June 23, 2010

Mr. Ashok S. Bhatnagar
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6A Lookout Place
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SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 – SUPPLEMENATAL REQUEST FOR ADDITIONAL INFORMATION REGARDING INDIVIDUAL PLANT EXAMINATION (TAC NO. ME3334)

Dear Mr. Bhatnagar:

In its letter dated February 9, 2010, as supplemented on June 8, 2010, Tennessee Valley Authority (TVA) provided the results of the Individual Plant Evaluation for Watts Bar Nuclear Plant, Unit 2. In U.S. Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," dated November 23, 1988, as supplemented, the staff requested licensees and construction permit holders to perform a systematic examination to identify any plant-specific vulnerabilities to severe accidents and report the results to the NRC.

The NRC staff is reviewing the information provided by TVA and has determined that it needs additional information to complete its review. The specific information is detailed in the enclosed supplemental request for additional information (RAI). In this regard, the NRC staff requests a response to this RAI within 30 days of receipt of this letter.

If you should have any questions, please contact me at 301-415-1457.

Sincerely,

/RA/

Patrick D. Milano, Senior Project Manager
Watts Bar Special Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-391

Enclosure: RAI

cc w/encl: Distribution via Listserv

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OFFICIAL AGENCY RECORD

REQUEST FOR ADDITIONAL INFORMATION

WATTS BAR NUCLEAR PLANT, UNIT 2

INDIVIDUAL PLANT EXAMINATION AND PROBABILISTIC RISK ASSESSMENT

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-391

In a letter dated February 9, 2010, as supplemented on June 8, 2010, Tennessee Valley Authority (TVA) provided the results of the Individual Plant Evaluation (IPE) for Watts Bar Nuclear Plant, Unit 2. TVA provided an IPE summary report and stated that the IPE was performed in accordance with the applicable portions of American Society of Mechanical Engineers Standard RA-Sb-2005, "Standard for Probabilistic Risk Assessment [PRA] for Nuclear Power Applications," and NRC Regulatory Guide (RG) 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities."

In its June 8 letter, TVA responded to a request for additional information (RAI) from the U.S. Nuclear Regulatory Commission (NRC) staff. The NRC staff has reviewed the summary report and the RAI response and finds that additional information is needed to complete its review. The specific information is described as follows in this supplemental RAI:

1. With regard to parameter uncertainties, the PRA standard requires, at a minimum, quantification of a core damage frequency/large early release frequency (CDF/LERF) point estimate using the mean values of the basic events (HLR-QU-A). For Capability Category III, a propagation of the uncertainty intervals is required, which takes into account the state of knowledge correlation. For model uncertainties (HLR-QU-E), the PRA standard only requires the identification of sources of model uncertainty and related assumptions as well as where the PRA model is affected. The scope of finding and observation (F&O) 3-6 appears to go beyond the minimum requirements of the PRA standard, but the sources of model uncertainty are not clearly identified. Provide a list of the sources of model uncertainty, and identify where the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criteria, and introduction of a new initiating event (IE)).

2. The NRC staff position in RG 1.200 is that the PRA model represents the as-designed, as-built, and/or as-operated plant. An accurate representation of the plant is essential when attempting to construct a model that satisfies each of the technical elements in the standard. Identify the assumptions made relative to the as-built and as-operated plant that could significantly impact the results and identify the resulting impacts, given these assumptions (Reference F&O 7-11, supporting requirement (SR) IFPP-A4).

3. Regarding all F&Os describing a resolution wherein changes have already been made (e.g. – F&Os 1-7, 3-13, 3-18, etc.), provide a description of the actual resolution and how, in practice, these changes were actually executed, along with an actual description of the change.

4. The general purposes of the IPE, as noted in NRC Generic Letter (GL) 88-20, is for the applicant/licensee to:
   1. Develop an appreciation for severe accident behavior
   2. Understand the most likely severe accident sequences
   3. Gain a more quantitative understanding of the overall probabilities of core damage and fission product releases.
   4. If necessary, reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

   With regard to fission product release, the PRA standard only analyzes large early release. It appears from TVA's submittal that all potential releases (large and small early, large and small late) were evaluated. However, the nature of the evaluation of releases is unclear, as well as whether any peer review was performed on such an evaluation. Consequently, the staff finds it is difficult to determine if the four purposes of the GL were adequately addressed. Describe how these purposes were achieved, in that the results of the analysis are technically adequate. In that regard, what method was used to analyze these severe accidents (beyond LERF)? How was the technical adequacy of this analysis evaluated since the peer review does not go beyond LERF?

5. The PRA standard requires that the operating systems be evaluated to determine if failure will result in an initiating event. It is not clear from F&O 4-14 what systems were omitted from this review (e.g., loss of component cooling water). In particular, failure of support systems can be risk significant. Provide the list of systems that were systematically evaluated for the determination of initiating events. Additionally, provide the justification for system screening.

6. The F&O problem statement is difficult to understand in F&O 4-3 without additional information. Provide a more comprehensive description of the problem statement associated with this F&O. In addition, provide the results of the recalculation and identify the extent to which the initiating event frequency changed and its impact on the final results (e.g., different contributors to CDF or LERF).

7. When using a fault tree approach to calculate an initiating event frequency, care must be taken to recognize that the calculation is for a frequency as opposed to a probability (such as not used in SR IE-C9). It appears from F&O 4-7, that a probability was calculated instead of a frequency. Describe in more detail the issue raised by the F&O and how it was resolved. In addition, indicate how the results and potential contributors are affected by modifying these common cause failure (CCF) events (Reference F&O 4-7, SR IE-C9/10/15).

8. F&O 5-1 is concerned with the "mission time" used for the room heatup calculation. Room heatup calculations and the time to failure of the equipment in the room from overheating can have a significant impact on the results. The staff finds confusing that this F&O has been related to SR SC-A5, which is concerned with the accident sequence mission time; that is, the time at which the plant is assumed to reach a stable state (e.g., 24 hours). The peer review finding is better related to requirements AS-B3, SY-A18, and SY-A22. Provide the basis for this finding under SR SC-A5. In addition, provide the assumptions and technical basis for the room heatup calculations.

9. Division of common cause groups for equipment in the same system may underestimate the impact of common cause and consequently underestimate the risk and incorrectly identify potential contributors. Provide a description of the in-progress resolution for F&O 1-8, and indicate whether the risk estimation and potential contributors were affected. If affected, describe the change in the results. If not affected, explain the basis for no change in the results.

10. Understanding the dependencies can help determine potential vulnerabilities associated with the plant, particularly those associated with initiation and actuation. Failure to model the actuation signal following loss of station power (LOSP) may cause some subtle, but significant dependencies to be missed. Identify those sequences potentially affected by this incomplete modeling of dependencies and their overall impact on the final results. Provide a description of the in-progress resolution for F&O 1-4 and how the results were impacted. If not impacted, explain the basis.

11. Regarding F&O 2-28, multiple human failure events (HFEs) in a single cutset can result in erroneously truncating the cutset. For example, three HFEs with human error probabilities (HEPs) of 1E-3 will result, at least, a combined HEP of 1E-9. The standard requires application of a threshold, because a combined HEP of 1E-9 implies a state of knowledge regarding human performance beyond the uncertainties. In addition, it can result in erroneous truncation of cutsets. Provide an explanation of why a threshold value (1E-5) for a lower limit was not applied to cutsets with multiple HFEs. Also, indicate what cutsets were potentially missed if a threshold value were applied (Reference F&O 2-28, SR HR-D5).

12. Generally, when identifying events as failures in a PRA, partial operation of a component is considered a failure. That is, a partially opened valve, a partially operating pump, etc. are considered failures. It appears from the following statement, "failures that would not have impacted any PRA success criteria," that this current PRA practice was not followed. Explain what is meant by "failures that would not have impacted any PRA success criteria," and provide the basis for why these failures are determined not to be applicable (i.e., why they are not included as possible failures (Reference F&O 2-3, SR DA-C4)).

13. SR DA-D1 states that, for Capability Category I, one must “use plant-specific parameter estimates for events modeling the unique design or operational features if available, or use generic information modified as discussed in DA-D2.” It is unclear how F&O 1-7 relates to SR DA-D1 since the F&O relates to the calculation of CCF probabilities, type codes assignment for basic events, and a mission time for a basic event. Please provide the basis for this finding under SR DA-D1. Provide an explanation of why the hand calculation and CAFTA calculation do not match. Further, if the hand calculation is more appropriate, indicate the impact on the final results given that hand calculations are used, the type codes are corrected, and the correct mission time is used.

14. Inappropriate truncation can result in significant accident sequences being erroneously eliminated; SRs QU-B3/LE-E4 provide the requirements for acceptable truncation. Provide the technical bases for using the same truncation limit for LERF; that is, provide the change in LERF if a lower truncation limit was issued. If the change in LERF is greater than 5 percent, identify what sequences were eliminated (Reference F&O 3-1).

15. By not including system successes, inappropriate cutsets can be generated. In turn, this can mask the real contributors to plant risk. Provide the technical justification for not accounting for system successes and provide the justification for the identified significant contributors given systems successes were not included in the evaluation of the accident sequences (Reference F&O 3-3, SR QU-B6).

16. A PRA is a complex model wherein a small error can lead to a gross misrepresentation of the plant risk profile and its contributors. As such the PRA standard requires a review of a sample of significant accident sequences/cutsets sufficient to determine whether the logic of the cutset or sequence is correct. It is unclear from F&O 3-15 the extent to which a review of the cutsets was performed. Describe the internal review that was performed and the results of the review (e.g., what inappropriate cutsets were identified). In addition, if an internal review was not performed, provide the technical bases explaining how the reasonableness of the cutsets was determined and that the cutsets reflect the risk of the plant.

17. For F&O 5-13, it appears in the resolution that the concern raised was addressed (e.g., CAFTA model revised); however, it is not clear if the revision appropriately addressed the concern. Describe the actual resolution implemented and how it addressed the issue raised by the F&O. In addition, state if any results were impacted by this issue.

18. The peer review in F&O 7-19 appropriately noted that a medium loss-of-coolant accident (LOCA) is generally considered not large enough to result in low pressure that is sufficient to alleviate a direct containment heating (DCH) concern. Provide justification for the classification of the 480-gpm pump seal LOCA as a low-pressure (i.e., medium LOCA) as opposed to a high-pressure scenario. This description should include a discussion of the relationship of this event to DCH and a justification of why this does not challenge the containment (Reference SR LE-B2).

19. HAPRZ is noted as a key operator action, which implies that it could have a significant impact on the results (e.g., the actual LERF and its contributors). At this point in the accident, there have been both equipment and operator failures and it is not apparent what has occurred to improve the operator's understanding and ability to take control of the accident. Provide the basis for the estimated 4.4E-4 HEP and provide an explanation of why the operator, for this HFE, is less likely to fail with a probability that is two orders of magnitude below the probability for HAOB1. In addition, provide an explanation of the effect on the calculated LERF and identification of the significant contributors if a much higher HEP value (approximately 1E-2) were used. Provide a description of the in-progress resolution for F&O 5-8.

20. A pipe break of less than 2 to 3 inches is not the appropriate basis for screening. Flooding that can negatively impact equipment can occur from this size pipe break depending on the location of the break, inventory, the size of the flood area, the location of the equipment, the flood type, etc. Provide the technical basis for excluding pipe breaks less than 3 inches (Reference F&O 7-4, SR IFSO-A1).

21. Identifying the characteristics of the flood release and the capacity of the sources is fundamental to performing an acceptable flooding analysis. Understanding the type of flood (e.g., leak, rupture, spray), its potential flow rate, and its capacity is fundamental in identifying the flood scenarios and ultimately the internal flood contribution to the CDF. Justify why it was not necessary to understand the characteristics of the release and the capacity of the source (Reference F&O 7-12, SR IFSO-A5).

22. F&O 7-15 does not appear to address the PRA standard SR IFSO-B3, which requires that the sources of model uncertainty and related assumptions associated with the internal flood sources be documented. The concern from the peer review does not seem to align with the identified SR from the standard. Identify the F&O finding relative to appropriate SR in the standard, describe its potential impact on the results, and describe how the actual resolution for addressing this issue.

23. The major purpose/objective of GL 88-20 is to "perform a systematic examination to identify any plant specific vulnerabilities to severe accidents." The PRA, as defined by the standard, provides an excellent systematic approach. HLR-IFSN-A requires development of the potential internal flood scenarios. The SRs provide a systematic structure for the development of the propagation paths by evaluating the individual flood sources, flood areas, and plant features. A zone-to-zone approach does not meet these requirements and, more importantly, would likely fail to identify plant weaknesses relative to internal floods. Based on the approach and the resolution used, justify the bases for no potential weaknesses in plant design and operation from internal floods (Reference F&O 7-1, SR IFSN-A1).

24. F&O 7-5 finding is related to Capability Category II for SR-IFEV-A2. It is not clear whether Capability Category I was met for this SR. Identify whether Capability Category I was met and the basis for using that category relative to the purpose of the GL.

25. The concern raised in F&O 7-9 is not clear. The basis given in the F&O is SRs IFSN-A9 and IFQU-A5. However, these SRs (IFSN-A9 and IFQU-A5) require the analyst to perform calculations for flood rate, time to reach the susceptible equipment, and the structural capacity of systems, structures, and components (SSCs). The analyst is also required to perform any human reliability analysis on internal flood HFEs in accordance with 2-2.5. The relationship of these two SRs to the concern raised in the F&O is not clear. Further, the staff finds that the actual issue raised in the F&O is not clear. Provide a detailed description of the issue identified by the peer review, describe how the issue is related to both IFSN-A9 and IFQU-A5, and describe the actual resolution being pursued.

26. The effects of an internal flood create new conditions in the plant that will affect SSC availability. The PRA standard appropriately notes additional performance shaping factors such as additional workload, timing, crew availability, which would affect operator (control room) performance. Justify why main control room operator actions are not impacted (e.g., no dependence) from an internal flood. In addition, provide the effect on the calculated HEP given that the operator performance is affected by the internal flood and identify how the results would be impacted (Reference F&O 7-6, SR IFQU-A6).

27. It is not clear how the peer review could indicate that SR IFQU-A10 was met when F&O 7-8 clearly states that the peer reviewer could find no evidence that the LERF analysis included internal floods. Further, in order to understand how and ensure that a more quantitative understanding of the overall probability of fission product releases was achieved, it is important that the PRA model addresses the potential contributors (e.g., internal flood). Provide a more complete description of how internal flooding contributes to the LERF sequences (e.g., the development of event trees, event sequences, timing, etc.).