October 15, 19

Mr. J. A. Scalice
Chief Nuclear Officer
and Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: BROWNS FERRY UNIT 3, PROPOSED RISK-INFORMED INSERVICE INSPECTION PROGRAM, REQUEST FOR ADDITIONAL INFORMATION (TAC NO. MA5355)

Dear Mr. Scalice:

By letter dated April 23, 1999, you submitted a proposed risk-informed inservice inspection (RI-ISI) program for Browns Ferry Unit 3 (BFN3). At a meeting on September 20, 1999 your staff provided additional information which provided us with a better understanding of the proposed program. Subsequently, our reviewers have identified important unreported differences between your proposed RI-ISI program, and the Westinghouse Owners Group (WOG) methodology upon which it was based. The staff has concluded that in order to continue to review your RI-ISI program for BFN3, it is necessary that you (1) address the deviations from the approved WOG methodology listed in the enclosure, (2) evaluate their impact on your RI-ISI program and (3) make any required adjustments in the ISI program, so that it is consistent with the approved methodology. For the staff to complete its safety evaluation by the agreed-upon schedule of December 31, 1999, we need a response addressing these issues by November 1, 1999.

If you choose to use a methodology which is not consistent with the approved WOG methodology, then you should provide a detailed submittal similar to those provided by the pilot applications. The staff would then perform a review of your complete methodology and inform you of the expected completion schedule upon the receipt of your RI-ISI application.

Our request for additional information is enclosed. These questions were discussed with John Sparks and Duncan Massey of your staff, in a telecon on October 1, 1999. If you have any questions regarding this issue, please contact me at 301-415-3026.

Sincerely,

Original signed by:
William O. Long, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-260 & 50-296

Enclosure: Request for Additional Information

cc w/enclosure: See next page

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

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Project Directorate II
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REQUEST FOR ADDITIONAL INFORMATION
BROWNS FERRY UNIT 3
REQUEST FOR CODE RELIEF DATED APRIL 23, 1999

BACKGROUND

The nuclear industry has developed two methodologies for implementing risk-informed inservice inspection (RI-ISI). One methodology has been jointly developed by American Society of Mechanical Engineers (ASME) and Westinghouse Owners Group (WOG) and the other methodology is being sponsored by Electric Power Research Institute (EPRI). The U.S. Nuclear Regulatory Commission staff and industry spent a considerable amount of time and effort to review the methodologies developed by WOG and EPRI and to resolve all the issues that were raised during the review. The staff and the industry then jointly developed a template for the submittal of licensees' RI-ISI programs to facilitate submittals and expedite staff reviews. The main objective of the template was that if a licensee develops an RI-ISI program in accordance with the approved methodology, the licensee would be able to submit a greatly reduced amendment request and the staff would be able to complete its review in an expedited manner.

For the Browns Ferry Unit 3 (BFN3) RI-ISI program, your submittal was based on the template developed for RI-ISI submittals and your letter dated April 23, 1999, stated that the RI-ISI program has been developed based on the approved WOG methodology. Although your submittal listed three deviations from the approved WOG methodology, subsequent meetings between the staff and TVA personnel identified four additional important deviations discussed below. Based on the review performed to date, it appears that the results of the proposed BFN3 RI-ISI program are different from those expected by the staff since the approved WOG methodology was not adhered to in developing the program.

STAFF QUESTIONS

1. In the approved WOG topical report WCAP-14572 methodology, segments are selected according to failure consequence. The BFN3 methodology further divides segments based on consequences into smaller segments with unique degradation mechanisms. The effect of this deviation from the approved WOG methodology is unclear. The BFN3 methodology may break a single WCAP high safety significant segment into two or more segments. Because segments without degradation mechanisms have very low failure probabilities, it would appear that most of the "new" segments would be of low safety significance. This difference is most likely the reason that TVA has no segment sections in zone 1(B). Because each segment section in zone 1(B) has one weld inspected in the WCAP methodology, the TVA method appears to result in fewer welds to inspect. Another potential impact is that one high safety significance (HSS) segment under the WOG methodology may, under the BFN methodology, become three (or more) smaller segments, two with high failure probabilities and one with a low failure probability. Because we are working with relative results the impact would be to raise the total core damage frequency and large early release frequency (CDF/LERF) and decrease the individual importance measures. Please provide an assessment of the impact of this

change in the methodology on the number and distribution of proposed weld inspections as compared to the WOG methodology.

2. The WOG methodology includes the effect of augmented inspections for intergranular stress corrosion cracking (IGSCC) and flow assisted corrosion (FAC) in its baseline Section XI calculations while TVA excludes them. The baseline calculations determine which segments are HSS and need to be inspected. The exclusion of the effects of the augmented IGSCC and FAC inspections in the baseline calculations greatly increases the failure probability of segments exposed to these degradation mechanism which, in turn, greatly increases their importance. This is probably the reason that the only segments having HSS are those subject to FAC and IGSCC, and the only elements subject to inspection, with one exception, are those in the current FAC and IGSCC augmented inspection programs. The WOG methodology included the impact of IGSCC and FAC inspection in its baseline calculations, which reduced the relative contribution from these segments allowing other degradation mechanisms to be represented in the HSS category. It is noteworthy that TVA includes credit for microbiologically induced corrosion (MIC) augmented inspections in its baseline calculations and, because of this credit, finds only low safety significance (LSS) segments in several support systems that the probabilistic risk assessment (PRA) has found to be very important. TVA should assess the impact of this change in methodology on the number and distribution of proposed weld inspections and report the results to the staff.

3. The WOG methodology estimates the CDF and LERF for the current and proposed programs at the system and plant level. These estimates are subtracted from each other to estimate the change in CDF/LERF. The topical report provides acceptance criteria for the system and plant level change estimates. The BFN methodology produces estimates that it labels "detected CDF and LERF." TVA has stated that "detected CDF and LERF" is different from "CDF and LERF." Without the different estimates of CDF and LERF, and the changes in risk obtained by subtraction, the proposed change in risk cannot be compared with the WOG acceptance criteria, nor with that of Regulatory Guide 1.174. Furthermore, if "detected CDF and LERF" are being estimated, as opposed to CDF and LERF, what is the relationship between the risk reduction worth (RRW) being calculated and plant risk? What relationship does it have to the quantity calculated by the WOG methodology, and why is it a suitable measure to identify the safety significance of the segments? Please provide an assessment of the impact of this change in methodology on the number and distribution of proposed weld inspections and report the results to the NRC staff.

4. The WOG methodology is based on the Structural Reliability and Risk Assessment (SRRA) code. The staff recognizes that TVA states in the submittal that the WinPRAISE code is the Windows-based version of the PRAISE code. We note, however, that the SRRA code was modified as a result of the WOG review to produce results that more fully support RI-ISI.

   A. The SRRA calculation yields the probability of pipe rupture over a 40-year life span with or without inspection as two different numbers. That is, when inspection is performed, crack growth is monitored and, if a crack grows to a

detectable size, it will (with some probability of detection) be detected and repaired. Please describe if the WinPRAISE code has this feature, and, if not, how the difference in failure probability with and without ISI is determined.

B. During the WCAP review, the staff determined that large leak probabilities were much larger then rupture probabilities, especially for large pipes. Consequently the SRRA code calculates both the probability of rupture (which requires a design-limiting event) and the probability of a large leak. Large leak is defined as the leak rate which would disable the system function. Crack leak rate is determined from the system pressure and crack size. Please describe what is used to define pipe failure in the WinPRAISE code.

C. The SRRA code included importance sampling, and increased the number of samples, as necessary, to always provide a pipe failure estimate, i.e., there were no zero pipe failure probability estimates. Some manipulation is involved because even very large weld failure estimates of 1E-5/yr (or 4E-4 over 40 years) would normally require tens of thousands of simulations without importance sampling. Your WinPRAISE code often yields zero pipe failure estimates which does not provide confidence that the Monte Carlo simulation is continued until a stable result is obtained. Please describe what techniques the WinPRAISE code uses to reduce the number of sample runs needed for the calculations of pipe failure probability.

D. The staff expressed a number of concerns regarding the values of the input parameters used in the SRRA code. These concerns are discussed in detail in the WCAP-14572 safety evaluation report, and the accepted resolution of the concerns by the WOG are discussed in detail in Supplement 1 to WCAP-14572, Revision 1-NP-A. Please confirm that selection of the input values used in your calculations are consistent with the resolutions described in the Supplement, and provide a description of any inconsistencies.

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BROWNS FERRY NUCLEAR PLANT

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