Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

December 13, 2000

10 CFR 50.55a(a)(3)(i)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of)
Tennessee Valley Authority)

Docket No. 50-260

BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 2 - PROPOSED RISK-INFORMED INSERVICE INSPECTION PROGRAM - SUPPLEMENTAL RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (TAC NO. MA8873)

This letter provides a supplemental response to the NRC September 29, 2000, request for additional information (RAI) regarding the proposed risk-informed inservice inspection (RI-ISI) program for BFN Unit 2. TVA is providing additional information to its responses submitted by letter dated October 16, 2000, for NRC questions six and seven. This response addresses items identified by the staff in an electronic mail from the NRC Project Manager dated November 16, 2000, and discussed in a teleconference with the staff and TVA personnel on December 4, 2000.

The BFN Unit 2 RI-ISI program was submitted to NRC by TVA letter dated June 1, 2000. The BFN Unit 2 RI-ISI program is an alternative to current ASME Section XI inservice inspection requirements for Code Class 1, 2, and 3 piping. The BFN Unit 2 RI-ISI program also provides an alternative to the inspection requirements of Intergranular Stress Corrosion Cracking (IGSCC) Category "A" welds.

The enclosure to this letter presents the specific NRC questions, and the corresponding TVA responses.

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There are no new commitments contained in this letter. If you have any questions, please contact me at (256) 729-2636.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosure
cc (Enclosure):

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ENCLOSURE

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 2

PROPOSED RISK INFORMED INSERVICE INSPECTION PROGRAM, NRC REQUEST FOR ADDITIONAL INFORMATION SUPPLEMENTAL RESPONSE

Background

The proposed BFN Unit 2 RI-ISI program was submitted to NRC by letter dated June 1, 2000. The BFN Unit 2 RI-ISI program is an alternative to current ASME Section XI inservice inspection requirements for Code Class 1, 2, and 3 piping. The program also provides an alternative to the inspection requirements for Intergranular Stress Corrosion Cracking (IGSCC) Category "A" welds. TVA's program was developed in accordance with the Westinghouse Owners Group Topical Report WCAP-14572, Revision 1, as modified by the September 30, 1998, letter to the NRC from the Westinghouse Owners Group (with some deviations) and the guidance contained in Generic Letter 88-01.

As a result of its review of the BFN Unit 2 RI-ISI program, the NRC staff requested additional information (RAI) by letter dated September 29, 2000. TVA provided its response to the staff's request by letter dated October 16, 2000. This response addresses items identified by the staff in electronic mail from the NRC Project Manager for BFN dated November 16, 2000, and discussed in a teleconference between the staff and TVA personnel on December 4, 2000.

Listed below are the staff's supplemental questions and the corresponding TVA responses.

NRC Supplemental Request For Question 6

Table 3.5-1 (E1-page 15 of 64 of the BFN Unit 2 RI-ISI submittal) indicates that the selection of "with Operator action" as the applicable case can reduce the system level risk estimates by two orders of magnitude or more. Describe the Operators' actions selected as the "applicable case" when the change in risk calculations were performed. Previous information submitted only addressed the seven segments that were reduced from HSS to LSS. Describe the actions used to reduce risk for all other "with Operator

action" cases or provide quantitative evaluations showing that the results are insensitive to these assumptions. The descriptions should include sufficient information to characterize the events and the circumstances the Operators are reacting to, and provide confidence that the assumption that the operator has an essentially 0.0 probability of error is a reasonable assumption.

TVA Supplemental Response

TVA performed quantitative analyses and determined that the results of the risk-informed analysis are insensitive to the assumed Operator actions with the exception of the seven system segments listed in NRC question seven. The first analysis was done by calculating the risk reduction worth (RRW) for each segment without taking credit for Operator action. For both core damage frequency (CDF) and large early release frequency (LERF), none of the segments where Operator action had been determined to be the applicable case had risk reduction worth (RRW)> 1.000 with the exception of the segments addressed in question seven. Additional "change in risk" calculations were also performed. The base case without ISI was calculated with no credit for Operator action except for the seven segments addressed in question seven. The risk under the current Section XI ISI and augmented programs as well as the proposed risk-informed program were calculated. The calculations are summarized in Table 3.10-2 (Enclosure 1, page 29 of 64) of TVA's June 1, 2000 submittal.

The new analyses, where no credit is taken for Operator action for each of the programs (no ISI, current section XI ISI, current augmented, current Section XI ISI plus augmented, and Risk-Informed), are presented in new Table 3.10-2a which is provided in Attachment A of this enclosure. The analyses in Table 3.10-2a (no credit for Operator action) provides essentially the same change in risk for CDF and LERF as that shown in Table 3.10-2.

Therefore, the TVA evaluation shows that the risk-informed analyses are insensitive to the assumed Operator actions with the exception of the seven system segments listed in NRC question seven.

NRC Supplemental Request For Question 7

The generic discussion supplied by BFN is not sufficient to support a finding that the Operators' responses to the actions, and the time available to respond, justify placing

these segments in Low Safety Significance (LSS) even though the methodology indicates they should be placed in High Safety Significance (HSS). Describe each individual action including the specific location of the rupture, the plant event associated with the rupture that the Operators are responding to, the time to detection (based on the rupture flow rates and the relative location of the detection equipment), the time the Operators have to respond after detection (based on the rupture flow rates and the relative location of the susceptible equipment), the actions the Operators need to take, and the locations that these actions are taken in.

TVA Supplemental Response

In the December 4, 2000, teleconference referenced above, TVA proposed to provide a quantitative analysis of the worst-case pipe segment leakage scenario for the seven segments in question. This worst-case scenario effectively bounds the other six segments. The staff agreed that TVA's approach was acceptable.

The largest disabling leak for the seven line segments placed in LSS rather that HSS, based on Operator actions, is 80 gallons per minute (gpm). However, for these seven line segments, TVA performed a quantitative analysis using the maximum system disabling leakage rate of 300 gpm. TVA determined that it would take at least 20 hours of leakage at 300 gpm (approximately 375,000 gallons) to cause failure of Emergency Core Cooling System pumps. This analysis assumes no Operator intervention (which would normally be initiated within 10 minutes based upon a reactor building floor drain sump high level alarm), the absence of various level/alarm switches, and the absence of two auto starting reactor building floor drain sump pumps with a combined pumping capacity in excess of 300 gpm.

However, the various level/alarm switches and Reactor Building Floor drain sump pumps are in place and operable. Also, Operators make their rounds at least once every shift (12 hours) and would observe any leakage (provided they had not already been alerted by a high level Reactor Building sump alarm) and make the necessary pipe segment isolations. The likelihood that a leak in one of the seven pipe segments in question would go undetected and unresolved for 20 hours is not a creditable scenario.

The above analysis confirms that the seven piping segments are appropriately classified as LSS rather than HSS. TVA's

analysis further demonstrates that there are no creditable leak/crack scenarios that would result in a loss of all ECCS Pumps.

Attachment B of this enclosure lists the seven line segments and provides a description of the analysis summarized above.

Attachment A

Table 3.10-2a

COMPARISON OF CDF/LERF FOR CURRENT PROGRAMS AND FOR RISK-INFORMED PROGRAM

With no credit for Operator Action in LSS segments

Program	Piping CDF	Piping LERF
Without ISI	2.166E-05	7.368E-06
Current Section XI	2.166E-05 (99.99%)	7.368E-06 (99.99%)
Current Augmented	1.337E-05 (61.7%)	4.548E-06 (61.7%)
Current Section XI +Augmented	1.337E-05 (61.7%)	4.547E-06 (61.7%)
Risk-Informed	1.337E-05 (61.7%)	4.548E-06 (61.7%)

Attachment B

Pipe Segments Reduced From HSS To LSS Based On Operator Action

Segment	Description	Portion Causing HSS Designation	Operator Action	Consequence w/o OA	Max leak flow rate	Flow rate resulting in HSS RRW
2-067-003	1"-10" Discharge line from CS pump room cooler 2A, RHR pump room coolers 2A & 2C, RHR pump seal heat exchangers 2A & 2C	1" discharge from the RHR Pump Seal HTX.	Isolate supply (close 550, 560, 567)	FLRB2 - Flood of Reactor Bldg	100	10
2-067-004	3"-8" Cross-tie line for RCW backup from FCV-67-51 to 2-67-888, 580, and 581	8" cross-tie line for RCW backup from FCV-67-51 to 2-67-888, 580, and 581.	Close 580 & 581	FLRB2 - Flood of Reactor Bldg	80	80
2-067-005	½"-8" Cross-tie line for RCW backup from 2-67-575 to 2-67-6926, FCV-67-51, 2-67-876 and 2-67-888	8" cross-tie line for RCW backup from 2-67-575 to2-67-6926, FCV-67-51, 2-67-876, and2-67-888.	Isolate segment (close 575)	FLRB2 - Flood of Reactor Bldg	80	80
2-067-007	1"-4" Discharge line from CS pump room cooler 2B, RHR pump room coolers 2B & 2D, RHR pump seal heat exchangers 2B & 2D, and SDBR ACU condenser 2B & 2A	1" discharge line from RHR pump seal heat exchangers 2B & 2D	Isolate supply (close 593, 613, 609)	FLRB2 - Flood of Reactor Bldg	40	10
2-071-011	6" RCIC suppression pool suction line from CS suction line off ring header to FCV-71-18	6" RCIC suppression pool suction line from CS suction line off ring header to FCV-71-18.	Close 75-1	FLRB3S- Drain of Supp pool to Reactor Bldg	60	60
2-074-008	24"-30" RHR pumps A & C supply line to FCV-74-1 & 12 from suppression pool isolation valve 74-85	4" drain to the RHR drain pumps.	Close 74-85	FLRB3S- Drain of Supp pool to Reactor Bldg	300	40
2-074-009	24"-30" RHR pumps B & D supply line to FCV-74-24 & 35 from suppression pool isolation valve 74-88	4" drain to the RHR drain pumps.	Close 74-88	FLRB3S- Drain of Supp pool to Reactor Bldg	300	40

Discussion:

The BFN Reactor Buildings are separated by flood doors; consequently, flooding in one unit will not impact the other units. All Reactor Building system disabling leaks result in water moving through stairwells and other large openings to elevation 519.00. The four RHR pumps, four Core Spray pumps, and both HPCI and RCIC pumps are located on elevation 519.00. Loss of these ECCS pumps is the consequence which renders the seven segments listed above high safety significant (HSS) without Operator action.

Loss of all ECCS pumps for each unit would conservatively occur at a flood level of four feet (RHR/Core Spray pump centerline) above 519.00 floor (i.e., elevation 523.00). Elevation 519.00 has a floor area of 17,750 square feet. Conservatively, 375,000 gallons of water are needed to raise the flood level in the Reactor Building to Elevation 522 feet, 3 inches. Use of this level versus elevation 523.00 provides additional conservatism in the time calculated to allow for Operator action. There are 12-inch curbs installed between the ECCS corner rooms and the torus room. There is also a 18-inch curb between the HPCI and RHR rooms. These curbs are not considered for the scenario in question since the flood level is well above the curb height in each of the rooms.

Pipe Segments Reduced From HSS To LSS Based On Operator Action

Discussion (continued):

The largest disabling leak for the seven segments listed above is 80 gallons per minute (gpm). However to ensure conservatism, the largest flow rate of 300 gpm for a system disabling crack was used. In addition, the Reactor Building sump pumps (in excess of 300 gpm capacity), which are highly reliable components within the scope of the BFN Maintenance Rule program, were considered unavailable. Further, the multiple level indications of Reactor Building flooding were also ignored. These include the items discussed in detail in TVA's October 16, 2000 response, e.g., sump pump level switches, general area level switches.

The time required to disable the ECCS pumps is found by dividing the volume of 375,000 gallons by 300 gpm, divided by 60 minutes/hour, which results in over 20 hours of leakage. Therefore, there is no creditable scenario where a disabling crack will result in a loss of all ECCS pumps.