August 20, 1993

Docket No. 50-390

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

Dear Dr. Medford:

SUBJECT: WATTS BAR UNIT 1 - REQUEST FOR ADDITIONAL INFORMATION ON THE INDIVIDUAL PLANT EXAMINATION (TAC NO. M74488)

Based on its ongoing review of the Watts Bar Nuclear Plant Individual Plant Examination (IPE) submittal and associated documentation dated September 1, 1992, the staff developed a request for additional information (RAI). The RAI relates to the internal event analysis in the IPE, and the containment performance improvement (CPI) program.

If you need clarification on this, we will be available to discuss it with you in a conference call. Please respond within 60 days of receipt of this RAI. This RAI is covered by Office of Management and Budget Clearance Number 3150-0011, which expires May 31, 1994.

Sincerely,
Original signed by

Peter S. Tam, Senior Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Request for Additional Information

cc w/enclosure:
See next page

| OFFICE: | PDII-4/LA | PDII-4/PM | PDII-4/D |
|---|---|---|---|
| NAME | BClayton | PTam:as | FHebdon |
| DATE | 8/20/93 | 8/20/93 | 8/20/93 |

DOCUMENT NAME: IPERAI





DF01
AA3

Distribution

Docket File
NRC & Local PDRs
WBN Reading
J. Partlow
S. Varga
G. Lainas
F. Hebdon
B. Clayton
P. Tam
OGC
ACRS (10)
E. Merschoff, RII
P. Fredrickson, RII
J. Crlenjak, RII

ENCLOSURE

# REQUEST FOR ADDITIONAL INFORMATION

# INDIVIDUAL PLANT EXAMINATION (IPE)

# WATTS BAR NUCLEAR PLANT, UNIT 1

# DOCKET NO. 50-390

GEN 1 The IPE submittal described various reviews performed by different review teams as well as the areas of review. It concluded that the peer review did not identify major findings. However, details of the review results are not included in the submittal. Provide the results of the review, consistent with Section 2.4 of NUREG-1335, which stated that as a minimum, details of the review should be included in the submittal.

F.E.1 NUREG/CR-4550 analysis for Sequoyah included a very small LOCA, with less than 1/2 inch equivalent diameter, as an initiating event. The Watts Bar Unit 1 IPE excluded this initiating event on the basis that it is within the makeup capacity of the normal charging system. Discuss the significance of very small LOCAs; including the differences between the Sequoyah and Watts Bar mitigation capability for "less than 1/2 inch LOCA," particularly the differences between the charging systems.

F.E.2 Some of the shared or common systems between the Watts Bar Units 1 and 2 are considered and credited for the Unit 1 IPE submittal. They include unit station service transformer, swing diesel, and raw cooling water system. Does the system level success criteria consider maintaining Unit 2 in shutdown while mitigating an accident in Unit 1? Provide specific information including operating and surveillance procedures of Unit 2 pertinent to the operation of Unit 1. [Multiunit considerations are addressed in Section 2.1.4, guideline #3 of NUREG-1335.]

F.E.3 Identify any Unit 2 systems, other than those mentioned above, that can be cross-tied to support Unit 1. Clarify the assumptions in the analysis regarding use of Unit 2 systems for Unit 1, particularly the use of onsite electrical power, both ac and dc. [Support systems are discussed in Section 2.1.4, guideline #1 of NUREG-1335.]

F.E.4 Describe the process you used to (1) evaluate contributors to dominant sequences, and (2) determine the need for safety enhancements to reduce their significance.

F.E.5 The FSAR success criterion for Residual Heat Removal (RHR) injection is the availability of two-out-of-four injection paths, while in Table A-2 of the submittal it is shown as one-out-of-four. Explain the basis for this difference.

F.E.6 Provide the justification for deviating from the FSAR success criteria for mitigating small and medium LOCAs. The FSAR states that one safety injection (SI) pump and one Chemical & Volume Control System (CVCS) pump are required to mitigate these LOCAs. The IPE assumes that for a small LOCA (less than 2 inch equivalent size break), only one of any four total SI/CVCS pumps are required; the IPE assumed that for a medium LOCA (2 to 6 inch equivalent size break), any two of the four SI/CVCS pumps are sufficient. Briefly discuss the thermal hydraulic analysis that supports the IPE success criteria for the smallest of the medium LOCAs with no heat removal from the steam generators other than that provided by their initial inventory. [Section 2.1.3, guideline #1 of NUREG-1335 discusses documentation of success criteria.]

F.E.7 It is our understanding that the steam generators at Watts Bar had been hydro-tested in 1981 and had been filled with water since then. From Table 3.3.1-4 the mean event frequency of the SGTR is 2.84E-2 per reactor year. How does this history impact the frequency of SGTR, compared to similar operating plants?

F.E.8 Provide the rationale for not modeling feed-and-bleed (SG depressurization and feedwater) as an alternate DHR function.

F.E.9 Section 3.4.1 of the submittal states that the time required to lose RCP seal integrity upon losing RCP thermal barrier cooling is at least 2 minutes. However, a realistic value of 10 minutes was assumed for the sequence modeling. Explain the difference.

F.E.10 The differences between the accident sequences no. 1 and 3, and sequence nos. 4 and 7 are refilling of the RWST. The sequence descriptions imply that sequences 1 and 4 successfully provided makeup water for the RWST refills, and sequences 3 and 7 failed to provide the refill water to the tank. However, no specific descriptions of the RWST refill operations were given for sequences 1 and 4. In fact, it appears that sequences 1 and 3 are identical. Provide step-by-step clarification of the sequences.

F.E.11 Table 3.4-8 shows that total losses of Unit 1 CCS contribute 1.3E-4 per reactor year to core damage frequency. The sensitivity analysis (Section 3.4.3.2.1) states that the proceduralized operator action to align the train A centrifugal charging pump to the ERCW in the event of CCS failure is modeled. The guaranteed failure of this action would raise the core damage frequency by an increment of 1.5E-3 per reactor year. However, Table 3.4.7 shows that the ninth-ranked, nonguaranteed, failed split fraction (SF) to importance is CCPRI, which is an operator action designator for ERCW alignment to charging pump. Also, Table 3.4-4 listed split fraction sorted by RISKMAN-generated importance, in which the CCPRI SF and frequency are given as 1.61E-2 and 2.4808E-5 respectively. The implication is that the guaranteed failure of CCPRI may increase the total CDF by, at least, 1.5E-3 per reactor year.

Explain the incremental changes of the core damage frequency by 1.5E-3 per reactor year due to CCS failure. Furthermore, explain how the procedure for CCPRI action has been implemented.

F.E.12 Explain what the effect of not tripping the reactor coolant pumps following a small LOCA would be. [Section 2.1.3, guideline #1 of NUREG-1335 discusses documentation of success criteria.]

F.E.13 Explain the control room cooling addressed in the IPE. From Section 3.1.1 of the submittal, where HVAC systems are discussed, it appears that loss of room cooling was not considered as an important initiating event because it is a well annunciated event, and because heatup calculations show there is ample time for the operator to recover. Elaborate on this issue; for example, are the actions proceduralized with adequate operator training? Also provide a brief summary of heatup calculations.

F.E.14 Loss of control air system is not included in the support and intersystem dependency analysis. However, the IPE considered the portion of the air system classified as "safety-related" in the containment response analysis. Discuss the significance of the "non-safety-related" portion of the system.

F.E.15 Provide further justification for not considering failure of RHR pumps due to overheating when on miniflow recirculation using the RWST. (This is addressed on page 3.1.2-9 of the submittal.) How is this affected by the statement on page 3.1.2-6 of the submittal that the operators will reset the phase B containment isolation signal and stop containment spray pumps once containment pressure falls below the phase B setpoint. [Section 2.1.3, guideline #1 of NUREG-1335 discusses documentation of success criteria.]

F.E.16 It appears that the IPE focused on modeling submergence-induced failures of equipment as a result of internal floods. Was consideration given to other modes of failure, such as spray? [Section 2.1.5 of NUREG-1335 addresses internal flooding.]

F.E.17 Provide a rationale for screening out failures of instrument penetrations to the vessel, from consideration as an initiating event. [Initiating events are discussed in Section 2.1.3, guideline #1 of NUREG-1335.]

F.E.18 Provide a rationale for not analyzing feedwater line break-initiated accident sequences. [Initiating events are discussed in Section 2.1.3, guideline #1 of NUREG-1335.]

F.E.19 The back-end portion of the IPE screened out Plant Damage States with frequencies less than E-7 per reactor year. What was the largest frequency Core Damage Sequence screened out as a result of this screening for Plant Damage States? [Screening is addressed in Section 2.1.5 of NUREG-1335.] In addition, provide the truncation limits used to screen Core Damage Sequences. [Screening is addressed in Section 2.1.5 of NUREG-1335.]

F.E.20 The ATWS contribution to core damage is approximately five times higher for Watts Bar than it is for Sequoyah. Discuss the Watts Bar IPE ATWS analysis, including important assumptions and data, associated calculations, and significant insights stemming from the analysis. Also discuss the following aspects of the analysis:

(1) Why does the value for failure to trip the reactor, considering both automatic and manual trips, appear relatively high?

(2) How did the ATWS model address early plant life conditions when the moderator temperature coefficient is least negative?

H.F.1 Describe the scope of the HRA review and the experience and the qualifications of the individuals reviewing the HRA portions of the IPE.

H.F.2 NUREG-1335 states "...unless proper justification is provided, all important recovery actions should have written procedures" (Appendix C, Section 9, Response to Question 9.1, page C-19). The IPE however, took credit for "non procedure-guided" recovery actions. Table B-19 included eight recovery actions that reference no procedure for guidance.

Identify and discuss the contribution of the "non procedure" recovery actions modeled in the IPE to the total core damage frequency. Also discuss the measures taken to assure that operators will be properly trained for these actions.

H.F.3 The IPE states that common cause failures at the systems level are treated either explicitly by means of identifying causes of dependent failure and incorporating them into the system or event sequence models, or implicitly by using certain parameter to account for their contribution to the unavailability of the systems. It is not clear from the Watts Bar submittal that an analysis of human errors from miscalibration was included.

For example, in the Peach Bottom NUREG-1150 analysis, examination of the maintenance practices indicated that a single crew performed the calibration of reactor pressure sensors in a single shift. Further evaluation, which included examination of the procedures, indicated a high dependence in miscalibration of these sensors. Failure of these sensors result in the failure of the LPCI and LPCS valves to open. It was also determined that the operators would fail to diagnose the cause of LPCI and LPCS failure from miscalibration with a very high probability. This failure resulted in a dominant contributor in the NUREG-1150 analysis of Peach Bottom.

Provide a concise discussion describing how the procedures and practices were evaluated such that this type of failure was not overlooked.

H.F.4 It appears that there are some inconsistencies in the use of the THERP tables in evaluating the recovery factors of "Assessment of Likelihood of Drain Plugs left in the Refueling Canal Following Refueling." The reviewers were not able to verify the table used for FHI-8 Step 10, and SI-6.28. Provide details of the calculation; include in the discussion how the quantification took into account recognition of procedure use and appropriate execution of SI-6.28. Also explain why item 4 from Table 20-6 (abnormal operating condition) applies to administrative control-type activity.

H.F.5 The methodology used for evaluating dynamic human errors (SLIM) is based on eliciting "the operators' judgement and convert their evaluations into quantitative error frequencies." An underlying hypothesis of this method is that the operators are at least licensed and the procedures are not in a developmental stage. However, the Watts Bar operators are inexperienced in operating Watts Bar. Therefore, one would assume that their input represents their state of knowledge and experience which is much smaller as compared to plants with operational experience. This lack of experience is reflected in the human error probabilities used in the IPE, which are higher than probabilities used in a typical IPE. Explain how anticipated human error probabilities will change after the operators have gained Watts Bar experience (i.e. are operators' errors comparable to those at current operating plants.)

B.E.1 General Release Category II, "Small, Early Containment Failures and Small Bypass", contains SGTR which is acknowledged to be a large early release (page 1-13). Explain why SGTR was not included in Release Category I, "Large, Early Containment Failures and Large Bypass"?

B.E.2 Page 1-14 indicates that the frequency of a large, early release is approximately 4% of CDF but page 1-15 states that probability is 6%. Please clarify.

B.E.3 Explain the difference between total containment failure and gross containment failure as shown on Figure 2-18. It is not clear why, for a given containment pressure, a gross failure (however defined) is more likely than a leak failure.

B.E.4 You state that vulnerability may exist if "...the mean large early release frequency exceeds 5E-05/yr." Discuss your reasons for defining the vulnerability threshold at this relatively high value.

B.E.5 You state on page 4.8-12 that the conditional failure probability is 0.1, given a DDT event. Discuss why this NUREG-1150 value is appropriate for Watts Bar.

B.E.6 Considering CET Top Event 3, Core Damage Arrested Prior to Vessel Breach (CV), address the following issue: For each of the KPDSs, what fraction of the top event success is due to operator action (e.g., feed and bleed), and what fraction is attributed to a reexamination of the thermal hydraulic success criteria (note pages 4.5-3 and 4)? (The submittal notes that ac power recovery after the inception of core damage but before vessel breach was not addressed in the IPE, page 4.8-17, therefore, ac power recovery does not contribute to the success of Top Event 3.)

B.E.7 Considering CET Top Event 3, Core Damage Arrested Prior to Vessel Breach (CV), also address the additional issues:

(1) It is stated on page 4.54 that detailed thermal-hydraulic analysis shows that scenarios that were originally assumed to result in core damage do not result in core damage. Provide a concise discussion of this detailed thermal-hydraulic analysis. Was the MAAP code used to justify the thermal hydraulic success criteria?

(2) Explain a failure faction of 0.00 for KPDS LNIYA when apparently, success depends on operator action.

B.E.8 What is the impact of CET Top Event 7, "No Induced RCS Hot Leg or Surge Line Failure (IP)," on the final release categories and containment failure probabilities? How important is the maintaining of a loop seal on the success of Top Event 6, "No Induced Steam Generator Tube Rupture (IS)"? Is there any consideration given to restarting the reactor coolant pumps under high temperature and dry steam generator conditions?

B.E.9 Both of the following key assumptions used in the WB1 IPE MAAP calculations (Section 4.7.1.2, pages 4.7-2) raise a question:

(1) The median containment failure pressure was assumed to be 105 psia, which is 5 psia below the lowest failure pressure given in Table 4.4-1 of the submittal report: the base slab flexure failure mode has a failure pressure of 95 psig, which converts to about 110 psia. Was a lower failure pressure intentionally used? If so, discuss other similar conservatisms incorporated into the analyses.

(2) A relatively small fraction (3%) of the dispersed debris was assumed to fragment finely and therefore participate in direct containment heating (DCH) and rapid zirconium oxidation. What is the basis for this assumption?

B.E.10 The table on page 4.6-1 of the submittal apparently contains a redundant column because the values are identical in both columns: "PDS Frequency per Reactor Year" and "KPDS Frequency per Reactor Year."

B.E.11 Section 4.6.2.9, page 4.6-7 of the submittal, notes that the containment is manually isolated for the KPDS HGI. However, the table on page 4.6-1 notes that the KPDS HGI subsumed the containment isolation failure PDSs, GNS, FCS, KTL, ETL and FNS, which together constitute 42% of the KPDS HGI core damage frequency (3.1E-6). Explain the apparent inconsistency.

B.E.12 In describing the PDS matrix in Figure 4.3-1, the steam generator cooling available in Column 2 of the table is identified as Y. This follows the explanation given in Section 4.3.2, page 4.3-6 of the submittal, which notes that "[t]he second character (Y, X, or N) denotes the status of steam generator cooling." However, Column 4 of the same figure and subsequent definitions of PDSs use the character A to represent availability of the steam generator cooling.

Although this change in definition does not affect the outcome of the IPE, consistency in nomenclature would help reduce confusion.

B.E.13 In response to the CPI Program recommendation, the WB1 IPE assumed that all of the CDF associated with KPDSs with the ignitors unavailable would result in containment failure at some time (Section 4.10.4, page 4.10). Despite this assertion in the IPE, hydrogen burn originated early containment failures appear to account for only 5% of the large, early release category. Based on the above assertion, it would appear that containment failure would result from all station blackout events. Where are the containment failures associated with station blackout events channeled, since presumably they will lead to containment failure?

B.E.14 As reported in NUREG-1150 for Sequoyah, the percentage contribution CDF associated with either early containment failure or basemat melt-through is significantly lower for WB1 than Sequoyah. Explain the possible reasons.

B.E.15 The KPDS LNIYA does not lead to core damage since core damage was arrested prior to vessel breach (note Table 1 of this report). Thus LNIYA ends up in Release Category Group IV (see page 4.117 of the submittal). However, the KPDS LNIYC does lead to core damage and ends up in Release Category Groups I and III (see pages 4.1-11 and 4.1-14 of the submittal). Explain why LNIYC goes to core melt, when LNIYA does not, when the only difference in these two KPDSs appears to be that in "A" the air return fans are available while for "C" they are not (see page 4.6-14 of the submittal).

B.E.16 Section 1.4.3.2 states that the contribution of hydrogen ignitor unavailability is relatively small to release Categories III and IV as concluded in Section 4.10 of the submittal. However, Section 4.10.6, page 4.10-5, states that the contribution of the hydrogen burn is important only in group III events. It further states that sequences without the air return fans and ignitors are substantially involved in core damage sequences typified by station blackout (11% of CDF or 3.6E-5 per reactor year). Explain the apparent contradiction.

B.E.17 Section 4.10.6 discussed the availability of the air return fans (ARFs) and hydrogen ignitors, with emphasis on manual actuation of the ARFs and ignitors during a loss of all ac power. However, the vulnerability and need for the uninterruptible power supply to the ARFs and ignitors were not discussed in the IPE.

Page 4.8-10 states that NUREG/CR-4551 for Sequoyah has treated ac power recovery during core degradation if the ignitors were not available. It further states that the Watts Bar IPE submittal did not include the power recovery aspects for conciseness. It appears that this omission is not consistent with GL 88-20, Supplement 4, on uninterruptible power supply to ignitors. Please clarify.

B.E.18 Explain under what environmental condition credit is taken for operation of the air return fans. For example, given hydrogen combustion inside containment, how is the operability of the system motors, dampers, and electrical cables assured (page 4.1-8).

B.E.19 Section 4.6.1, page 4.6-2 of the submittal notes that the KPDS EIB subsumed the PDSs, GNS, FCB, FCS, KTL, ETL, and FNS yielding a total frequency of 5.5E-6 per reactor year. This is consistent with the PDS frequency values given on Table 4.6-1, page 4.6-11. The PDS EIB frequency value of 3.7E-6 changes to 5.5E-6 for the KPDS EIB as it subsumed the six PDSs with a total frequency of 1.8E-6. This fact is not reflected in the frequency value for KPDS EIB given on the table on page 4.6-1. (Also, this table shows KPDS HGI but not EIB subsumed the other PDSs.) Explain the apparent discrepancy.

B.E.20 Page 4.7-3 indicates that "a relatively small fraction (3%) of dispersed debris was assumed to ....participate in DCH." Discuss the basis for this fraction in comparison to the range of NUREG-1150 values.

Principal Contributors: Erasmia Lois
Jin Chung