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

August 2, 1999

ORGANIZATION: Babcock and Wilcox Owners Group (B&WOG)

SUBJECT: SUMMARY OF JULY 19, 1999, MEETING WITH B&WOG TO DISCUSS OPEN ITEMS IN THE DRAFT SAFETY EVALUATION (DSE) FOR BAW-2248-REACTOR VESSEL INTERNALS

On July 19, 1999, representatives of B&WOG and Duke Energy Corporation (Duke) met with the Nuclear Regulatory Commission (NRC) staff to discuss the open items in the DSE for topical report BAW-2248-Management of Aging Effects for the Reactor Vessel Internals. A list of meeting attendees is provided as Enclosure 1. The meeting notice provided a list of issues to be discussed at the meeting. B&WOG presentation materials discussed at the meeting are provided as Enclosure 2.

The staff on May 26, 1999, issued a DSE for BAW-2248 Reactor Vessel Internals. This DSE identified five Topical Report Open Items and six Renewal Applicant Action Items. By letter dated June 29, 1999, the B&WOG addressed each topical report open item and recommended changes to the appropriate sections of BAW-2248. In the meeting, B&WOG discussed DSE Open Item No. 1 concerning a change in dimension that would be required before the reactor vessel internals would not be able to meet its intended function. For reactor vessel internals items that will receive high fluence, such as the baffle plates and baffle bolts, the B&WOG representatives stated that any significant effect of void swelling would most likely lead to cracking of the baffle bolts. B&WOG discussed the aging management programs proposed for baffle bolts and baffle plates that will provide adequate assurance that the void swelling will not compromise the ability of the reactor vessel internals to perform its intended function in the period of extended operation. The staff provided a report to B&WOG prepared by the Argonne National Laboratory dated August 31, 1998, concerning a preliminary assessment for the review of void swelling of reactor internals for license renewal. This is provided as Enclosure 3. The B&WOG requested that the staff redirect this issue to the Nuclear Energy Institute and resolve the issue on an industry wide basis. In addition, B&WOG committed to revise their response on void swelling based on the discussion with the staff at the meeting.

Duke provided a table which addressed the six renewal applicant action items. This is attached as Enclosure 4. Duke discussed the proposed Oconee responses to BAW-2248 renewal applicant action items concerning reactor vessel internals and cast austenitic stainless steels (CASS). Duke requested a separate meeting with the staff to discuss the management of the aging effects of thermal embrittlement, and possibly synergistic effects with neutron embrittlement, on the fracture toughness of CASS reactor vessel components. The staff agreed to expedite issuing a position on generic renewal issue 98-0030 "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components". Duke provided a table giving a comparison of renewal applicant action items and open items listed in BAW-2248 to open items on the Oconee License Renewal Safety Evaluation Report. This table also contains proposed Oconee-specific responses to these open items. The staff stated that it would provide feedback to Duke on the

1/1

DF03

PROJ 683

REVGP PROJ 683

sufficiency of Oconee-specific responses contained in the table. This table is attached as Enclosure 5.

Original Signed By

Raj K. Anand, Project Manager
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project No. 683

Attachments: As stated

Distribution: See next page

DOCUMENT NAME:G:\RLSB\ANAND\sumjuly19mtg.wpd

OFFICE	:LA	RLSB	DE:DD	RLSB:BC
NAME		RKAnand RAnand	RHWessman	CIGrimes
DATE	7/26/99	7/26/99	7/28/99	8/2/99

OFFICIAL RECORD COPY

Project No. 683

Babcock & Wilcox Owners Group
Generic License Renewal Program

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

AGENDA LIST
JULY 19, 1999
NRC MEETING WITH BABCOCK AND WILCOX OWNERS GROUP (B&WOG)
BAW-2248
(10 CFR PART 54)

NAME	ORGANIZATION
1. Raj Anand	NRC/NRR/DRIP/RLSB
2. Keith Wichman	NRC/NRR/DE/EMCB
3. Allen Hiser	NRC/NRR/DE EMCB
4. Joe Sebrosky	NRC/NRR/DRIP/RLSB
5. David Terao	NRC/NRR/DE/EMEB
6. Francis Grubelich	NRC/NRR/DE/EMEB
7. Stephen Fyfitch	Framatone Technologies, Inc.
8. Jeff Gilreath	Duke Energy Corporation (Duke)
9. Greg Robison	Duke
10. Bob Gill	Duke
11. David Forsyth	Westinghouse Electric Company
12. Bill Gray	Framatone Technologies, Inc.
13. Bill Macay	Energy Operations, Inc.
14. David Whitaker	Duke
15. Frank Gregory	Framatone Technologies
16. Mark Rinkel	Framatone Technologies
17. Dick Wessman	NRC/NRR/DE
18. Lois Lambros	NRC/NRR/DSSA/SRXB
19. Sam Lee	NRC/NRR/DRIP/RLSB

Enclosure 1

U.S. NUCLEAR REGULATORY COMMISSION MEETING

DUKE ENERGY/B&W OWNERS GROUP

Status of Oconee-Specific Responses to BAW-2248 Items and Draft SER Open Items Concerning Reactor Vessel Internals and Cast Austenitic Stainless Steels

One White Flint North
Rockville, Maryland

Enclosure 2

INTRODUCTION AND PURPOSE OF MEETING

R. (Bob) L. Gill

Duke Energy Corporation

AGENDA

Time	Item	Presenter
1:00 PM	Introduction and Purpose of Meeting	R.L. Gill, Duke Energy Corporation
1:05 PM	Status of Industry and B&WOG RV Internals Programs	S. Fyfitch, Framatome Technologies, Inc.
1:40 PM	Applicant Action Items and DSE Open Items	R.L. Gill and J.D. Gilreath, Duke Energy Corporation M.A. Rinckel, Framatome Technologies, Inc.
2:45 PM	Open Discussion/NRC Comments	All
3:00 PM	Adjourn	

PURPOSE OF MEETING

- Provide the status of the B&WOG RV Internals Aging Management Program and recent Industry program changes
- Discuss the Oconee-specific responses to BAW-2248 Items and the Draft SER Open Items concerning RV Internals and cast austenitic stainless steels (CASS)

STATUS OF INDUSTRY AND B&WOG RV INTERNALS PROGRAMS

S. (Steve) Fyfitch

Framatome Technologies, Inc.

RV INTERNALS PROGRAMS BACKGROUND

- B&WOG program initiated (1996)
  - To evaluate baffle bolt cracking observed in French PWRs
  - To determine its relevance to the B&W-designed plants
- B&WOG action plan prepared (1997)
  - Comprehensive program to manage RV internals aging effects
  - Reactor Vessel Internals-Aging Management Program (RVI-AMP) identified in BAW-2248

RV INTERNALS PROGRAMS BACKGROUND

- Industry-wide PWR Materials Reliability Program (MRP) developed (1998)
  - To serve as industry focal point for resolution of generic issues related to PWR materials degradation management
  - Created an Issue Task Group (ITG) to manage RV Internals aging issues

RV INTERNALS PROGRAMS BACKGROUND

- Details of the B&WOG RVI-AMP presented to NRC during B&WOG meeting (April 23, 1998)
- MRP and industry RV Internals/JoBB ITG presented to NRC by EPRI during WOG meeting (March 25, 1999)
- B&WOG RVI-AMP Status report sent to NRC (May 10, 1999)

RV INTERNALS PROGRAMS ISSUES SUMMARY

- Aging effects identified in BAW-2248
  - Cracking
    - Irradiation-assisted stress corrosion cracking
    - Stress corrosion cracking
    - Fatigue(TLAA)
  - Reduction of fracture toughness
    - Irradiation embrittlement
    - Thermal embrittlement

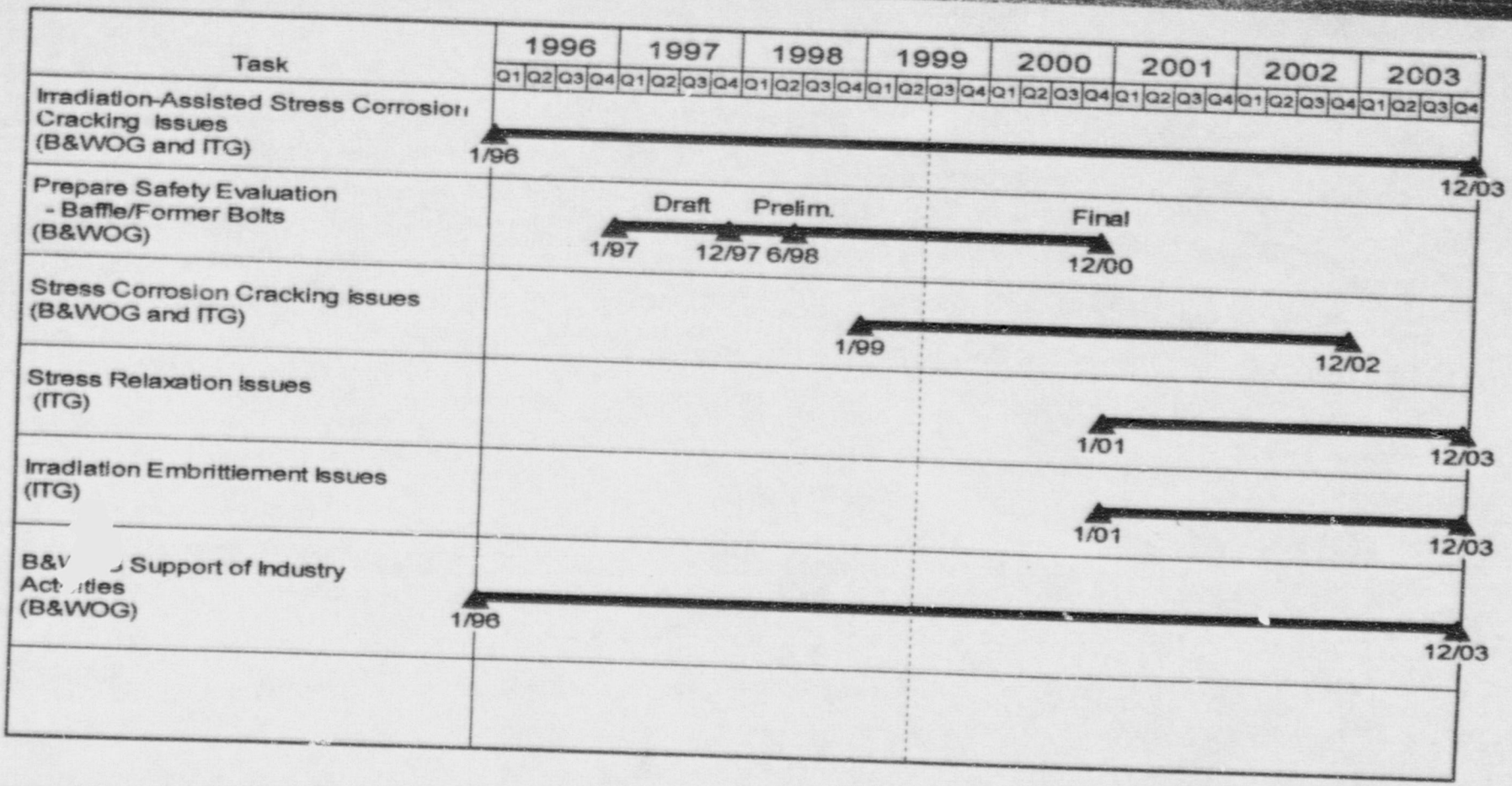
RV INTERNALS PROGRAMS ISSUES SUMMARY

- Aging effects identified in BAW-2248
  - Loss of preload
    - Stress relaxation (includes irradiation creep)
  - Loss of material
    - Wear
  - Dimensional changes (void swelling) [NRC DSER Open Item 1]

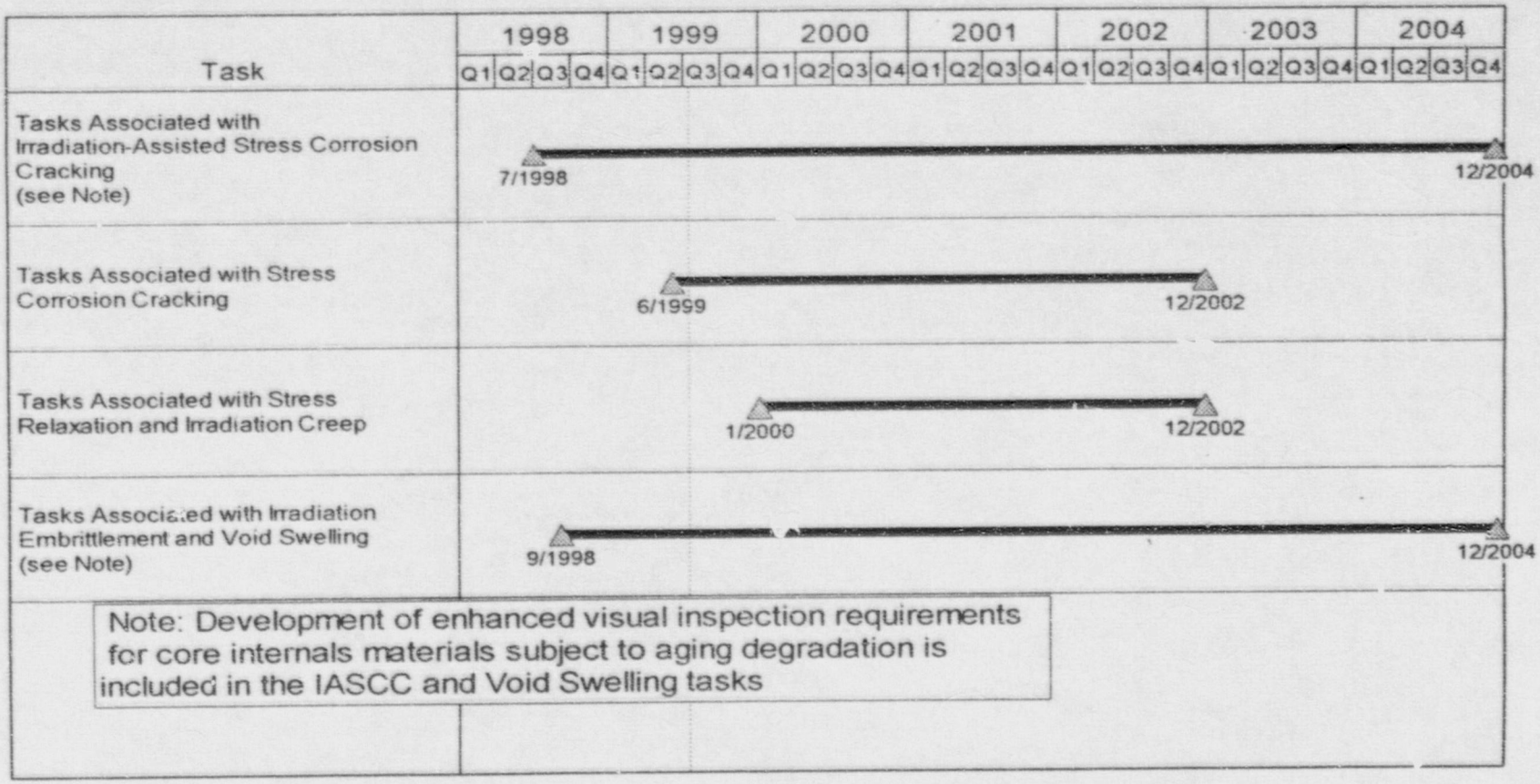
RV INTERNALS PROGRAMS ISSUES SUMMARY

- B&WOG and ITG Program tasks currently identified for the following issues
  - Irradiation-assisted stress corrosion cracking
  - Stress corrosion cracking
  - Irradiation embrittlement
  - Void swelling
  - Stress relaxation (including irradiation creep)

B&WOG RVI-AMP SCHEDULE



INDUSTRY RV INTERNALS/JoBB ITG PROGRAM



BAFFLE BOLT INSPECTIONS

- BAW-2248 DSER Applicant Action Item 4
- Three U.S. industry inspections (Farley-1, Point Beach-2, and Ginna)
- ITG/JoBB program activities to provide data for RVIAMP

BAFFLE BOLT INSPECTIONS (Continued)

- **B&WOG safety assessment**
  - Preliminary assessment submitted to NRC in 1998
  - Ongoing stress analyses scheduled for completion in 2000
  - Final assessment scheduled for late 2000
  - Results used to determine sample size and acceptance criteria for inspections, if necessary
- **Baffle bolt inspection methods to be developed by 2003**

LOSS OF DUCTILITY

- BAW-2248 DSER Applicant Action Item 6
- High fluence (~5 X 10^{20} n/cm^2, E > 1 MeV) mechanical properties of annealed Type 304
  - Data obtained from ITG/JoBB program
- B&WOG update of BAW-10008
- Locations of high stress intensity of most concern (e.g., upper and lower core support shield flanges)

VOID SWELLING

- BAW-2248 DSER Topical Report Open Item 1
- High fluence data for annealed Type 304
  - Data obtained from ITG/JoBB Program
- Assessment of void swelling impact on internals intended functions
  - Data available in 2003-2004 timeframe
- Baffle bolts and baffle plates are believed to be lead items for this potential issue

AUGMENTED INSPECTIONS OF NON-BOLTING ITEMS

- BAW-2248 DSER Topical Report Open Items 2 and 5
- Data obtained from ITG/JoBB for use in determining necessity of inspection
- ITG task to develop visual inspection for PWR RV Internals
- Complete response in June 29 B&WOG letter to the NRC
- For non-bolting items, baffle plates are lead items (IASCC and IE)

REDUCTION OF FRACTURE TOUGHNESS

- BAW-2248 DSER Topical Report Open Items 3 and 4
- CRGT assembly spacer castings, vent valve bodies and retaining rings, and Oconee-3 RV internals outlet nozzles are affected items (TE and IE aging degradation)
- Augmented inspections to be performed, as necessary
- ITG task to develop visual inspection for PWR RV internals

File: (22)

ARGONNE NATIONAL LABORATORY

9700 SOUTH CASS AVENUE, ARGONNE, ILLINOIS 60439

TELEPHONE 630/252-5127
FAX 630/252-3250

August 31, 1998

PC2783

Dr. C. M. Regan
NRR/DRPM/PDLR
U.S. NRC – Mail Stop 012 G15
11555 Rockville Pike
Rockville, MD 20852-2738

Subject: Preliminary Assessment and List of Queries for Task Order No. 13 of Job Code Number J-2076 entitled "Review of Void Swelling of Reactor Internals for License Renewal"

Dear Dr. Regan:

Enclosed for NRC review and comment is a preliminary assessment and a list of queries prepared by Dr. M. C. Billone on the "Review of Void Swelling of Reactor Internals for License Renewal," Task Order 13 of J-2076. .

Please contact either Dr. Billone (x7146) or me If you have any questions regarding the preliminary assessment and the list of queries.

Sincerely,

Yung Y. Liu
Energy Technology Division

xc: R. Brady

Enclosure 3

xc: D. T. Raske
O. K. Chopra
D. R. Diercks
H. C. Tsai
R. B. Poeppel
R. A. Valentin
W. J. Shack

Preliminary Assessment and List of Queries for Task Order No. 13 (JCN J-2076)

REVIEW OF VOID SWELLING OF REACTOR INTERNALS FOR LICENSE RENEWAL

Industry Position

The Industry position on the issue of void swelling of reactor internals is summarized in document TR-107521 prepared for EPRI in April 1998. Twelve references are cited: the first four are used to acknowledge the issue of "...'change in dimension' that must be considered in the license renewal application for baffle/former assembly components in Westinghouse and B&W plants, and core shroud assembly components in CE plants"; the other eight references are technical references which address the modeling and database for void swelling of austenitic stainless steels. Based on the information presented in these references, the Industry concludes that void swelling (dimensional changes) is "...insignificant for the license renewal term, and that no further evaluation is required." They add that this conclusion "...is particularly true for plants that have adopted a low-leakage fuel loading pattern, in-lieu of an in-out fuel replacement strategy, that reduces the end-of-life exposure by about 50%." The Industry conclusion is based on the following four points listed on page 4-66 of TR-107521:

(1) absence of evidence from current service experience that void swelling is significant

(2) the results from laboratory studies of Foster, et al.

(3) the uncertainty of applying LMFBR data to predict void selling in LWR environments

(4) continuing studies

Preliminary Assessment

During the past 30 years of material science and engineering research on the swelling of austenitic stainless steels (particularly Types 304 and 316), theories and models have lagged the evolution of the database. Each new set of data, as well as re-interpretations of previous data, generates new physical interpretations, models and correlations for the swelling behavior of these materials. In TR-107521 Ref. 5, Garner discusses in detail (over 60 pages) the evolution of the theories, models and database associated with stainless steel swelling. In particular, he re-interprets previous data to demonstrate why misconceptions persisted in this area. In subsequent publications (e.g., TR-107521 Ref. 9), he uses newer data to contradict some of the conclusions he documented in Ref. 5. In trying to assess the degree of importance of the swelling issue for austenitic stainless steel LWR reactor internal components, it is essential to distinguish what is well-established in this area vs. what is pure speculation and/or large extrapolation from the existing database.

Garner [5] presents convincing evidence that the steady-state swelling rate of Types 316 and 304 stainless steels is ≈1 vol.%/dpa within a temperature range which may be dependent on the dose rate. This conclusion appears to be valid for a wide range of elemental impurities and/or solute additions, heat treatments, damage rates, temperatures within the swelling temperature window, and temperature history. Based on fast-reactor data, the steady-state swelling rate is characterized by three regimes: an increase from zero swelling at ≈350°C to ≈1 vol.%/dpa at 430°C; uniform steady-state swelling rate of ≈1

vol.%/dpa from ≈430°C to ≈700°C; and a decrease in swelling from ≈1 vol.%/dpa at ≈700°C to ≈800°C. In a subsequent publication [9], Garner suggests that there may be a downward shift of the swelling temperature boundaries for LWR austenitic steel components exposed to low damage rates over long periods of time. The steady-state swelling rate vs. temperature in a low damage rate environment remains an area of speculation because the current database is not extensive enough, particularly for the Type 316 and 304 alloys used in commercial U.S. LWRs.

More uncertainty is involved in predicting austenitic stainless steel swelling in the transition region (also called transient regime and incubation regime) from zero swelling to steady-state swelling. Small variations of impurities/solute-additions (e.g., Ti, Si, Zr, and P), heat treatments, operating temperatures, and transmutation products can have a large impact on the transition region swelling vs. dpa. Combining data from different heats and heat treatments of Types 316 and 304 stainless steels in order to determine swelling trend curves in the transition region can lead to invalid conclusions and inaccurate extrapolations.

With the background established in the previous two paragraphs, a preliminary assessment of the technical basis for the Industry position on void swelling (dimensional change) is presented as follows:

(1) Absence of evidence from current service experience that void swelling is significant

The results of field service experience are encouraging in that only limited swelling (< 1% linear change, < 3% volumetric change if swelling is isotropic) has been observed in austenitic steel PWR internal components (e.g., Type 304 baffle/former plates and Type 316 baffle former bolts) for damage levels up to about 50 dpa. However, the field experience is limited to visual examinations, along with feeler gauge measurements. Because the components are in-service, no destructive examinations have been performed to determine the extent of helium bubble and void formation which would give some indication of the components to increase in swelling rate per dpa at higher damage levels. The extrapolation of the field experience from 50 to 150 dpa is not meaningful. Other uncertainties associated with the field experience have to do with the dpa level, the dose rate history, and the temperature history of the examined components. If the Industry estimations are correct, the shift from an "in-out" fuel replacement to a "low-leakage" fuel management system may have resulted in lower gamma heating rates, lower dose rates, and lower dpa values. Without explicit values for the temperature history, dose rate history and dpa level, it is difficult to correlate the field service observations with swelling-dependent operating parameters.

(2) The results from laboratory studies of Foster, et al. [12]

The Foster et al. [12] data are based on cw316SS thimble tubes irradiated in a PWR for eleven effective full power years at 310±5°C to a damage level of 35 dpa and a helium level of 208 appm. While small cavities (≈ 1 nm in diameter) were observed, which were most likely finely distributed He bubbles, the swelling was estimated from thin foil samples to be 0.03 vol.%. This estimation has a high degree of uncertainty. The data from this study are at too low a damage level to allow meaningful extrapolation to damage levels of 100-150 dpa. The authors combine their data with HFIR (mixed spectrum), FFTF (fast spectrum) and EBR-II (fast spectrum) data on 20%cw316SS, along with data on a Ti-modified 20%cw316SS (PCA). The two alloys differ considerable in Ti content and Si content, as well as in levels of Ni and Cr. The mixing of the data from different alloys, along with the use of higher temperature HFIR and EBR-II data, to formulate their correlation for extrapolation of swelling at 100 dpa is not convincing, although the authors have done the

best that they could with the data available. In summary, the use of the Foster, et al. correlation to extrapolate from damage levels of 35 dpa to 100-150 dpa is neither convincing nor meaningful.

(3) The uncertainty of applying LMFBR data to predict void selling in LWR environments

The reviewer agrees that there is a high degree of uncertainty in extrapolating higher-temperature/higher-dose-rate LMFBR swelling data to lower-temperature/lower-dose-rate LWR environments. However, this point undermines Item 3 above, as Foster, et al. relied heavily on EBR-II data and 400-440°C HFIR data in the formulation of their correlation relating swelling to dpa. In essence, the data of Foster, et al. at ~310°C and 35 dpa and the HFIR data at 300-340°C and 40-60 dpa represent a data set with very low values of swelling, large scatter, and relatively high uncertainty. The reviewer believes that data on the specific types of cw316SS and 304SS used in commercial power plants are needed at lower damage rates, higher damage levels, and relevant temperatures (300-400°C). The use of data from different heats and types of austenitic stainless steel irradiated at a wide range of damage rates, damage levels, and temperatures to form a master swelling curve for the transient swelling regime does not lead to a correlation which can be extrapolated with any degree of confidence to the 100-150 dpa range.

(4) Continuing studies

This reviewer is in agreement with Industry's recommendation for continued studies: "However, given the uncertainties involved in predicting the eventual amount of void swelling for the most affected internals components, the industry would be prudent to follow or participate in research and development activities that may produce information that will determine whether or not swelling would be a potentially significant issue for PWRs in the future." Given this statement, it is not clear how a recommendation for closure can be made at this time: "Reference 4 should be modified to find that the effect of void swelling, which is referred to as "dimensional changes" is insignificant for the license renewal term, and that no further evaluation is required."

List of Queries and Requests for Additional Information

1. The following information are needed to assess the application of the swelling database to the LWR reactor internals:

   1.1 What was the composition and amount of cold work of the cw316SS used by Foster, et al.?

   1.2 What is the level of cold work and what is the composition of the 316SS used for PWR baffle/former plate bolts?

   1.3 What is the composition of the 304SS used in the baffle/former plates?

2. For the "in-out" fuel replacement management previously used, Garner, et al. [8] calculate peak metal temperatures of baffle/barrel/former components under steady state operation in the range of 329-393°C. They also calculate 40-year damage levels as high as 90 dpa, along with He levels of 1350 appm for the cw316SS baffle bolt. What are the Industry-calculated operating conditions (including uncertainties) for these components?

3. Please provide a draft copy of Reference 10 which is listed as "in press" in TR-107521 (April 1998). Please also provide a draft copy of Reference 11, which has neither been presented nor published.

4. If the current database is insufficient to extrapolate to a 40-year lifetime, what is the basis for concluding that void swelling is a non-issue for operation up to 60 years?

5. Please provide more information on plans for the use of "low-leakage" fuel management for PWRs during the license renewal period? What will be the mix of "in-out" vs. "low-leakage" fuel management schemes during the license renewal period?

6. Whether void swelling is an issue or not depends greatly on the operating temperatures and damage rates during the license renewal period. Please supply Industry-calculated damage rates, helium generation rates and temperatures for austenitic steel internals of PWRs with the "low-leakage" fuel management system.

References Listed in TR-107521 for Evaluation of Swelling Issue for Austenitic Steel LWR Internals

[5] F.A. Garner, "Irradiation performance of cladding and structural steels in liquid metal reactors," Chapter 6, Nuclear Materials Science and Technology: A Comprehensive Treatment, Volume 10A, VCH Publishers, 1995, pp. 419-543.

[6] F.A. Garner, L.R. Greenwood, and D.L. Harrod, "Potential High Fluence Response of Pressure Vessel Internals Constructed from Stainless Steels," Proceedings, Sixth International Symposium on Environmental Degradation of Materials in Nuclear Power Reactors -- Water Reactors, San Diego, August 1993, pp. 783-790.

[7] F.A. Garner, "Materials Issues Involving Austenitic Pressure Vessel Internals Arising from Void swelling and Irradiation Creep," Transactions, American Nuclear Society, Volume 71, 1994, pp. 190-191.

[8] F.A. Garner, M.B. Toloczko, S.I. Porollo, A.N. Vorobjev, A.M. Dvoriashin, and Yu. V. Konobeev, "Potential Irradiation Creep and Void Swelling of Stainless Steel in PWR Internals," Proceedings, Eighth International Symposium on Environmental Degradation of Materials in Nuclear Power Reactors -- Water Reactors, Amelia Island, August 1997, pp. 839-845.

[9] F.A. Garner and M.B. Toloczko, "Irradiation Creep and Void Swelling o Austenitic Stainless Steel at Low Displacement Rates in Light Water Energy Systems," Journal of Nuclear Materials, Vol. 251, 1997, pp. 252-261.

[10] S.I. Porollo, A.N. Vorobjev, Yu. V. Konobeev, A.M. Dvoriashin, N.I. Budylkin, E.G. Mironova, and F.A. Garner, "Void-Induced Embrittlement of Austenitic Stainless Steel Irradiated to 73-82 dpa at 335-365°C," Journal of Nuclear Materials, in press.

[11] V.S. Neustroev, V.K. Shamardin, Z.E. Ostrvsky, A.M. Pecherin, and F.A. Garner, "Temperature Shift of Void Swelling Observed at PWR-Relevant Temperatures in Annealed Fe-18Cr-10Ni-Ti Stainless Steel Irradiated at High and Low dpa Rates in BOR-60," to be presented at the International Symposium on Contribution of

Materials Investigation to the Resolution of Problems Encountered in Pressurized Water Reactors, September 1998, Fontevraud, France.

[12] J.P. Foster, D.L. Porter, D.L. Harrod, T.R. Mager, and M.G. Burke. "316 Stainless Steel Cavity Swelling in a PWR," Journal of Nuclear Materials, Vol. 224, 1995, pp. 207-215.

Proposed Oconee-Specific Responses to BAW-2248 Items and SER Open Items concerning
Reactor Vessel Internals and CASS
July 19, 1999

BAW- 2248 **Draft Safety Evaluation Items**	**Oconee SER** **Open Items**	**Proposed** **Oconee-Specific Response**
RENEWAL APPLICANT ACTION ITEMS		
(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).	No equivalent SER Open Items	Duke participated in the development of BAW-2248 by providing Oconee-specific design and operational information. Duke has reviewed the current design and operation of the Oconee reactor vessel internals using the process described in Sections 2.4.1 and 2.4.2 of the Application. Duke has determined that the Oconee reactor vessel internals are bounded by the descriptions contained in BAW-2248 with the exception of the thermal shield and thermal shield upper restraint, which is within scope for Oconee. Also see pages 2-20 and 3-87 of the Oconee SER dated June 16, 1999 for resolution of this item.
(2) A summary description of the programs and evaluation of TLAAs is to be provided in the license	SER Open Item 3.0-1 applies to UFSAR Supplement in general	See Exhibit B of the Application for the summary description of the credited programs and activities.

Enclosure 4

Proposed Oconee-Specific Responses to BAW-2248 Items and SER Open Items concerning
Reactor Vessel Internals and CASS
July 19, 1999

BAW- 2248 **Draft Safety Evaluation Items**	**Oconee SER** **Open Items**	**Proposed** **Oconee-Specific Response**
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 with10 CFR 54.21(a)(3).	No equivalent SER Open Items	Section 2.4.6 and Table 2.4-4 of Exhibit A of the Application states that the Oconee reactor vessel internals provide gamma and neutron shielding of the reactor vessels. Also see pages 2-20 and 3-87 of the Oconee SER dated June 16, 1999 which address this topic.
(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	**3.4.3.3-4** A specific RVI component that has demonstrated susceptibility to IASCC (although not specifically in B&W nuclear steam supply systems) is the baffle-former bolts. At the present there are no requirements for supplemental examination of baffle former bolts, and no plans to implement periodic supplemental examinations. This situation may change as several one-time volumetric examination and replacement programs at specific plants are completed and the results are fully analyzed. In response to RAI 12 to topical report BAW-2248, the B&WOG stated that future inspection plans for baffle former-bolts would be on a plant-specific basis, possibly beginning with the inspection at ONS Unit 1	A proposed response to this open item is provided by the draft Baffle Bolt inspection Plan discussed during today's meeting.

BAW- 2248 **Draft Safety Evaluation Items**	**Oconee SER** **Open Items**	**Proposed** **Oconee-Specific Response**
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.	during their fourth inservice inspection (ISI) interval (2003 – 2013). It should be noted that accessibility limitations eliminate visual inspection as a viable alternative for this bolting; a volumetric method is necessary for effective examination. In a February 18, 1999, response to RAI 12 and RAI 13 the B&WOG stated that the renewal applicant would be responsible for using the tools provided by the Issues Task Group (ITG) and the owners groups to determine the necessary steps (e.g., inspections, operability determinations, and replacements) to manage the applicable baffle-bolt aging effects. The ITG on reactor vessel internals is currently addressing the issues of cracking, reduction of fracture toughness, and loss of preload for 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 ONS Unit 1 during their fourth inservice inspection (ISI) interval. Duke did not provide a plant specific plan, therefore, the information requested of the B&WOG about BAW-2248 in a letter dated December 2, 1998, for RAI 12 and RAI 13 with regard to the aging management of the effects of baffle bolt age-related degradation is an Open Item.	
(5) If flaws have been detected in the reactor	**4.2.5.3-2** BAW-2248 also identifies a fourth	Section 5.4.1.2 of Exhibit A of the Application

Proposed Oconee-Specific Responses to BAW-2248 Items and SER Open Items concerning
Reactor Vessel Internals and CASS
July 19, 1999

BAW- 2248 **Draft Safety Evaluation Items**	**Oconee SER** **Open Items**	**Proposed** **Oconee-Specific Response**
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.	TLAA regarding flaw growth acceptance in accordance with the ASME B&PV Code, Section XI ISI requirements. This TLAA is identified in the topical report as requiring a plant-specific evaluation, and as such is not evaluated in the topical report. The applicant does not address the applicability of this flaw growth TLAA to ONS.	discusses this TLAA. Based upon our review of ISI records, no flaws in the RV Internals have been identified. Thus, no flaw growth evaluation needs to be performed for Oconee. Also see page 4-11of the Oconee SER dated June 16, 1999 for the evaluation of this topic.
(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.	**4.2.5.3-1** The TLAA described as "reduction in fracture toughness" is related to the acceptability of the reactor vessel internals under loss-of-coolant-accident (LOCA) and seismic loading. BAW-2248 states that Appendix E to BAW-10008, Part 1, Revision 1, "Reactor Internals Stress & Deflection Due to LOCA & Max Hypothetical Earthquake," 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." BAW-2248 also states 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. Section 5.4.3 of Exhibit A of the LRA states that the RVIAMP will assure that appropriate action will be taken in a timely manner to assure continued validity of the design of the ONS reactor vessel internals. Plant-specific analysis is required to demonstrate that, under LOCA	Design basis event loads (i.e., design, LOCA and seismic) were evaluated in BAW-10008 to determine if deflections due to those loads would lead to loss of internals functions. The effect of design basis loads on the stress and deflection of the internals was evaluated assuming elastic behavior of the materials. Radiation effects will not impact the elastic calculations reported in BAW-10008, Part 1, Revision 1, since the elastic behavior of stainless steel is unaffected by irradiation. In addition, the strength of stainless steel is enhanced with irradiation (i.e., yield strength and ultimate strength increase) and the margins of safety that are reported in BAW-10008 will increase with time (i.e., irradiation). Appendix E of BAW-10008 concludes that the internals will have adequate ductility at the end of a 40-year lifetime to absorb local strain at the regions of maximum stress intensity (i.e., upper and lower core support shield flanges). This conclusion is based on

BAW- 2248 **Draft Safety Evaluation Items**	**Oconee SER** **Open Items**	**Proposed** **Oconee-Specific Response**
	and seismic loading and with irradiation accumulated at the expiration of the period of extended operation, the internals have adequate ductility to absorb local strain at the regions of maximum stress intensity and will meet the deformation limits. The applicant must provide a plan to develop data to demonstrate that the internals will meet the deformation limits through the period of extended operation. The plan must be submitted for staff review and approval.	the assumption that the neutron fluence at the flange locations (upper and lower core support shield) will be less than 10^{20} n/cm^2 at 40 years and the uniform elongation of the material will exceed 20% for a fluence of 10^{20} n/cm^2. The basis for concluding that a uniform elongation greater than 20% will ensure adequate ductility at regions of maximum stress intensity appears to be qualitative as the peak stress intensities, which were compared to 2/3 of the ultimate strength of the material (Su), were calculated assuming elastic behavior of the material. Should a plastic evaluation of internals items be performed, plastic strains would be limited to the strain corresponding to 2/3 Su. As the material is irradiated, ultimate strength increases, uniform elongation decreases, and the plastic strain corresponding to 2/3 Su will decrease. In order to ensure that the analyses in BAW-10008, Part 1, Revision 1, remain valid for the period of extended operation, the regions of highest stress intensity should be reevaluated to ensure that the ductility is sufficient to absorb local strain at the regions of peak stress intensities (i.e., upper and lower flange). The B&WOG has estimated the bounding 48 EFPY fluence to be 5.0E20 n/cm2; this fluence should bound the 48 EFPY fluence at the upper flange. The RVIAMP will develop material property data (i.e., uniform elongation, yield strength, and ultimate strength) for Type 304 material irradiated

BAW- 2248 **Draft Safety Evaluation Items**	**Oconee SER** **Open Items**	**Proposed** **Oconee-Specific Response**
		to a fluence of 5.0E20 n/cm². Once the data is available, the regions of peak stress intensities (i.e., upper and lower core support shield flanges) will be evaluated to ensure that adequate ductility is available at the end of the period of extended operation to absorb local strain.
4.2 TOPICAL REPORT OPEN ITEMS		
No equivalent BAW-2248 DSE Item	**3.4.3.2-1** The LRA states that the aging effect for the spray head is cracking due to reduction in fracture toughness. The staff does not agree since reduction in fracture toughness does not cause cracking. Reduction in fracture toughness causes cracked components to fail at lower stresses than they otherwise would fail, but reduction in fracture toughness is not the cause of the cracking. The staff believes that the aging effects for the spray head are cracking and reduction in fracture toughness due to thermal aging of the cast stainless steel. Until Duke has responded to this apparent discrepancy, the staff cannot conclude that Duke has properly identified the potential aging effects for the heater bundle penetration welds, cladding, spray line and spray head.	Duke agrees with the staff that the applicable aging effects for the pressurizer spray head are cracking by fatigue and reduction of fracture toughness by thermal embrittlement.

Proposed Oconee-Specific Responses to BAW-2248 Items and SER Open Items concerning
Reactor Vessel Internals and CASS
July 19, 1999

BAW- 2248 **Draft Safety Evaluation Items**	**Oconee SER** **Open Items**	**Proposed** **Oconee-Specific Response**
(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.	**3.4.3.2-2** Section 3.1 of topical report BAW-2248 dismisses changes in dimensions of the RVI components due to void swelling as a significant aging effect because there is no of evidence of void swelling under PWR conditions. However, EPRI TR-107521 "Generic License Renewal Technical Issues Summary," cites several sources with different estimates of void swelling. One source predicts swelling as great as 14 percent for PWR baffle-former assemblies over a 40-year plant lifetime, whereas another source states 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 effect of change of dimensions due to void swelling on the ability of the RVI to perform their intended function. Duke must provide the basis for concluding that void swelling is not an issue for RVI or must provide an AMP.	Duke is awaiting the staff response to BWOG letter dated June 29, 1999 and the BAW-2248 FSE prior to providing a response to this open item.
(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	**3.4.3.3-3** For detection of cracking, the NRC staff proposed to the B&WOG a modified approach to manage cracking of RVI non-bolting components. This approach involves a supplemental (enhanced VT-1) examination of the components believed to be the limiting components for cracking, considering the susceptibility of the components to the aging	Duke is awaiting the staff response to BWOG letter dated June 29, 1999 and the BAW-2248 FSE prior to providing a complete response to this open item.

Proposed Oconee-Specific Responses to BAW-2248 Items and SER Open Items concerning
Reactor Vessel Internals and CASS
July 19, 1999

BAW- 2248 **Draft Safety Evaluation Items**	**Oconee SER** **Open Items**	**Proposed** **Oconee-Specific Response**
(excluding RVI bolting).	mechanisms the material properties of the components (in particular the fracture toughness), and the operating stresses on the components. 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. 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. Duke has not identified the limiting components and incorporate this program into the ISI program.	A partial response to this open item is provided by the draft RVI Non-Bolting Components Inspection Plan discussed during today's meeting.
(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."	**3.4.3.3-5** 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. To account for this synergistic loss of fracture toughness, a modified approach for CASS RVI components is proposed. This modified approach would involve either a supplemental (enhanced VT-1) examination of the affected components as part of Duke's 10-year ISI program during the period of extended operation or a component-specific evaluation to determine the susceptibility to loss of fracture toughness. For the component-specific evaluation refer to "Embrittlement of CASS RVI Components" in Section 3.4.3.3 of this SER.	Duke is awaiting the staff response to BWOG letter dated June 29, 1999 and the BAW-2248 FSE prior to providing a response to this open item.

BAW- 2248 **Draft Safety Evaluation Items**	**Oconee SER** **Open Items**	**Proposed** **Oconee-Specific Response**
	In addition, to determine whether CASS components are above or below the effective threshold value of 1×10^{17} n/cm^2, discussed above, Duke 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. Note: In addition to the above Section 3.4.3.3 also contains the following references to this open item: (2) Open Item 3.4.3.3-5 described under the heading "Embrittlement of CASS RVI Components" also applies to valve bodies in the RCS piping. The reduction in fracture toughness due to thermal embrittlement is a significant factor in cast stainless steel components that do not satisfy the criteria specified by the staff in this section (criteria are discussed under "Embrittlement of CASS RVI Components"). This is part of Open Item 3.4.3.3-5, which is discussed later in this section. If the ONS pressurizer spray heads do not satisfy these criteria, they could be subject to significant thermal embrittlement and the proposed examination may require an enhanced VT-1 examination.	

Proposed Oconee-Specific Responses to BAW-2248 Items and SER Open Items concerning
Reactor Vessel Internals and CASS
July 19, 1999

BAW- 2248 **Draft Safety Evaluation Items**	**Oconee SER** **Open Items**	**Proposed** **Oconee-Specific Response**
No equivalent BAW-2248 DSE Item	**3.4.3.3-6** Besides visual inspection (VT-3) in accordance with Examination Category B-N-3 of the ASME Code Section XI inservice inspection program, the response to RAI 5 on the topical report cited by the LRA, BAW-2248, states that 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 the Pump and Valve In-Service Test Program at ONS Units 1, 2, and 3. A description of this program must be included in the LRA to allow an evaluation. The vent valve retaining rings (fabricated from precipitation-hardened stainless steel) should be subject to supplemental (enhanced VT-1) examination. This examination could be modified or eliminated, provided that Duke 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 use the same framework as proposed for CASS RVI components.	The reactor vessel internals vent valves allow flow out of the core to the broken cold leg in the event of a cold leg LOCA to prevent pressure from building in the core and restricting core cooling. The vent valves open on differential pressure. The primary means to inspect the vent valves at Oconee is the general visual inspection required by Examination Category B-N-3. The Oconee Pump and Valve Inservice Test Program implements the requirements of 10 CFR 50, §50.55a. The reactor vessel internals vent valves are identified in the program. The valves are self-actuated check valves. A full-stroke test of each valve is performed every refueling outage. The plant procedure that implements the vent valve exercise test also requires that each valve seat be inspected for visible damage. Duke is awaiting the staff response to BWOG letter dated June 29, 1999 and the BAW-2248 FSE prior to providing a response to the request for a supplemental examination.
No equivalent BAW-2248 DSE Item	**3.4.3.3-7** The loss of fracture toughness in CASS is caused by thermal embrittlement at reactor operating temperatures. EPRI topical report TR-106092 discusses the effect of thermal embrittlement on CASS and describes a program for detecting the loss of fracture toughness. The staff has reviewed this	Duke is not clear how the criteria beyond those provided in EPRI TR-106092 are to be applied at Oconee. In order to assess how these criteria may relate to the process Duke has used, please identify the reference for these additional criteria.

Proposed Oconee-Specific Responses to BAW-2248 Items and SER Open Items concerning
Reactor Vessel Internals and CASS
July 19, 1999

BAW- 2248 **Draft Safety Evaluation Items**	**Oconee SER** **Open Items**	**Proposed** **Oconee-Specific Response**
	topical report and concluded that CASS components must be evaluated to the criteria in EPRI TR-106092 and the additional criteria described previously under the heading "Embrittlement of CASS RVI Components." (This applies to the reactor coolant pump casing.)	The process Duke used to consider the applicability of loss of fracture toughness for CASS components does not involve screening criteria. The screening portion of EPRI TR-106092 is not being applied at Oconee; rather, all Class 1 CASS components are considered subject to loss of fracture toughness and managed via ASME Section XI examination and the modified analysis methods described in EPRI TR-106092.
(4) To determine whether CASS components are above or below the threshold neutron fluence value of 1×10^{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.	SER Open Item 3.4.3.3-5 is equivalent (see above)	Duke is awaiting the staff response to BWOG letter dated June 29, 1999 and the BAW-2248 FSE prior to providing a response to this open item.
(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).	SER Open Item 3.4.3.3-3 is equivalent (see above)	Duke is awaiting the staff response to BWOG letter dated June 29, 1999 and the BAW-2248 FSE prior to providing a response to this open item.

DRAFT INSPECTION PROGRAM FOR OCONEE REACTOR VESSEL INTERNALS JULY 19, 1999		
Program and Activity Attributes (from Section 4.2 of the Oconee Application)	**Draft Reactor Vessel Internals Baffle Bolt Inspection**	**Draft Inspection of Reactor Vessel Internals Non-Bolted Component Items**
Purpose – A clear statement of the reason why the program or activity exists for Oconee license renewal.	The purpose of the Reactor Vessel Internals Baffles Bolt Inspection will be to confirm the required number of baffle bolts remain functional, ensuring the functionality of the reactor vessel internals. BAW 2248 Safety Evaluation Report, Renewal Action Item number 4. Oconee's SER open item 3.4.3.3-4. Should data or evaluations from any industry research or inspection activity indicate that the inspection can be modified or eliminated, justification will be provided to the staff demonstrating the basis for its modification or elimination.	The purpose of the Reactor Vessel Internals Non-Bolted Component Items Inspection will be to ensure aging effects are not adversely affecting the functionality of the reactor vessel internals. BAW-2248, Open Items DSE 2& 5. Oconee's SER open item 3.4.3.3-3. Should data or evaluations from any industry research or inspection activity indicate that the inspection can be modified or eliminated, justification will be provided to the staff demonstrating the basis for its modification or elimination.
Scope – A description of the set of Oconee structures and components encompassed by the actions of the program or activity.	Reactor Vessel Internals Baffle Bolts for Oconee Units 1, 2 and 3.	Oconee Units 1, 2 and 3 reactor vessel internal stainless steel non-bolted components (e.g. plates, forgings and welds).
Aging Effects – A description of the applicable aging effects to be managed or the relevant physical conditions to be monitored for the identified scope of structures and components.	Cracking due to IASCC. Reduction of fracture toughness due to IE. Dimensional changes due to Swelling.	Cracking due to IASCC. Reduction of fracture toughness due to IE. Dimensional changes due to Swelling.

DRAFT INSPECTION PROGRAM FOR OCONEE REACTOR VESSEL INTERNALS JULY 19, 1999		
Program and Activity Attributes (from Section 4.2 of the Oconee Application)	**Draft Reactor Vessel Internals Baffle Bolt Inspection**	**Draft Inspection of Reactor Vessel Internals Non-Bolted Component Items**
Method – A description of the type of action or technique used to identify or manage the aging effects or relevant conditions (e.g., visual examination of the component).	Volumetric A bid specification is being prepared by the B&WOG, which will be used to further develop the method.	Visual The EPRI-MRP RV Internals ITG is developing visual inspection requirements for PWR internal materials subject to aging degradation.
Sample Size – For new programs or activities, a sample population can be identified from the total population of affected structures and components for inspection or monitoring. If a sample population is chosen for inspection or monitoring, a description of the sample population is provided.	The sample size will be a selected number of baffle bolts from one Oconee unit. The minimum sample size will be based on the final detail safety assessment being performed by the B&WOG.	A selected region of the item most limiting non-bolted component item (baffle plate) at one Oconee unit.
Industry Codes or Standards – A description of an industry code (e.g., ASME Section XI, IEEE) or an industry standard (e.g., ASTM or NRC-approved BWOG report) that guides or governs the program or activity. This attribute may not be applicable to some programs and activities.	No code or standard currently exists to guide or govern this inspection.	No code or standard currently exists to guide or govern this inspection.
Frequency – A description of the frequency of action that is established detection of aging effects or of relevant physical conditions.	The *Reactor Vessel Internals Baffle Bolt Inspection* is a one-time inspection.	The *Reactor Vessel Internals Non-Bolted Component Items Inspection* is a one-time inspection.

DRAFT INSPECTION PROGRAM FOR OCONEE REACTOR VESSEL INTERNALS JULY 19, 1999		
Program and Activity Attributes (from Section 4.2 of the Oconee Application)	**Draft Reactor Vessel Internals Baffle Bolt Inspection**	**Draft Inspection of Reactor Vessel Internals Non-Bolted Component Items**
Acceptance Criteria or Standard – Acceptance criteria or standards are described for the relevant conditions to be monitored or the chosen examination methods.	An acceptance criterion will be developed based on the number of required baffle bolts and a detectable indication.	An acceptance criterion will be developed based on ITG and B&WOG activities.
Corrective Action – A description of the action to be taken when the established acceptance criteria or standard is not met. Generally, the *Duke Quality Assurance Program* is applicable.	If the results of the inspection are not acceptable, then the result may be used as a baseline inspection for establishing a longer-term programmatic action to inspect and or replace baffle bolts in all three Oconee units.	To be determined.
Timing of New Program or Activity – For any new programs or activities, an identification of the specific timing for the new program or activity.	Following issuance of renewed operating licenses for Oconee Nuclear Station, this inspection will be performed during the 4th 10-year ISI Interval of the Oconee unit being inspected.	Following issuance of renewed operating licenses for Oconee Nuclear Station, this inspection will be performed during the 4th 10-year ISI Interval of the Oconee unit being inspected.
Administrative Controls – An identification of the Oconee administrative structure under which the programs and activities are executed. The *Duke Quality Assurance Program* is applicable.	The *Reactor Vessel Internals Baffle Bolt Inspection* will be implemented by plant procedures in accordance with the *Duke Quality Assurance Program.*	The *Inspection of Reactor Vessel Internals Non-Bolted Component* will be implemented by plant procedures in accordance with the *Duke Quality Assurance Program.*
Regulatory Basis – An identification of any existing Oconee regulatory basis for these actions such as the Technical Specifications. This attribute may be not applicable to some programs and activities.	BAW 2248 Safety Evaluation Report, Renewal Action Item number 4. Oconee's SER open item 3.4.3.3-4.	BAW-2248, Open Items DSE 2& 5. Oconee's SER open item 3.4.3.3-3.

Comparison of
Renewal Applicant Action Items and Open Items of DSE to BAW-2248
To
Open Items on the Oconee License Renewal SER

BAW-2248 Draft Safety Evaluation Items	Oconee License Renewal SER Open Items	Proposed Response Provided	Awaiting BAW-2248 FSE
RENEWAL APPLICANT ACTION ITEMS			
1. Verify BAW-2248 bounds Oconee	-	✔	
2. Provide UFSAR Summary	3.0-1	✔	
3. Verify Intended Function of Thermal Shield	-	✔	
4. Provide Baffle Bolt Inspection Details	3.4.3.3-4	✔	
5. Provide plant specific evaluation of flaw growth TLAA	4.2.5.3-2	✔	
6. Provide plan to obtain data	4.2.5.3-1	✔	
BWOG OPEN ITEMS			
-	3.4.3.2-1 (CASS Pressurizer Spray Head)	✔	
1. Void Swelling	3.4.3.2-2		✔
2. Commit to Enhanced VT-1 on non-bolting components	3.4.3.3-3		✔
3. Commit to Enhanced VT-1 on RVI CASS components	3.4.3.3-5 (also applies to CASS Valve Bodies in the RCS)		✔
-	3.4.3.3-6 (CASS Vent Valves)	✔	✔
-	3.4.3.3-7 (CASS RCP Casings)	✔	
4. Fluence Estimation	See 3.4.3.3-5 above		✔
5. Neutron Embrittlement	See 3.4.3.3-3 above		✔

Distribution:
Hard copy
PUBLIC
Docket File
RLSB RF
N. Dudley, ACRS - T2E26
E. Hylton

E-mail:
R. Zimmerman
W. Kane
D. Matthews
S. Newberry
C. Grimes
C. Carpenter
B. Zalcman
J. Strosnider
R. Wessman
E. Imbro
W. Bateman
J. Calvo
H. Brammer
T. Hiltz
G. Holahan
T. Collins
C. Gratton
B. Boger
R. Correia
R. Latta
J. Moore
J. Rutberg
R. Weisman
M. Mayfield
S. Bahadur
N. Chokshi
A. Murphy
D. Martin
W. McDowell
S. Droggitis
RLSB Staff
