August 4, 1993

Docket No. 50-260

Dr. Mark O. Medford, Vice President
Technical Support
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Dr. Medford:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE BROWNS FERRY UNIT 2 INDIVIDUAL PLANT EXAMINATION (TAC NO. M74385)

By letter dated September 1, 1992, the Tennessee Valley Authority (TVA) submitted its Individual Plant Examination (IPE) for the Browns Ferry Nuclear Plant (BFN) Unit 2. The NRC staff has examined your submittal and determined that additional information, as described in the enclosure, is required to complete its review. These topics were discussed with your staff during a telephone call on July 8, 1993. In order to support our review schedule, please provide your response within 45 days of receipt of this letter.

Please contact Joe Williams at (301)504-1470 if you have any questions regarding this request. This request affects nine or fewer respondents, and therefore is not subject to Office of Management and Budget review under P.L. 56-911.

Original signed by Thierry M. Ross for

Frederick J. Hebdon, Director
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Enclosure:
As Stated

cc w/enclosure:
See next page

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REQUEST FOR ADDITIONAL INFORMATION

BROWNS FERRY UNIT 2 INDIVIDUAL PLANT EXAMINATION

The questions below are arranged by functional review area. Questions F.E.1 through F.E.10 pertain to the "front-end" or Level 1 Probabilistic Risk Analysis (PRA) evaluation. Questions H.F.1 through H.F.8 apply to human factors analysis, and questions B.E.1 through B.E.9 apply to the "back-end" or containment performance analysis.

F.E.1

Per Generic Letter 88-20 and NUREG-1335 (Section 2.1.2.3), licensees are required to provide a "description of the walkthrough activity of the IPE team, including the scope and team makeup." Provide a summary description of the Browns Ferry Unit 2 walkthrough activities, including the scope, team makeup, and summary list of findings made during each walkthrough. Explain how these findings were considered in the Individual Plant Examination (IPE).

F.E.2

Per Generic Letter 88-20 and NUREG-1335 (Section 2.1.2.3), the IPE should include "a concise discussion of the process used to confirm that the IPE represents the as-built, as-operated plant." It is not clear how the Browns Ferry Unit 2 IPE assured that the analysis represents the "as-built, as-operated plant," after the extensive modifications performed during the extended plant shutdown. Please provide a concise summary description to address this issue. Include in your description details regarding the human reliability (operational) assessment portion of the IPE.

F.E.3

To what extent did the personnel listed in Table 5-2 participate in both the IPE development process and the independent review? Were event trees reviewed by any other utility personnel aside from a senior reactor operator (SRO)? Provide a sample of peer-review tools used, such as procedures and checklists. Provide a summary of the major findings of the peer-review activity. Discuss how these findings were resolved.

F.E.4

Provide additional discussion on the flood cause/impact analysis method used to identify potential design and operational problems. Discuss the effect of water intrusion on motor control centers (MCCs) and distribution panels following reactor building floods, and whether your February 10, 1992, facility walkdown looked for such problems.

F.E.5

Provide the IPE results related to overfilling scenarios. In particular, provide results from overfilling the Nuclear Steam Supply System (NSSS), including effects on systems which interface with the main steam (MS) system, such as high pressure coolant injection (HPCI) or the reactor core isolation cooling (RCIC) system.

F.E.6

The containment interface trees and plant damage state assignment trees of the submittal seem to only address scenarios in which core damage has occurred. It appears that the core vulnerable sequences (i.e. containment failure) are not developed in these trees [IPE Section 3.1.3]. However, the Level 2 analysis considers sequences for which core damage occurs as result of containment failure [IPE Section 4.3.3]. It is not clear how the Level 1 Plant Damage State (PDS) assignment process was performed for those sequences in which containment failure precedes core damage. For example, if containment cooling is lost with suppression pool recirculation available, what fails first, the containment due to steam overpressure or the core cooling systems due to suppression pool overheating? Provide a discussion, including an example of this process.

F.E.7

Provide a discussion of the operability status (failure due to pump cavitation and inadequate net positive suction head (NPSH) requirements, if any) of the HPCI system, the RCIC system, the low pressure coolant injection (LPCI) system, and the low pressure core spray (LPCS) system for the following small loss of coolant accident (LOCA) long-term scenarios:

1. Reactor is pressurized and containment is not failed due to over pressure and over temperature.

2. Reactor is depressurized and containment is not failed due to over pressure and over temperature.

F.E.8

Discuss whether any design and operational enhancements regarding the Browns Ferry Unit 2 containment venting scheme during a pool overpressurization event were identified during the IPE.

F.E.9

Provide a discussion of the IPE treatment of zone-specific room heatup with respect to Class 1E and Non-Class 1E switchgears (as applicable to the Browns Ferry Unit 2 facility). Include zone-specific heatup effects on supporting systems and initiating events.

F.E.10

Provide a summary description of the method used to determine the time-dependent unavailability of the battery system, including common cause failures.

H.F.1

The quantification of routine human actions was performed using generic error rates from Table B-1 of the submittal. The Table B-1 error rates were taken from a source referenced as a Pickard, Lowe, and Garrick (PLG) letter to Tennessee Valley Authority (TVA). Explain the underlying data source(s) of Table B-1 in the submittal, and the derivation of the generic error rates listed in this table.

H.F.2

A proprietary data base (Reference B-6) was used to calibrate the mathematical formula used to derive dynamic and recovery action errors. Explain the technical bases of the database of Reference B-6, and how the database is used to perform the calibration of the failure likelihood index.

H.F.3

The IPE used three different operator groups to rate the performance shaping factors (PSF) for the dynamic (i.e., response to initiating event) actions, and only one group for the recovery actions. Explain the difference in the analytical approach, including measures taken to assure that one-group biases did not influence the derived human error rates.

H.F.4

Per NUREG-1335 (Appendix C, Section 9, pg. C-19), the recovery actions for which the IPE takes credit should have written procedures. The Browns Ferry Unit 2 IPE modeled some "nonprocedural-guided actions." Please discuss the effect of these nonprocedural-guided actions to the total risk reduction. Discuss measures taken to assure consistency of the IPE with any guidance and or training available to personnel for these actions.

H.F.5

Explain the error rates derived for the recovery actions HOVS1 and HOVS2 that are "nonprocedural-guided" actions and must be performed within 2 minutes.

H.F.6

Provide information regarding the review of the Human Reliability Analysis (HRA) portion of the IPE, including the scope, comments, and comment resolution.

H.F.7

NUREG-1335 (page 2-8) states that core damage sequences screened out due to low human error rates should be discussed in the submittal, including the timing and the complexity of the postulated actions. Provide a discussion of how the Browns Ferry Unit 2 IPE screened core damage sequences that include human actions, especially sequences that include more than one human action.

H.F.8

Clarify the discussion of the automatic depressurization system (ADS) inhibit in Section 6.3, including the nature of the problem, the rationale as to why this "potential enhancement" (or existing operational feature) resolved the problem, and the status of the proposed enhancement.

B.E.1

As shown in Table 4.8-1 for the split fraction for the Top Event DS, the linear failure rate for a case of high vessel pressure is higher than a corresponding low pressure case. Please explain the reasons for this.

B.E.2

With respect to Figure 4.9-1 of the submittal, explain the following:

1. As noted in page 4.9-4, a state "T," thermally-induced late containment failure, is defined for the sixth character of release category definition. However, the "T" state is not used in Figure 4.9-1.

2. The state on the second line under Top Event 3, "Containment status: intact, vented, or failure time," appears to read as "E." Please explain. Is this correct?

B.E.3

Provide a complete list of radionuclide groups including tellurium, strontium, ruthenium, lathanum, cesium, and barium for release categories.

B.E.4

What is the contribution to early containment failure from liner failure by corium thermal attack?

B.E.5

How important to preventing early containment failure or reducing the radionuclide releases is the operator action of flooding the drywell?

B.E.6

Define symbols used in figures (e.g., Figure 4.7-2).

B.E.7

The description of Table 4.7-2 on page 4.7-6 of the submittal does not reflect the actual contents of Table 4.7-2. Clarify the description.

B.E.8

As noted in Tables 4.8-1 through 4.8-5, no credit was taken for operator venting (i.e., dirty vent failure (DVF) probability = 1.0) for the five top key plant damage states (KPDS) which consist of 96.3 percent total core damage frequency: For KPDSs PIH, MIA, and NIH, "[It was assumed] for extended blackout cases that dirty venting would not occur due to electric power and plant air vent valve dependencies." For KPDSs OIA and PID, it is stated that "In the absence of hard vents no manual venting is deemed possible." Is there a plan for procedural or hardware improvements for venting?

B.E.9

The KPDS MKC which represents the group for which containment failed early, or is not isolated, has a conditional core damage frequency of 0.0082. What is the fraction of this for containment isolation failure?