May 25, 2005

10 CFR 54

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 1, 2, AND 3 - LICENSE RENEWAL APPLICATION (LRA) – RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION CONCERNING REACTOR PRESSURE VESSEL (RPV) INTERNALS (TAC NOS. MC1704, MC1705, AND MC1706)

By letter dated December 31, 2003, TVA submitted, for NRC review, an application pursuant to 10 CFR 54, to renew the operating licenses for the Browns Ferry Nuclear Plant, Units 1, 2, and 3. As part of its review of TVA's LRA, the NRC staff, through an informal request on March 29, 2005, requested additional information concerning RPV internals.

The enclosure to this letter contains the specific NRC requests for additional information and the corresponding TVA responses.

If you have any questions regarding this information, please contact Ken Brune, Browns Ferry License Renewal Project Manager, at (423) 751-8421.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 25th day of May, 2005.

Sincerely,

Original signed by:

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosure:
cc: See page 3

Enclosure
cc (Enclosure):

State Health Officer
Alabama Department of Public Health
RSA Tower - Administration
Suite 1552
P.O. Box 303017
Montgomery, Alabama 36130-3017

Chairman
Limestone County Commission
310 West Washington Street
Athens, Alabama 35611

(Via NRC Electronic Distribution)
Enclosure
cc (Enclosure):

U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-8931

Mr. Stephen J. Cahill, Branch Chief
U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-8931

NRC Senior Resident Inspector
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, Alabama 35611-6970

NRC Unit 1 Restart Senior Resident Inspector
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, Alabama 35611-6970

cc: continued page 4

cc: (Enclosure)
Margaret Chernoff, Project Manager
U.S. Nuclear Regulatory Commission
(MS 08G9)
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

Eva A. Brown, Project Manager
U.S. Nuclear Regulatory Commission
(MS 08G9)
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

Yoira K. Diaz-Sanabria, Project Manager
U.S. Nuclear Regulatory Commission
(MS 011F1)
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

Ramachandran Subbaratnam, Project Manager
U.S. Nuclear Regulatory Commission
(MS 011F1)
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

JEM:TLE:BAB
Enclosure
cc (Enclosure):

A. S. Bhatnagar, LP 6-C
K. A. Brune, LP 4F-C
J. C. Fornicola, LP 6A-C
R. G. Jones, NAB 1A-BFN
K. L. Krueger, POB 2C-BFN
R. F. Marks, Jr., PAB 1A-BFN
F. C. Mashburn, BR 4X-C
N. M. Moon, LP 6A-C
J. R. Rupert, NAB 1F-BFN
K. W. Singer, LP 6A-C
M. D. Skaggs, PAB 1E-BFN
E. J. Vigluicci, ET 11A-K
NSRB Support, LP 5M-C
EDMS, WT CA-K

s://Licensing/Lic/BFN LR The RPV Internals Pressure Vessel.doc

ENCLOSURE

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3
LICENSE RENEWAL APPLICATION (LRA)

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI)
CONCERNING REACTOR PRESSURE VESSEL (RPV) INTERNALS

(SEE ATTACHED)

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3
LICENSE RENEWAL APPLICATION (LRA)

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) CONCERNING REACTOR PRESSURE VESSEL (RPV) INTERNALS

By letter dated December 31, 2003, TVA submitted, for NRC review, an application pursuant to 10 CFR 54, to renew the operating licenses for the Browns Ferry Nuclear Plant, Units 1, 2, and 3. As part of its review of TVA's LRA, the NRC staff, through an informal request on March 29, 2005, requested additional information concerning RPV internals. This enclosure contains the specific NRC requests for additional information and the corresponding TVA responses.

NRC Question (1)

TLAA-4.2.6 – RPV Circumferential Weld Examination Relief:

The UFSAR Supplement summary description adequately describes the TLAA for BFN, Units 2 and 3, the staff concludes that the UFSAR supplement summary description for the TLAA on RV circumferential weld examination relief for BFN, Units 2 and 3 is acceptable. However, the UFSAR supplement summary description should include RV circumferential weld examination relief for the BFN Unit 1.

TVA Response to NRC Question (1)

The reactor vessel circumferential weld examination analysis is addressed for Unit 1 as shown in the revised version of UFSAR Supplement A.3.1.6 contained in Attachment 1.

NRC Question (2)

TLAA-4.2.7 – RPV Axial Weld Failure Probability:

The UFSAR supplement should include the analysis on the RV axial weld failure probability for the BFN, Unit 1 for the period of extended operation.

TVA Response to NRC Question (2)

The reactor vessel axial weld conditional failure probabilities analysis is addressed for BFN Unit 1 as shown in the revised version of UFSAR Supplement A.3.1.7 contained in Attachment 1.

NRC Question (3)

TLAA–4.7.6 – IASCC of the RPV Internals:

The applicant in its response to RAI-B.2.1.12(A), indicated that LRA section 4.7.6 considered fluence at the top guide as a TLAA. BFN manages this TLAA with the Chemistry Control Program and the BWR Vessel Internals Program. The BWR Vessel Internals Program implements the requirements of NRC accepted BWRVIP-26. NRC letter to Carl Terry, BWRVIP Chairman, dated June 10, 2003 states the following: "The staff believes that a comprehensive evaluation of the impact of IASCC and multiple failures of the top guide beams is necessary, and that an inspection program for top guide beams for all BWRs should be developed by the BWRVIP to ensure that all BWRs can meet the requirements of 10 CFR Part 54 throughout the period of extended operation." The applicant made a commitment to work as part of the BWRVIP to resolve these issues generically. When resolved, the applicant will follow the BWRVIP recommendations resulting from that resolution. Prior to the period of extended operation, the applicant will develop a site-specific inspection program, if these issues are not generically resolved. The staff determines that the applicant must submit for NRC review and approval, a site-specific AMP that addresses the potential multiple failures of the top guide grid beams.

TVA Response to NRC Question (3)

The austenitic stainless steel top guide is considered susceptible to irradiation assisted stress corrosion cracking (IASCC) in the BWR environment as the neutron fluence is greater than 5×10^{20} n/cm^2 (E > 1 MeV), the industry accepted threshold for IASCC susceptibility. Therefore, the aging effect associated with IASCC, crack initiation and growth, will require aging management.

The location on the top guide that will see this high fluence is the grid beam, with the highest fluence being near the core centerline. This is location 1, as identified in BWRVIP-26, Table 3-2, "Matrix of Inspection Options." In accordance with BWRVIP-26, Table 3-2, location 1 does not require inspections. BFN agrees to perform inspections of the guide beams similar to the inspections of the Control Rod Drive (CRD) guide tube. The inspection of the CRD guide tube is performed in accordance with BWRVIP-47, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines. The examination extent and frequency is a 10% sample of the total population within 12 years, one-half (5%) to be completed within six years. The sample will emphasize the components exposed to the highest fluence. The method of examination will be EVT-1. The proposed aging management inspection is similar to the program accepted by the NRC staff in NUREG-1769 Safety Evaluation Report Related to the License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3, in paragraph 4.5.2.

To implement this change, LRA Appendix B.2.1.12, Boiling Water Reactor Vessel Internals Program, has been revised as follows:

Revised “Program Description” to include paragraph (c):

(c) Inspection will be performed of the top guide beams with examination extent and frequency similar to the CRD guide tube (BWRVIP-47) examination requirements. This program enhancement will be implemented prior to the end of the initial operating license term. However, TVA reserves the right to modify the above agreed upon inspection program should BWRVIP-26 be revised in the future. Any changes, however, will not be final until a Safety Evaluation is issued by the NRC.

Revised “NUREG-1801 Consistency” to read

> The BWR Vessel Internals Program is an existing program that, with enhancements, will be consistent with or, in relationship to the top guide, exceed NUREG-1801 XI.M9 evaluation elements.

Revised “Enhancements” to include a second enhancement:

> In addition to the requirements of the BWRVIP guidelines, BFN agrees to perform inspections of the top guide beams similar to the inspections of the Control Rod Drive (CRD) guide tube prior to the end of the initial operating license term. The inspection of the CRD guide tube is performed in accordance with BWRVIP-47, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines. The examination extent and frequency is a 10% sample of the total population within 12 years, one-half (5%) to be completed within six years. The sample will emphasize the components exposed to the highest fluence. The method of examination will be EVT-1.

NRC Question (4a)

TLAA-4.7.8 - IASCC of the RPV Internals:

The staff however, requires that the applicant revise the UFSAR supplement to indicate that the inspection guidelines of the BWRVIP-25 "Boiling Water Reactor Core Plate Inspection and Flaw Evaluation Guidelines," will be implemented to effectively manage the aging effect on core plate.

TVA Response to NRC Question (4a)

The revised version of LRA Sections 4.7.6 and A.3.5.5 are contained in Attachment 2. The question reference to 4.7.8 is incorrect; 4.7.8 is the Emergency Equipment Cooling Water Weld Flaw Evaluation.

NRC Question (4b)

AMR – Reactor Vessel Nozzles and Safe Ends:

The applicant also claimed that improvements in RCS water chemistry provide mitigative measures to preclude IGSCC in the dissimilar welds in nozzle-to-safe end, pipe-to-safe end and nozzle-to cap components. The staff accepts the proposed program for stainless steel safe ends because it conforms to the recommendations in the BWRVIP-75. However, if the safe ends contain nickel alloy weld metals that are susceptible to SCC, BWRVIP-75 would require more frequent examinations than that specified for BWRVIP-75 Category A welds. In order for the staff to determine whether the applicant has adequately implemented BWRVIP-75, the staff requests that the applicant identify : (1) the weld metal that was used for the butter, nozzle-to-safe end welds, pipe-to-safe end welds and nozzle-to cap welds ; (2) the grade of stainless steel that was used as a safe end; and (3) the examination requirements for butter, nozzle-to-safe end welds, pipe-to-safe end welds and nozzle-to cap welds that are more susceptible to SCC than the BWRVIP-75 Category A welds.

TVA Response to NRC Question (4b)

As shown in LRA Table 3.1.2.1, the only materials identified for reactor vessel recirculation nozzles and safe ends for BFN Units 1, 2, and 3 are carbon and low alloy steel and stainless steel. This includes the materials found in the nozzle to safe end welds. No nickel alloy weld materials were used in the nozzle to safe end welds identified. The following tables identify the metals requested.

Reactor Vessel Penetration	Nozzle Butter Weld Material	Nozzle to Safe End Weld Material	Safe End Material	Safe End to Pipe Weld Material	BWRVIP-75 Examination Category
Unit 1 Reactor Vessel Recirc Inlet	308/309 with 308L corrosion resistant cladding	308/308L (dual certified)	ASME SA 182 GRADE F 316NG	308L	BWRVIP-75 will be implemented prior to Unit 1 Restart
Unit 1 Reactor Vessel Recirc Outlet	308/309 with 308L corrosion resistant cladding	316/316L (dual certified)	ASME SA 182 GRADE F 316NG	308L	BWRVIP-75 will be implemented prior to Unit 1 Restart
Unit 1 Core Spray	308/309 with 308L corrosion resistant cladding	308/308L (dual certified)	ASME SA 182 GRADE F 316NG	309L	BWRVIP-75 will be implemented prior to Unit 1 Restart
Units 2 and 3 Reactor Vessel Recirc Inlet	308/309 with 308L corrosion resistant cladding	308L/309L	ASME SA 182 GRADE F 316NG	308L/309L	Category A Included in the Risk Informed ISI Program
Units 2 and 3 Reactor Vessel Recirc Outlet	308/309 with 308L corrosion resistant cladding	308L/309L	ASME SA 376 GRADE 316	308L/309L	Category C
Units 2 and 3 Core Spray	308/309 with 308L corrosion resistant cladding	308L/309L	ASME SA 182 GRADE F 316NG	308L/309L	Category A Included in the Risk Informed ISI Program

Reactor Vessel Penetration	Nozzle Butter Weld Material	Nozzle to Nozzle Cap Weld Material	Nozzle Cap Material	BWRVIP-75 Examination Category
Unit 1 CRD Return Line Cap	308/309 with 308L corrosion resistant cladding	308L/309L	ASME SA 182 GRADE F 316L	BWRVIP-75 will be implemented prior to Unit 1 Restart
Units 2 and 3 CRD Return Line Nozzle Cap	308/309 with 308L corrosion resistant cladding	308L/309L	ASME SA 182 GRADE F 316L	Category D - Unit 2 Category C - Unit 3

NRC Question (5)

AMR — Feedwater Nozzles:

To RLEP for Resolution: The applicant shall revise A.1.8 of the UFSAR supplement to indicate GE-NE-523-A71-0594-A, Revision 1 which is the applicable document for feedwater nozzle analysis and inspection.

TVA Response to NRC Question (5)

This question was previously addressed in the response to NRC RAI B.2.1.8-1 (page E-61 of reference 1). The response to NRC RAI B.2.1.8-1 is repeated below:

> "The BWR Feedwater Nozzle program will implement feedwater nozzle inspection recommendations based on GE Report GE-NE-523-A71-0594-A Revision 1. Replace references to GE-NE-523-A71-0594 in LRA Appendix A.1.8 and Appendix B.2.1.8 with GE-NE-523-A71-0594-A Revision 1."

NRC Question (6)

AMR - CRD Return Line Welds:

In order for the staff to determine whether the applicant has adequately implemented BWRVIP-75, for the Category A CRD return line nozzle welds in all BFN units, the staff requests that the applicant identify: (1) the delta ferrite in the weld metal, (2) the grade of stainless steel that was used for the CRD return line cap, (3) the examination requirements for CRD return line welds that meet BWRVIP-75, and (4) plans to implement MSIP in BFN, Units 1 and 2.

TVA Response to NRC Question (6)

The following table provides the specific information requested for the CRD return line caps and welds:

Reactor Vessel Penetration	Nozzle to Cap Weld Material	Cap Material	BWRVIP-75 Examination Category	Mechanical Stress Improvement
Unit 1 Control Rod Drive Return Line Cap	308L/309L with a minimum FN 8.0 delta ferrite content	ASME SA 182 GRADE F 316L	BWRVIP-75 will be implemented prior to Unit 1 Restart	MSIP will be performed.
Unit 2 Control Rod Drive Return Line Cap	308L/309L with a minimum FN 8.0 delta ferrite content	ASME SA 182 GRADE F 316L	Category D	No MSIP performed or planned.
Unit 3 Control Rod Drive Return Line Cap	308L/309L with a minimum FN 8.0 delta ferrite content	ASME SA 182 GRADE F 316L	Category C	MSIP has been performed.

NRC Question (7)

AMR - Access Hole Covers:

RLEP/EMCB For Resolution: Since Section IV-B1.1.4 of NUREG-1801, requires UT of AHC welds, the staff requests that the applicant revise AMP B.2.1.12 and UFSAR supplement A.1.12 to include UT for the BFN, Units 2 and 3 AHC welds to the maximum extent possible. The staff requests that the applicant identify its previous experience on the extent to which UT was performed on the AHC welds at the BFN units.

TVA Response to NRC Question (7)

The Unit 1 AHCs will be changed to bolted design prior to Unit 1 restart and do not require enhanced inspections.

The welded Access Hole Covers (AHCs) inspection requirements for Units 2 and 3 were previously provided to the NRC in the second paragraph of our response to part (C) of NRC RAI B.2.1.12-1 (page E-70 of reference 1). The Units 2 and 3 welded AHC inspection requirements are being changed as shown below:

> "Prior to the period of extended operation, BFN will enhance the BWR Vessel Internals Program to require

visual inspection of the AHCs and inspection of the AHCs welds by UT unless tooling constraints prohibit performance of a UT. In the event tooling constraints prohibit inspection by UT during the period of extended operation, NRC approval will be requested to substitute an EVT-1 inspection. BFN inspects the AHCs utilizing the BWR Vessel Internals Program rather than the ASME Section XI Inservice Inspection Program currently specified in NUREG-1801."

TVA has determined that revision of LRA Sections B.2.1.12 and A.1.12 is not required; however, TVA does commit to include the above examination requirements into the Reactor Pressure Vessel Internals Inspection (RPVII) Units 1, 2, And 3 Procedure.

The operating experience for the access hole covers is described below:

A Unit 1 access hole cover currently has indications of cracking and both Unit 1 access hole covers will be replaced prior to Unit 1 restart.

Unit 2 access hole cover inspections results are as follows:

- 1996 - UT examination with no reportable indications.
- 1999 - UT examination with no reportable indications.
- 2005 - EVT-1 examination with no reportable indications.

Unit 3 access hole cover inspections results are as follows:

- 1994 - UT examination with no reportable indications.
- 1998 - UT examination with no reportable indications.
- 2004 - EVT-1 examination with no reportable indications.

Based on the plant operating experience, the Unit 1 Access Hole Covers will be replaced prior to Unit 1 restart. No repairs have been performed on the Units 2 and 3 Access Hole Covers.

NRC Question (8)

AMR – In-core housing Guide Tube/Dry Tube:

To RLEP For Resolution: The applicant stated that it will replace all BFN, Unit 1 dry tubes prior to restart. The applicant must commit to replace all BFN, Unit 1 dry tubes prior to restart. This is a commitment which would be either contained in a tracking process for either BFN, Unit 1 restart or license renewal.

TVA Response to NRC Question (8)

TVA commits to replace all BFN Unit 1 dry tubes prior to restart.

NRC Question (9)

AMP-B.2.1.28 – Reactor Vessel Surveillance Program:

Any changes to the BWRVIP ISP capsule withdrawal schedule must be submitted for NRC staff review and approval. Any changes to the BWRVIP ISP capsule withdrawal schedule which affects the time of withdrawal of any surveillance capsules must be incorporated into the licensing basis. If any surveillance capsules are removed without the intent to test them, these capsules must be stored in manner which maintains them in a condition which would support re-insertion into the RV, if necessary.

TVA Response to NRC Question (9)

The revised version of LRA Section A.1.25 is contained in Attachment 3.

NRC Question (10)

AMP-B.2.1.28 – Reactor Vessel Surveillance Program:

The staff requests that the applicant should include the following statement in the UFSAR, A.1.25, addressing the issue related to the standby capsules.

> The BFN, Units 1 and 3 surveillance capsules (standby capsules) will remain in place and will continue to be irradiated during plant operation, including the period of extended operation. Therefore, the BFN, Units 1 and 3 irradiated material samples continue to remain available to the ISP, if needed.

TVA Response to NRC Question (10)

The revised version of LRA Section A.1.25 is contained in Attachment 3.

NRC Question (11)

AMR – BWRVIP Program Commitment: (This AMP is reviewed by the audit team)

The staff reviewed the response, and determines that if the applicant takes exceptions to the implementation of BWRVIP inspection guidelines as a part of AMP for the RVIs, it must submit the exceptions (including the exceptions that were taken on BWRVIP-18 and BWRVIP-41 as stated in the applicant's response dated January 31, 2005), to the staff for review and approval either now or two years prior to the commencement of the extended period of operation. If the applicant does not submit the exception now, it must commit to submit the exception for staff's review and approval two years prior to the commencement of the extended period of operation.

TVA Response to NRC Question (11)

Currently, no exception to the BWRVIP guidelines has been identified. The response to NRC RAI-B.2.1.12-1(B) (page E-69 of reference 1) should have indicated that three issues were currently being investigated to determine actions required to meet the BWRVIPs. The second un-bulleted paragraph of the response to NRC RAI-B.2.1.12-1(B) should read:

> "No augmented inspection program has been implemented for any of the components listed under RAI-B.2.1.12 (B) as part of this AMP. The CRD Nozzle Welds, ICM Nozzle Welds, and SLC Nozzle Welds are inspected in accordance with ASME Code Section XI as specified in BWRVIP-27 (SLC) and BWRVIP-47 (CRD, ICM). The Core Shroud, Top Guide, Core Spray Piping and Spargers, and Jet Pump Assembly are inspected in accordance with their respective BWRVIP Inspection and Evaluation Guideline as part of BFN's BWR Vessel Internals Program. Three potential concerns with BWRVIP implementation have been identified and an investigation into how compliance with the BWRVIP guidance will be achieved is currently in progress. These concerns are:

- In accordance with BWRVIP-27, an enhanced leakage inspection of the Standby Liquid Control (SLC) safe-end-to-nozzle weld during the ASME Section XI, IWB-2500, Code Category B-P system leakage test will be performed.

- The effect that the Unit 3 Jet Pump #5 repair design has on the implementation of the weld inspection requirements specified in BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," is currently being evaluated in the BFN Corrective Action Program.

- The effect that the Unit 3 Core Spray repair designs have on the implementation of the inspection requirements specified in BWRVIP-18, "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines" is currently being evaluated in the BFN Corrective Action Program.

NRC Question (12)

AMR - Jet Pump Nozzle Thermal Sleeve Welds: (This AMP is reviewed by the audit team)

The BWRVIP-41 report was approved by the staff. According to paragraph A.2 in Appendix A of the BWRVIP-41 report, jet pump thermal sleeve welds require a plant-specific AMR, and the applicant should submit this AMR to NRC for review and approval.

TVA Response to NRC Question (12)

The BFN jet pump thermal sleeve type is Type A configuration as shown in Figure 2.3.3-1 of BWRVIP-41, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines. The Type A thermal sleeve has only one weld (TS-2). Table 3.3-1, Matrix of Inspection Options, specifies the inspection recommendations for the TS-2 weld. BWRVIP-41, Table 3.3-1 notes that the TS-2 weld is currently inaccessible and that the BWRVIP Inspection Committee is addressing the need to develop an inspection technique for this weld. The BWRVIP notes that analysis may alleviate this inspection requirements.

The BFN jet pump nozzle and forged integral portion of the thermal sleeve are type 316NG stainless steel. This portion of the thermal sleeve is subject to fatigue; however, this material is not subject to IGSCC according to NUREG-0313.

The remaining portion of the thermal sleeve is type 304 stainless steel that is subject to both fatigue and IGSCC aging mechanisms. The TS-2 weld is a stainless steel (316NG) to stainless steel (304) weld. The aging effects evaluation for the thermal sleeves is documented in lines 28 and 29 of LRA Table 3.1.2.2. As identified in line 28, the Chemistry Control Program and the BWR Reactor Vessel Internals Program manages SCC(IGSCC) for the thermal sleeves. The BWR Reactor Vessel Internals Program implements the inspection requirements for TS-2. As noted in Table 3.3-1 of BWRVIP-41, the TS-2 weld is currently inaccessible and is not inspected. BFN will implement the inspection technique for this weld that the BWRVIP Inspection Committee is developing when available.

REFERENCES

1. T. E. Abney (TVA) to Document Control Desk (NRC) letter dated January 31, 2005: BROWNS FERRY NUCLEAR PLANT (BFN) – UNITS 1, 2, AND 3 - LICENSE RENEWAL APPLICATION – REACTOR VESSEL AND INTERNALS MECHANICAL SYSTEMS SECTIONS 3.1, 4.2, AND B.2.1 – RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) (TAC NOS. MC1704, MC1705, AND MC1706)

ATTACHMENT 1

Revision to LRA Sections A.3.1.6 and A 3.1.7

A.3.1.6 Reactor Vessel Circumferential Weld Examination Relief

Units 2 and 3 have received relief[1] from reactor vessel circumferential weld examination requirements under Generic Letter 98-05 for the remainder of the 40 year licensed operating period. In addition, Unit 1 has applied for relief[2, 3, 4, 5] from reactor vessel circumferential weld examination requirements under Generic Letter 98-05 for the remainder of the 40 year licensed operating period. The circumferential weld examination relief analyses are based on probabilistic assessments that predict an acceptable probability of failure per reactor operating year. The analysis is based on reactor vessel metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period.

Although a conditional failure probability has not been recalculated, an analysis that concluded values at the end of a 60 year life are less than the 64 EFPY value provided by the NRC leads to the conclusion that the BFN reactor vessel conditional failure probability is bounded by the NRC analysis in its safety evaluation report (SER) for BWRVIP-05.[6] The procedures and training used to limit cold over-pressure events will be the same as that approved by the NRC when BFN requested the BWRVIP-05 technical alternative be used for the current term. An extension of this relief request will be submitted to the NRC for approval prior to entering the period of extended operation.

A.3.1.7 Reactor Vessel Axial Weld Failure Probability

The BWRVIP-05 recommendations for inspection of reactor vessel shell welds contain generic analyses supporting an NRC SER[7] conclusion that the generic plant axial weld failure rate is no more than 5×10^{-6} per reactor year. BWRVIP-05 showed that this axial weld failure rate of 5×10^{-6} per reactor year is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds as described in A.3.1.6. The BFN limiting weld chemistry, chemistry factor and 60 year life mean RT_{NDT} values for Units 1, 2 and 3 are within the limits of the values assumed in the analysis performed by the NRC staff in its BWRVIP-05 SER supplement[8]. Therefore, the probability of failure for the axial welds is bounded by the NRC evaluation.

---

1. NRC letter to TVA, "Browns Ferry Nuclear Plant Unit 2, Relief Request 2-ISI-9, Alternatives for Examination of Reactor Pressure Vessel Shell Welds (TAC No. MA8424)," 11/18/1999, and NRC letter to TVA, "Browns Ferry Nuclear Plant Unit 3, Relief Request 3-ISI-1, Revision 1, Alternatives for Examination of Reactor Pressure Vessel Shell Welds (TAC No. MA5953)," 11/18/1999

2. TVA letter to U.S. Nuclear Regulatory Commission, Document Control Desk, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - American Society of Mechanical Engineers (ASME) Section XI and Augmented Inspections - Request for Relief, 1-ISI-19, Regarding Reactor Pressure Vessel (RPV) Circumferential Shell Welds," 5/12/2004

3. TVA letter to U.S. Nuclear Regulatory Commission, Document Control Desk, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - American Society of Mechanical Engineers (ASME) Section XI and Augmented Inspections – Response to Request For Additional Information - Request for Relief, 1-ISI-19, Regarding Reactor Pressure Vessel (RPV) Circumferential Shell Welds," 8/13/2004

4. TVA letter to U.S. Nuclear Regulatory Commission, Document Control Desk, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - American Society of Mechanical Engineers (ASME) Section XI and Augmented Inspections – Revision to Request for Relief, 1-ISI-19, Regarding Reactor Pressure Vessel (RPV) Circumferential Shell Weld Examinations," 11/8/2004

5. TVA letter to U.S. Nuclear Regulatory Commission, Document Control Desk, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - American Society of Mechanical Engineers (ASME) Section XI and Augmented Inspections – Request for Relief 1-ISI-19, Regarding Reactor Pressure Vessel (RPV) Circumferential Shell Weld Examinations – Supplemental Information," 3/4/2005

6. NRC letter from Gus C. Lainas to Carl Terry, Niagara Mohawk Power Company, BWRVIP Chairman, "Final Safety

Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report," (TAC No. M93925), July 28, 1998

7. NRC letter from Gus C. Lainas to Carl Terry, Niagara Mohawk Power Company, BWRVIP Chairman, "Final Safety Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report," (TAC No. M93925), July 28, 1998

8. NRC letter from Jack R. Strosnider, to Carl Terry, BWRVIP Chairman, "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report," (TAC No. MA3395), March 7, 2000

ATTACHMENT 2

Revision to LRA Sections 4.7.6 and A.3.3.5

4.7.6 Irradiation Assisted Stress Corrosion Cracking (IASCC) of Reactor Vessel Internals

Summary Description

Austenitic stainless steel reactor internal components exposed to neutron fluence greater than 5 x 10^{20} n/cm^2 (E > 1 MeV) are considered susceptible to Irradiation Assisted Stress Corrosion Cracking (IASCC) in the BWR environment. As described in the SER (ML003776810, 12/07/2000) to BWRVIP-26, "BWR Top Guide Inspection and Flaw Evaluation Guidelines," IASCC of reactor internals is considered a TLAA.

Analysis

Fluence calculations have been performed for the reactor vessel and internals. Four components have been identified as being susceptible to IASCC for the period of extended operation: (1) Top Guide; (2) Shroud; (3) Core Plate and (4) In-core Instrumentation Dry Tubes and Guide Tubes.

Disposition: 10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The top guide, shroud, core plate and in-core instrumentation dry tubes and guide tubes are considered susceptible to IASCC. The aging effect associated with IASCC, crack initiation, and growth will require aging management. IASCC of three components, top guide, shroud and in-core instrumentation dry tubes and guide tubes, have been evaluated by the BWRVIP, as described in the Inspection and Evaluation Guidelines for each component: BWRVIP-26 (Top Guide), BWRVIP-76 (Shroud), and BWRVIP-47 (in-core instrumentation dry tubes and guide tubes). In addition, to the BWRVIP inspections for BWRVIP-26, -47, and -76, the Chemistry Program and the BWR Vessel Internals Program implement the recommendations of BWRVIP-25 for the Core Plate.

A.3.5.5 Irradiation Assisted Stress Corrosion Cracking of Reactor Vessel Internals

Austenitic stainless steel reactor vessel internal components exposed to a neutron fluence greater than 5×10^{20} n/cm^2 (E > 1 MeV) are considered susceptible to irradiation assisted stress corrosion cracking (IASCC) in the BWR environment. Fluence calculations have been performed for the reactor vessel and internals. Four components have been identified as being susceptible to IASCC for the period of extended operation: (1) Top Guide; (2) Shroud; (3) Core Plate and (4) In-core Instrumentation Dry Tubes and Guide Tubes. IASCC of three components (top guide, shroud and in-core instrumentation dry tubes and guide tubes) have been evaluated by the BWRVIP, as described in the Inspection and Evaluation Guidelines for each component: BWRVIP-26 (Top Guide), BWRVIP-76 (Shroud), and BWRVIP-47 (in-core instrumentation dry tubes and guide tubes). In addition, to the BWRVIP inspections for BWRVIP-26, -47, and -76, the Chemistry Program and the BWR Vessel Internals Program implement the recommendations of BWRVIP-25 for the Core Plate.

ATTACHMENT 3

Revision to LRA Section A.1.25

A.1.25 Reactor Vessel Surveillance Program

The BFN Reactor Vessel Surveillance Program is mandated by 10 CFR 50 Appendix H. The BFN Reactor Vessel Surveillance Program is an integrated surveillance program (ISP) in accordance with 10 CFR Part 50 Appendix H Paragraph III.C that is based on requirements established by the BWR Vessel and Internals Project reports.

This program will be enhanced to implement either BWRVIP-116, as approved by the staff, or, if the ISP is not approved two years prior to the commencement of the license renewal period, a plant-specific surveillance program for each BFN unit will be submitted that ensures the BFN Unit 1, Unit 2, and Unit 3 reactor vessels meet the requirements of 10 CFR 50 Appendix H.

The BFN, Units 1 and 3 surveillance capsules (standby capsules) will remain in place and will continue to be irradiated during plant operation, including the period of extended operation. Therefore, the BFN, Units 1 and 3 irradiated material samples continue to remain available to the ISP, if needed. If any surveillance capsules are removed without the intent to test them, these capsules will be stored in manner which maintains them in a condition which would support re-insertion into the RV, if necessary.