responses [B&WOG Generic License Renewal Topical Report BAW-2248, Demonstration of the Management of Aging Effects for the Reactor Vessel Internals RAIs 1 through 14 from December 14, 1998, OG-1744, 2/18/99], the RPV Internals ITG has developed generic programs that address applicable aging effects such as cracking, reduction of fracture toughness and loss of mechanical closure integrity. The mechanisms that are being considered include irradiation-assisted stress corrosion cracking, stress corrosion cracking, irradiation embrittlement, void swelling, and stress relaxation. The proposed ITG work regarding void swelling includes the following: (1) preparation of a white paper that summarizes available void swelling data and determines the effect on RV internals components; (2) determination of the occurrence and magnitude of swelling in CE-designed core shroud materials; and, (3) examination of a highly irradiated instrumentation thimble (~75 dpa) from a PWR in an internationally-sponsored program. In addition, the ongoing EPRI Joint Baffle Bolt (JoBB) Program, established in 1996, has been incorporated into the RPV Internals ITG to provide plant inspection and research test data. The JoBB Program will examine highly irradiated PWR component items as well as research test reactor specimens for evidence of swelling. Swelling data will be acquired and industry assessments will be completed under both the ITG and JoBB programs. The B&WOG will utilize data and information from these programs to perform design-specific analyses (i.e., safety evaluations), as necessary.	None.	

NRC Open Item No.	Open Item	B&WOG Response	Proposed revision to BAW-2248	Remarks
		A status report of the B&WOG RV Internals Program, which included information regarding the above industry programs, was provided to the NRC in May 1999 [Status Report of B&W Owners Group Reactor Vessel Internals Program, OG-1753, May 10, 1999]. The status report (see the discussion under stress relaxation and irradiation embrittlement issues in Section 3) indicates that tasks associated with void swelling began in 1998; however, the majority of the activities are scheduled to begin in 2001. While it is premature to evaluate the effects of dimensional changes in the internals without knowledge of a reasonable bounding value, sufficient data should be available in the 2003-2004 timeframe to determine whether void swelling could impact the intended function of the internals. For reactor vessel internals items that will receive high fluence, such as the baffle plates and baffle bolts, any significant effect of void swelling would most likely lead to cracking of the baffle bolts. Therefore, the aging management programs proposed for baffle bolts and baffle plates will provide adequate assurance that void swelling will not compromise the ability of the RV internals to perform their intended function in the period of extended operation. From a process standpoint and a technical standpoint, the B&W Owners Group requests that the staff redirect this BAW-2248 open item to the Nuclear Energy Institute (NEI) to be addressed and resolved on an industry-wide basis. The following paragraphs provide the basis for this request. From a process standpoint, the concern for void swelling is most appropriately addressed on a generic industry-wide basis. The concern of void swelling was identified in EPRI TR-107521, *Generic License Renewal Technical Issues Summary*, April 1998 which was submitted to the NRC staff by NEI letter dated June 1, 1998. It has also been identified as a Priority 3 Generic License Renewal Issue (see NRC letter dated April 8, 1999 to NEI–License Renewal Issues List). The staff in a letter to NEI dated		

NRC Open Item No.	Open Item	B&WOG Response	Proposed revision to BAW-2248	Remarks
		August 12, 1998, previously provided guidance for establishing priorities of Generic License Renewal Issues. Priority 3 issues include generic issues determined to be important, but not important enough to include under Priority 2. Priority 3 issues have non-mandatory due dates driven by scheduled revisions to SRP-LR, DG-1047, and NEI-95-10. Void swelling is a Priority 3 generic technical issue that is not unique to B&W-designed reactor vessel internals. The staff has already determined that it is a relatively low priority. No additional requirements relative to void swelling should be imposed on the B&W Owners Group in order to complete BAW-2248 or on license renewal applicants that may choose to use BAW-2248. The discussion of reactor vessel internals void swelling in TR-107521 provides a comprehensive technical basis for the industry position. The B&W Owners Group supports the industry position that it would be prudent to follow or participate in research and development activities that may produce information that will determine whether or not void swelling would be a potentially significant issue for PWR's in the future. Upon completion of the research, appropriate actions will be taken by licensees to ensure that the reactor vessel internals will continue to perform its intended function. From a technical standpoint, the issue of void swelling is most appropriately addressed generically.		
DSE-2	The B&WOG must modify their aging management program as described in Section 3.3.1 of the safety evaluation for managing the effects of cracking (SCC and IASCC). One acceptable option is the program described in Section 3.3.1. In addition, the B&WOG must identify the limiting components (excluding RVI bolting).	The staff has suggested in Section 3.3.1 of the DSE that the B&WOG identify selected reactor vessel internals items (i.e., welds, plates, or forgings; bolting is addressed separately through the RVIAMP and RAIs 12 and 13) that are considered to be the most susceptible to cracking by irradiation-assisted stress corrosion cracking (IASCC) and reduction of fracture toughness by irradiation embrittlement (IE). Once the limiting items are identified, an acceptable method for aging management may include a supplemental (enhanced VT-1) examination of the limiting items. The B&WOG understands that an enhanced VT-1	Add text in "response" column to Section 4.6 of BAW-2248.	

NRC Open Item No.	Open Item	B&WOG Response	Proposed revision to BAW-2248	Remarks
		examination is analogous to the visual examination that is performed on BWR core shroud welds. However, the definition of an "enhanced VT-1" examination does not appear to exist in a document that is available to the public. As an alternative to an "enhanced VT-1" examination, the B&WOG Reactor Vessel Internals Aging Management Program (RVIAMP) will develop an "augmented inspection program" as described below. Inspection of the limiting items would bound the other reactor vessel internals items that may be susceptible to IASCC and IE. As described in BAW-2248, the reactor vessel internal items adjacent to the active fuel assemblies that may be susceptible to cracking by IASCC and reduction of fracture toughness by IE include the baffle plates and baffle bolts, former plates, and core barrel. Aging management of internals bolting applications is addressed by the RVIAMP, and is not within the scope of this response. Selected internals items (i.e., plates, forgings, and welds) that will receive the highest fluence (E> 1.0 MeV) at 48 EFPY and that are subjected to the highest design stresses may be identified as leading candidates for augmented inspection. Stress summaries for the reactor vessel internals items were reviewed and the maximum loads for design plus operational basis earthquake (OBE) are predicted to occur in the baffle plates (14.8 ksi per BAW-10008, Revision 1, Part 1, June 1970—Table 1). Fluence estimates for the baffle plates at 48 EFPY are higher than the former plates and the core barrel. Therefore, with regard to plate, forging, and weld materials adjacent to the active fuel assemblies, selected regions of the baffle plates that are adjacent to the fuel assemblies would be the limiting locations that are most susceptible to cracking and reduction of fracture toughness. The B&WOG RV Internals Program, which was presented to the NRC on April 23, 1998 [NRC Meeting Summary dated May 6, 1998, entitled "Summary of Meeting on April 23, 1998, Between the U.S. Nuclear Regulatory		

NRC Open Item No.	Open Item	B&WOG Response	Proposed revision to BAW-2248	Remarks
		Commission and B&WOG Representatives to Discuss the Status of the B&WOG Generic License Renewal Program, Project Number 683], contains a number of tasks to assess the susceptibility of the B&W-designed reactor vessel internals to cracking and reduction of fracture toughness. In addition, industry efforts, which include the reactor vessel internals ITG and JoBB programs, will provide cracking and reduction of fracture toughness susceptibility data (SCC, IASCC, and IE) for the materials used in the manufacture of the B&W-designed RV internals (e.g., Type 304 stainless steel, Type 308 weld metal, and Alloy X-750 material). This work is scheduled to be completed in 2004, although some data will be available on a yearly basis, beginning later in 1999. The results of these programs will assist the B&WOG in identifying limiting locations and establishing appropriate monitoring and inspection programs. The B&WOG agrees that an augmented inspection of selected accessible regions of the baffle plates is appropriate to manage cracking and reduction of fracture toughness during the period of extended operation. Details of the augmented inspection (e.g., sample size using risk-based methods, examination method, and acceptance criteria) will be developed by the RVIAMP as discussed above. Additional details (e.g., timing) of the plant-specific augmented inspections will be addressed at the time of license renewal application. Should data or evaluations from the MRP or any other industry research activity indicate that the augmented inspections of the baffle plates can be modified or eliminated, justification will be provided to the staff demonstrating the basis for its modification or elimination.		
DSE-3	The B&WOG must modify their aging management program as described in Section 3.3.3 of the safety evaluation, under "Thermal Embrittlement" for managing the effects of thermal	The applicable aging effect for the CASS items and precipitation-hardening stainless steel items is reduction of fracture toughness that may prevent the RV internals from performing their intended functions under design basis conditions. The reactor vessel internals items fabricated	Add text in "response" column to Section 4.6 of BAW-2248.	

NRC Open Item No.	Open Item	B&WOG Response	Proposed revision to BAW-2248	Remarks
	embrittlement, and possibly synergistic effects with neutron embrittlement, on the fracture toughness of cast austenitic stainless steel (CASS) RVI components. One acceptable option is the program described in Section 3.3.3 under "Thermal Embrittlement."	from cast austentitic stainless steel include the CRGT assembly spacer castings, Oconee-3 outlet nozzles, and the vent valve bodies. The vent valve retaining rings are fabricated from precipitation-hardening stainless steel. Programmatic activities to manage reduction of fracture toughness for these items are discussed below. <u>Control Rod Guide Tube Assemblies—Spacer Castings</u> The outer portion of each CRGT assembly (69 assemblies per plant) consists of a tall pipe (or guide housing) welded to a CRGT assembly flange at the bottom. The inside of each CRGT assembly consists of an internal sub-assembly with ten parallel horizontal spacer castings to which are brazed 12 perforated vertical rod guide tubes and 4 pairs of vertical rod tube guide sectors, also called "C-tubes". The spacer castings are 3/4-inch thick disks, with internal spaces to conform to the general shape of the control rod spider (see Figure 2-6 from BAW-2248) and margins to permit RCS flow around the castings, rod guide tubes, and rod guide sectors. Each CRGT assembly contains 10 spacer castings and there are 69 CRGT assemblies per plant (i.e., 690 spacer castings per plant). All of the spacer castings are fabricated from cast austenitic stainless steel, ASTM-A351, Grade CF3M, and all are believed to be statically cast owing to the geometry of the part. Fabrication records were reviewed and the chemical compositions of approximately 175 heats of material for the spacer castings were obtained; this is believed to encompass the majority of heats for the participating plants since numerous spacer castings were fabricated from a single heat of material. The amount of ferrite was estimated for each heat by calculating chromium and nickel equivalent values using Hull's equivalent factors. The average amount of ferrite for the 175 heats is 20%. In accordance with the screening criterion reported in EPRI TR-106092, the spacer castings may be susceptible to reduction of fracture toughness by thermal embrittlement. However, as reported in EPRI TR-106092, the lower bound fracture toughness of the CF3M material (assumed		

NRC Open Item No.	Open Item	B&WOG Response	Proposed revision to BAW-2248	Remarks
		analogous to CF8M with respect to fracture toughness) is equivalent to that of selected austenitic stainless steel piping weldments. The residual fracture toughness of the spacer castings should be sufficient to assure the function of the spacer castings under design basis conditions. The function of the spacer castings is to support the guide tubes and guide sectors, and the function of the guide tubes and guide sectors is to ensure unimpeded travel of the control rods and spider under design basis conditions. The maximum loads on the guide tubes and guide tube sectors under design basis conditions are insignificant [see BAW-10008, Revision 1, Figure 13]. These loads, which are distributed to two spacer castings, would not lead to loss of function even with a reduction in fracture toughness of the spacer castings. An estimate of the 48 EFPY fluence at the spacer casting closest to the center fuel assembly, which is approximately 5-inches above the upper grid rib section in the plenum assembly, is approximately one order of magnitude below the threshold for irradiation embrittlement of 5.0E20 n/cm^2 reported in NUREG/CR-6048, Section 4.1, Page 31. Since the estimated fluence exceeds the NRC's screening criterion of 1.0E17 n/cm^2, neutron embrittlement cannot be eliminated as an applicable aging mechanism for the spacer castings. While the CRGT spacer castings may be susceptible to reduction of fracture toughness by thermal embrittlement and neutron embrittlement, no specific aging mechanism has been identified that would initiate a crack within the spacer castings. Since the function of the spacer castings is to support the rod guide tubes and rod guide sectors, and these items merely serve as a conduits for the control rods, it is unlikely that the CRGT spacers will fail and prohibit control rod motion under design basis conditions. A significant number of castings would have to fail in each CRGT assembly to impede control rod motion and failure of a single casting owing to reduction of fracture toughness		

NRC Open Item No.	Open Item	B&WOG Response	Proposed revision to BAW-2248	Remarks
		is believed to be unlikely. However, the B&WOG agrees that a limited augmented inspection of a sample population of accessible control rod guide tube spacer castings is appropriate to manage reduction of fracture toughness during the period of extended operation. Details of the augmented inspection (e.g., sample size using risk-based methods, examination method, and acceptance criteria) will be developed by the RVIAMP. Additional details (e.g., timing) of the plant-specific augmented inspections will be addressed at the time of license renewal application. Should data or evaluations from the MRP or any other industry research activity indicate that the augmented inspections of the spacer castings can be modified or eliminated, justification will be provided to the staff demonstrating the basis for its modification or elimination. <u>Vent Valves—Retaining Rings and Valve Bodies</u> The B&WOG agrees that a limited augmented inspection of a sample population of retaining rings and vent valve bodies is appropriate to manage reduction of fracture toughness for these items during the period of extended operation. Details of the augmented inspection (e.g., sample size using risk-based methods, examination method, and acceptance criteria) will be developed by the RVIAMP. Additional details (e.g., timing) of the plant-specific augmented inspections will be addressed at the time of license renewal application. Should data or evaluations from the MRP or any other industry research activity indicate that the augmented inspections of the retaining rings and valve bodies can be modified or eliminated, justification will be provided to the staff demonstrating the basis for its modification or elimination. <u>Oconee 3 Outlet Nozzles</u> The B&WOG agrees that an augmented inspection of the Oconee 3 outlet nozzles is appropriate to manage reduction of fracture toughness for these items during the period of		

NRC Open Item No.	Open Item	B&WOG Response	Proposed revision to BAW-2248	Remarks
		extended operation. Details of the augmented inspection (e.g., sample size, examination method, and acceptance criteria) will be developed by the RVIAMP. Additional details (e.g., timing) of the plant-specific augmented inspections will be addressed at the time of license renewal application. Should data or evaluations from the MRP or any other industry research activity indicate that the augmented inspections of the ONS-3 outlet nozzles can be modified or eliminated, justification will be provided to the staff demonstrating the basis for its modification or elimination.		
DSE-4	To determine whether CASS components are above or below the threshold neutron fluence value of 1 x 10^{17}n/cm^2, as discussed in Section 3.3.3, the B&WOG must provide estimates of the neutron fluence of each CASS component at the expiration of the license renewal term, identify the method of determining the neutron fluence and provide justification for applicability of the method to components above or below the core.	The fluence screening criterion (i.e., 10^{17}n/cm^2) was not used to eliminate reduction of fracture toughness as an applicable aging effect for any RV internals item. Instead, the B&WOG has agreed that augmented inspections of selected locations will be sufficient to manage reduction of fracture toughness during the period of extended operation.	See proposed response to DSE-3.	
DSE-5	The B&WOG must modify their aging management program as described in Section 3.3.3 under "Neutron Embrittlement" for managing the effects of neutron embrittlement on the fracture toughness of RVI components. One acceptable option is the program described in Section 3.3.3 of the safety evaluation, under "Neutron Embrittlement." In addition. The B&WOG must identify the limiting components(excluding RVI bolting).	Please see the response to DSE-2 for a discussion of aging management of cracking and reduction of fracture toughness by neutron embrittlement.	See proposed revision to DSE-2.	

Duke Energy Corporation Oconee 1, 2, 3
Entergy Operations, Inc. ANO-1
Florida Power Corporation Crystal River

The B&W Owners Group

GPU Nuclear, Inc. TMI-1
Toledo Edison Company Davis-Besse
Framatome Technologies, Inc

Working Together to Economically Provide Reliable and Safe Electrical Power

Suite 525 • 1700 Rockville Pike • Rockville, MD 20852 • (301) 230-2100

August 16, 1999
OG-1766

Project No. 683

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. David B. Matthews, Deputy Director
Division of Reactor Program Management

Subject: B&WOG Generic License Renewal Program Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" - Response to DSE Topical Report Open Item 1 Regarding Void Swelling

References:
1) Letter from Raj K. Anand to William R. Gray, May 26, 1999, entitled "Draft Safety Evaluation Concerning the Babcock & Wilcox Owners Group Generic License Renewal Program Topical Report Entitled, Demonstration of the Management of Aging Effects for the Reactor Vessel Internals, BAW-2248, JULY 1997"
2) Letter from William R. Gray to Raj K. Anand, June 29, 1999, entitled "B&WOG Generic License Renewal Program Topical Report BAW-2248, 'Demonstration of the Management of Aging Effects for the Reactor Vessel Internals' - Response to DSE Topical Report Open Items"

Gentlemen:

The Reactor Vessel Internals Report (BAW-2248) Draft Safety Evaluation is contained in Reference 1. Members of the B&WOG met with the NRC on July 19, 1999, to discuss the B&WOG responses to the DSE Topical Report Open Items (Reference 2). It is the understanding of the B&WOG that Open Items 2 through 5 require no further information to complete the staff's review and Topical Report Open Item 1 regarding void swelling remains open. The NRC suggested that in addition to providing a commitment for further characterization of void swelling through research and development programs, the B&WOG should identify critical locations that may be susceptible to void swelling and propose an augmented inspection program to ensure that dimensional changes would not lead to loss of the internals intended functions. The B&WOG agreed to revise its position on void swelling and provides the following response, which supersedes the response to Open Item 1 provided in the June 29, 1999, transmittal (Reference 2).

NRC Open Item 1 states that the B&WOG should identify:

(a) How much of a change in dimension would be required before the internals would not be able to meet their intended function,

(b) What ongoing programs, if any, will evaluate the impact of the void swelling on the intended function of the internals; and

(c) When these programs will provide data to determine whether void swelling could impact the intended function of the internals.

Deflections of internals items that may lead to loss of reactor vessel internals functions are listed in the topical report entitled "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake," BAW-10008, Part 1, Revision 1, June 1970, Table 2 (see attachment). This report was reviewed and approved by the Atomic Energy Commission [AEC letter from R.C. Deyoung to J.F. Mallay dated September 28, 1972]. In accordance with BAW-10008, Table 2 for example, a reduction of the core barrel diameter could result in distortion of the fuel assembly spacer grids thus affecting core cooling geometry during design basis accidents. However, analyses will be performed over the next 3 to 5 years to assess potential broken baffle bolts and the subsequent effect on fuel grid deformation during design basis accidents. Therefore, the diametrical reduction of the core barrel that results in loss of function will be re-examined in those analyses and the deflection criteria in BAW-10008, Table 2, will be revised accordingly.

Void swelling is a complex function of neutron flux, neutron fluence, operating temperature, operating stress, material composition, and material fabrication (e.g., solution annealed vs. cold worked). Components of the core barrel sub-assembly (i.e., baffle bolts, former plates, and baffle plates) are considered to be the susceptible items to dimensional changes by void swelling during the period of extended operation. The core barrel items will receive the highest neutron fluence, are at the highest operating temperatures due to gamma heating, contain the highest operating stresses when compared to the other internals items, and are fabricated from solution annealed Type 304 stainless steel.

At present, data is not available that shows a specific threshold for the onset of void swelling in solution annealed Type 304 stainless steel in a PWR environment. However, the onset of void swelling in solution annealed and 10, 20, and 30 percent cold worked Type 304 stainless steel exposed to a <u>breeder reactor environment</u> is available and is estimated to start at fluence levels of approximately 4 to 8E22 n/cm^2, E> 0.1 MeV at a temperature of 440 C [Effects of Radiation on Materials, ASTM STP725, Comparison of High-Fluence Swelling Behavior of Austenitic Stainless Steels, Page 484]. For the B&W-designed RV internals, which operate significantly below 440 C (approximately 290-340 C and lower), the only sub-assembly that will receive fluence on the order of 4 to 8E22 n/cm^2 (E> 0.1 MeV) is the core barrel sub-assembly. Estimated 48 EFPY fluence levels for the remaining reactor internals sub-assemblies are one or more orders of magnitude less than 4 to 8E22 n/cm^2 (E> 0.1MeV). However, additional research needs to be completed to determine the applicability of breeder reactor test data to PWRs.

It is also anticipated that the occurrence of void swelling during the period of extended operation would be extremely localized. With regard to the core barrel sub-assembly, the most likely location for void swelling to occur is believed to be within selected regions of the former plates and baffle bolts that are adjacent to the locations where the edges of the baffle plates are joined. These regions are subjected to some of the highest operating temperatures (due to gamma heating), operating stresses, and fluence when compared to other regions of the core barrel sub-assembly.

<u>B&WOG and ITG Programs to Manage Void Swelling</u>

As stated in the B&WOG cover letter to the RAI responses [B&WOG Generic License Renewal Topical Report BAW-2248, Demonstration of the Management of Aging Effects for the Reactor Vessel Internals RAIs 1 through 14 from December 14, 1998, OG-1744, 2/18/99], the RPV Internals ITG has developed generic programs that address applicable aging effects such as cracking, reduction of fracture toughness, and loss of mechanical closure integrity. The mechanisms that are being considered include irradiation-assisted stress corrosion cracking, stress corrosion cracking, irradiation embrittlement, void swelling, and stress relaxation. The ongoing EPRI Joint Baffle Bolt (JoBB) Program, established in 1996, has been incorporated into the RPV Internals ITG to provide plant inspection and research test data. After this data is available, the ITG will prepare a white paper that summarizes available void swelling data and determines the effect on reactor vessel internals components. In addition, the ITG plans to work with Dr. Frank Garner (Battelle, PNL) to ensure that the applicability of void swelling to PWR reactor vessel internals is appropriately characterized. The JoBB Program and other international programs will examine highly irradiated PWR component items as well as research test reactor specimens for evidence of swelling. The B&WOG has supplied solution annealed Type 304 baffle bolt material and EdF has supplied solution annealed Type 304 baffle plate material for irradiation in this program. Swelling data will be acquired and industry assessments will be completed under both the ITG and JoBB programs. The B&WOG will utilize data and information from these programs to perform design-specific analyses (i.e., safety evaluations), as necessary.

A status report of the B&WOG RV Internals Program, which included information regarding the above industry programs, was provided to the NRC in May 1999 [Status Report of B&W Owners Group Reactor Vessel Internals Program, OG-1753, May 10, 1999]. The status report (see the discussion under stress relaxation and irradiation embrittlement issues in Section 3) indicates that tasks associated with void swelling began in 1998; however, the majority of the activities are scheduled to begin in 2001. Sufficient data should be available in the 2003-2004 timeframe to determine whether void swelling could impact the intended function of the internals. Until this time, the B&WOG will continue to participate in and follow industry activities and evaluate research data regarding void swelling.

Based on currently available research data, there are no RV internals critical locations that may be adversely impacted by void swelling. Absent any evidence to the contrary, the B&WOG believes that the current ASME Section XI inspection requirements (Examination Category B-N-3, VT-3) for the internals and normal refueling operations would be effective in determining if gross deformation by void swelling of the baffle and former plates were occurring.

Errata Regarding Response to Open Item 3

The incore guide tube assembly spider castings, which are fabricated from A351 Grade CF8, were omitted from the list of CASS items in the B&WOG response to DSE Open Item 3. The aging management program for incore guide tube assembly spider castings will be consistent with the position on other CASS items in BAW-2248. That is, the B&WOG agrees that a limited augmented inspection of a sample population of incore guide tube assembly spider castings is appropriate to manage reduction of fracture toughness during the period of extended operation. Details of the augmented inspection (e.g., sample size using risk-based methods, examination method, and acceptance criteria) will be developed by the RVIAMP. Additional details (e.g., timing) of the plant-specific augmented inspections will be addressed at the time of license renewal application.

Should data or evaluations from the MRP or any other industry research activity indicate that the augmented inspections of the spider castings can be modified or eliminated, justification will be provided to the staff demonstrating the basis for its modification or elimination.

Please call me at 804/832-2783 if you need any additional information.

Sincerely,

W. R. Gray

W. R. Gray
Project Manager
B&W Owners Group Services

WRG/mcl

c: w/Attachment

R. K. Anand	- US NRC/NRR
R. L. Gill	- Duke Energy Corporation
G. D. Robison	- Duke Energy Corporation
J. D. Gilreath	- Duke Energy Corporation
D. E. Whitaker	- Duke Energy Corporation
G. G. Young	- Entergy Operations, Inc.
R. W. Clark	- Entergy Operations, Inc.
D. F. Spond	- Entergy Operations, Inc.
D. J. Masiero	- GPU Nuclear, Inc.
G. L. Lehmann	- GPU Nuclear, Inc.
Z. B. Fu	- GUP Nuclear, Inc.
B.W. Doroshuk	- Balitmore Gas & Electric
M. A. Rinckel	- Framatome Technologies
R. N. Edwards	- Framatome Technologies
S. Fyfitch	- Framatome Technologies
F. M. Gregory	- Framatome Technologies
B. C. Cardona	- Framatome Technologies

Table 2 from BAW-10008, Part 1, Revision 1, entitled "REACTOR INTERNALS STRESS AND DEFLECTION DUE TO LOSS-OF-COOLANT ACCIDENT AND MAXIMUM HYPOTHETICAL EARTHQUAKE," June 1970

Table 2. Deflection Summary

Component	Safety implication	Deflection: Allowable, in.	Deflection: No loss of function, in.	Deflection: Calculated, in.
Core support shield	Mode 1 — Outward deflection of the shell will reduce the effectiveness of the internals vent valves, resulting in increased probability of uncovering the core during blowdown.			
	a. Uniform radial expansion of the shell at the vent valve.	1/4	3/8	1/64
	b. Outward local radial displacement of two valves (elliptical deformation of the shell).	1/2	1	3/16
	Mode 2 — Inward deflection of the shell at the outlet nozzles to prevent contact with guide assemblies.	1	1-15/16	Negligible
	Mode 3 — Deformation limit of the upper flange to ensure that the core support assembly does not drop.			
	Uniform decrease in diameter.	1	1-7/8	<1/32
	Mode 4 — Limit axial elongation of the core support shield to ensure engagement of the fuel assemblies in the grid plates.	(See note 1)		+1/32
Core barrel	Mode 1 — Decrease in diameter (local or average to prevent distortion of fuel assembly spacer grids.	3/4	15/16	13/64[a]
	Mode 2 — Limit axial elongation of the core barrel to ensure engagement of the fuel assemblies in the grid plates.	(See note 1)		+1/32
Upper plenum assembly	Mode 1 — Limit radial expansion to maintain clearance between the shell and the internals vent valve wedge ring to ensure valve operation.	1-1/2	3	1/8[b]
	Mode 2 — Limit radial compression to maintain clearance between the shell and the upper guide tube structure.	2-7/16	4-7/8	Negligible
	Mode 3 — Bending of the cover, as a plate is limited to ensure engagement of the internals grid plates and the fuel assemblies.	(See note 1)		−3/8

[a] Includes 0.062-inch uniform decrease in barrel diameter plus 0.146-inch local inward deflection of baffles across a diameter.

[b] Includes 0.011-inch radial expansion of the upper plenum assembly and 0.070-inch decrease in diameter of the core support shield.

Table 2. (Cont'd)

Component	Safety implication	Deflection: Allowable, in.	Deflection: No loss of function, in.	Deflection: Calculated, in.
Control rod guide assembly	Mode 1 — Limit axial deflection to ensure engagement of the fuel assemblies in the upper grid plate.	(See note 1)		Negligible
	Mode 2 — Bending as a beam—contact between the guide tube and the control rod resulting from deflection of the guide—tube must be limited so that the resultant frictional drag on the control rod will be small enough to permit control rod insertion.	1/4	1/2	1/16
	Mode 3 — Cross-sectional distortion of individual tubes is limited to maintain clearance between the guide tubes and the control pins.	0.014	0.029	Negligible
	Mode 4 — Bending as a beam of individual tubes is limited to maintain clearance between the guide tubes and the control pins.	0.022	0.045	0.017
Fuel assembly guide tube	Mode 1 — The cross-sectional distortion of guide tubes is limited to maintain clearance between the tube and the control pin.	0.013	0.027	Negligible
Lower grid plate assembly	Mode 1 — Downward deflection as a plate is limited to ensure engagement of the fuel assemblies in the grid plates.	(See note 1)		+15/64
Thermal shield	No safety implication.			
Flow distributor	No safety implication.			
Note 1	The combined axial displacement of the lower grid plate, the core barrel, the core support shield, the upper plenum assembly, and the control rod guide assembly is limited to prevent disengagement of the fuel assemblies from the grid plates.	1	1-3/4	+19/64[c] -3/8[c]

[c]Sum of calculated deflections.
+ implies separation at the upper grid plate.
− implies separation at the lower grid plate.