May 10, 2006

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

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 1 — REQUEST FOR ADDITIONAL INFORMATION REGARDING THE NUCLEAR REGULATORY COMMISSION STAFF AUDIT ON THE CONTAINMENT SUMP MODIFICATIONS (TAC NO. MC4730)

Dear Mr. Singer:

The Nuclear Regulatory Commission staff is continuing its audit of the proposed modifications to the containment emergency sump at the Watts Bar Nuclear Plant, Unit 1 to address Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance." In order for the staff to complete its review, we will need responses to the enclosed request for additional information. Based on discussions with your staff, it is our understanding that you plan on responding by approximately June 15, 2006.

Please feel free to contact me at 301-415-1364 if you have any questions regarding the enclosure.

Sincerely,

/RA/
Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosure: Request for Additional Information

cc w/enclosure: See next page

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Mr. Karl W. Singer
Tennessee Valley Authority

WATTS BAR NUCLEAR PLANT

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REQUEST FOR ADDITIONAL INFORMATION

WATTS BAR NUCLEAR PLANT, UNIT 1

NRC AUDIT OF CONTAINMENT SUMP MODIFICATIONS

RESULTING FROM GENERIC LETTER 2004-02

DOCKET NO. 50-390

Break Selection and Zone of Influence Analysis

1. Tennessee Valley Authority (TVA, the licensee) stated that because the quantity of reflective metallic insulation is not a significant contributor to head loss, and the quantity of fibrous material, Min-K, would remain relatively unchanged for each break, the bounding case for each loop is the reactor coolant system break which would destroy the most coatings. The licensee indicated that a thorough analysis showed that a break in each of the crossover legs near the steam generator nozzle yielded the most coating debris due to the size of the zone of influence (ZOI) applied in the analyses. The Nuclear Regulatory Commission (NRC) staff (the staff) determined that such an analysis was not clearly documented in the calculations and information provided for the staff's audit. Please provide the referenced analysis to verify that the limiting break is at the base of the steam generator.

2. As discussed in Sections 3.1 - 3.4 of Watts Bar calculation ALION-CAL-TVA-2739-03, the licensee credits the reactor annulus and refueling canal as robust barriers in the analysis. As stated, the licensee's analysis showing that a break in each of the crossover legs near the steam generator nozzle yielded the most coating debris was not clearly documented in the calculations and information provided for the staff's audit. Therefore, Watts Bar calculation ALION-CAL-TVA-2739-03 does not clearly show the extent to which the licensee credited truncation due to robust barriers. Using the response to question 1 above, please show the extent to which truncation is credited.

3. Steam line breaks in the debris generation calculation are ruled out because recirculation is not required for cooling the core following a steam line break. However, recirculation using spray flow for environmental qualification of equipment is required. Please explain why this scenario was not analyzed.

Debris Generation

1. Please provide the complete walk-down report, “Report on Watts Bar Unit 1 Containment Building Walkdowns for Emergency Sump Strainer Issues,” TVAW001-RPT-001, Revision 0.

Chemical Effects

1. Please provide the amounts of various Watts Bar containment materials (I) submerged and (ii) in the containment spray zone for the following materials: aluminum, zinc (from

galvanized steel and inorganic zinc (IOZ) coatings), copper, carbon steel, and uncoated concrete. These amounts should include any scaffolding material or metallic-based paints (e.g., aluminum-based paints used on pressure vessels).

2. Provide a discussion concerning the post loss-of-coolant accident (LOCA) containment pool pH, including the range of pH values possible. The values discussed by the licensee at the audit meeting were more refined than the licensee's response to the NRC Generic Letter (GL) 2004-02. Please clarify.

3. If possible, provide the containment pool temperatures as a function of time during the emergency core cooling system (ECCS) mission time for the limiting combination of conditions that would produce (i) the highest pool temperatures with time, and (ii) the lowest pool temperatures with time.

4. Provide the Watts Bar plant-specific chemical effects analysis. Indicate if any more chemical effects related testing is planned.

5. During the integrated chemical effects testing (ICET), in certain chemical environments such as sodium tetraborate, precipitates formed as the solution cooled from the 140ºF test temperature. These products could interact with other downstream debris to cause clogging in narrow passages of downstream components such as valves and pump internals, or affect internal surfaces of heat exchangers or the reactor vessel. Describe your evaluation of potential downstream effects related to interaction with chemical products and the criteria used to determine that performance of downstream components is acceptable for your plant-specific chemical products and debris combination.

6. If all the coatings are assumed to fail, justify why this large additional debris loading would not increase the analyzed amount of chemical effects, or add another unanalyzed chemical product.

Net Positive Suction Head / Loss-of-Coolant Accident

1. Section 2.3 of ALION-REP-TVA-2739-02, Revision 0, notes that the maximum containment sump temperature used to establish the available net positive suction head (NPSH) for the containment spray pumps during the recirculation phase was 190ºF. Please provide the temperature used to establish the available NPSH for the residual heat removal (RHR) pumps during the recirculation phase, and justify if it is different from that used for the spray pumps during recirculation.

2. Please summarize the methodology and assumptions used to determine the maximum sump pool water temperature at the initiation of sump recirculation. Please justify if there is a deviation of this temperature from the calculated maximum containment temperature following a LOCA. If such calculation assumptions were used to maximize containment pressure, please explain the effect of such assumptions on containment temperature.

3. Please provide copies of the following calculation reports referenced in Section 2.5 of ALION-REP-TVA-2739-02, Revision 0:

   - N2-72-4001, R-15 - Containment Spray System

   - N3-74-4001, R12 - RHR System

   - Watts Bar calculation EPM-RCP-120291 Revision 2, Containment Spray Pump Net Positive Head (NPSH) Calculation.

   - Westinghouse calculation FSDA-C-597 dated 11/6/94 - RHR Pump NPSH.

Debris Transport

1. Please provide ALION's FLOW-3D Version 9 executable and the corresponding input deck for the Watts Bar analysis.

Downstream Effects (Core)

These questions refer to the Watts Bar downstream effects calculations found in calculation CN-CSA-05-36, Fuel Evaluation:

1. Page 5 states that a fiber bed of less than 0.125 inch at the core inlet is acceptable. Page 40 states that a 7-foot head loss is predicted for a 1/8-inch fiber bed. What head loss would be produced at the core inlet following a large cold leg break? Please explain and justify whether adequate flow to the core would be provided with this head loss.

2. Page 7 states that 95 percent of fibrous material would be trapped in the bottom fuel nozzle and that the remaining 5 percent is assumed to be returned to the sump. This assumption is stated to be based on the similarity of the dimensions of the flow path through the sump screen and the dimensions through the screen at the bottom of the fuel.

   a. Please provide drawings of the fuel element inlet screens showing the dimensions of the flow path into the fuel.

   b. Provide comparisons of the dimensions of the sump screen holes to the debris screen at the inlet at the fuel elements.

3. Page 10 lists the volume concentration for 3M fiberglass passing through the sump screens as 2.351e-3 and the total fibrous concentration to be 2.559e-3. Page 5 of calculation CN-CSA-05-14 lists the fibrous concentration passing through the sump screens as 5 parts per million. Please relate these quantities.

4. Page 10 states that decay heat is based on American Nuclear Society (ANS) Standards 79 with 2σ. Since this is a LOCA calculation, please explain why the decay heat was

not calculated using ANS Standard 71 + 20 percent to be consistent with Appendix K to *Title 10 Code of Federal Regulations* Part 50.

5. Page 17 shows that following a hot leg break, the fiber bed at the core inlet will exceed the 1/8-inch acceptance criterion within the first hour of recirculation. Please explain the effect of this condition on the core. Describe alternate flow paths for water to reach the core. Describe the transport and deposition of debris through these alternate flow paths.

6. The staff plans to perform audit calculations using the TRACE code to evaluate flow of water to the core through alternate flow paths in the event that the core inlet becomes blocked. Please provide the staff with the location and dimensions of any alternate flow paths through which water could reach the core under these circumstances. Provide the height of flow holes above the bottom of the core as well as their radial distribution about the core periphery.

7. Pages 18 and 19 show the depletion of fibrous material in the recirculating water for hot and cold leg breaks. A range of 97 percent to 95 percent depletion on the sump screens and a range of 95 percent to 50 percent depletion on the fuel screens is assumed. The depletion fraction is assumed to remain constant with time for each cycle as the recirculating water passes the screens. Please explain whether a fiber so short or a particle so small that it can pass through the sump screen and the fuel inlet screens once, will also pass through the sump screens and fuel inlet screens for sequent recirculation passes. Please justify your assumptions.

8. Pages 36 and 37 state that the fuel assembly support grids typically have flow dimensions of 0.04 to 0.115 inches. How do these dimensions compare with those of the Watts Bar fuel? Page 37 further states that the support grids may cause a fiber bed to form across a given elevation to resemble a bed forming across a flat plate. Please explain how the trapping of debris within the support grids and the resulting effect on core heat transfer has been evaluated for Watts Bar. In particular, consider the possibility that a layer of debris and steam forms between a fuel rod and the adjacent support grid so as to prevent water from contacting the fuel rod surface within the support grid. Please explain whether excessive local temperatures would be encountered in this scenario.

9. Pages 43 through 47 evaluate the potential of particulate material such as reflective metal fragments, concrete, latent containment debris and paint chips to flow into the core. It is generally concluded that this material will not reach the core, but will settle out in the lower plenum of the reactor vessel. Please provide an evaluation of the potential to clog the core inlet due to filling the lower reactor vessel with a volume of debris.

10. Page 43 refers to recent internal studies using disk-like particulates of various shapes with a specific gravity of 1.6. These studies were reported to have shown that particulates having a characteristic length of about 70 mils and thickness of 5 mils or greater would settle out in a reactor vessel lower plenum. Please provide documentation for this study describing the test apparatus and procedures. What vertical velocities were used?

11. Page 47 states that coating debris no larger than 0.02 inch are expected to be transported through the fuel. Although this statement may be true for hot leg breaks, it would not be true for large cold leg breaks where the boiling process would cause this material to congregate in the core. Please provide the results of an evaluation of the effect of paint debris on core boiling heat transfer, including the effect of reaction products from the mix of chemicals which would be concentrated in the core by the boiling process following a cold leg break. The effect of the high-radiation field within the core on the chemical and physical nature of the mixture within the core needs to be considered. The potential for heat transfer loss from a chemical film that might form or be plated out by the boiling process needs to be evaluated. Please justify that adequate heat transfer will be maintained during the long-term cooling period.

12. Please provide an evaluation of the concentration of various materials that would occur following a large cold leg break under the conditions that water enters the bottom of the core and is boiled leaving all dissolved and suspended material behind. Consider that hot leg injection begins at 3 hours after the accident. Consider all the constituents within the ECCS water including boric acid, containment spray buffering agents, paint and fibrous debris.

    a. Provide graphs showing the concentration of each constituent as a function of time.

    b. Concentration of material within the reactor core will depend on the water volume that is assumed to be available for mixing. Since the core will be boiling at low pressure it will be in a highly voided condition as will the upper plenum. Please provide and justify the values used for core void fraction and upper plenum void fraction used in the concentration analysis. Provide justification for the fraction of the lower plenum volume, which is included, as well as for any other contribution to the total mixing volume.

    c. Provide the flow rates into the reactor system as a function of time during cold leg recirculation and during hot leg recirculation.

    d. Provide and justify the concentrations flowing into the reactor core as a function of time for each constituent in the ECCS water for both cold leg and hot leg recirculation. Consider boric acid, containment spray buffering solution, paint debris, and fibrous debris.

13. Following the initiation of hot leg recirculation, material which passes through the sump screen will be available to flow to the reactor core from the top. Please provide a comparison of flow restrictions at the top of the core including the fuel elements to that of the sump screens.

Head Loss Testing

1. Please provide the Sequoyah head loss test report that may provide validation that the paint chips would not have transported in the Watts Bar tests had the flow velocities been more prototypical.

2. Please provide the paint chip specification parameters used in the cell floor drain analyses, specifically the floor tumbling velocity and the settling velocity for the turbulence model.

3. Please provide an evaluation of the 3M fiber glass insulation to justify why other fiber surrogate material can be used to represent the 3M fiber glass in the head loss test.

<u>Downstream Effects (Component)</u>

1. Please provide the downstream component hardware change plan, design and completion report.

2. Chemical Considerations

   a). During the ICET, in certain chemical environments such as sodium tetraborate, precipitates formed as the solution cooled from the 140°F test temperature. These products could interact with other downstream debris to cause clogging in narrow passages of downstream components such as valves and pump internals, or affect internal surfaces of heat exchangers or the reactor vessel. Describe your evaluation of potential downstream effects related to interaction with chemical products and the criteria used to determine that performance of downstream components is acceptable for your plant-specific chemical products and debris combination.

   b). Explain how the interaction of downstream chemical effects combined with debris will be evaluated.

3. Throttle Valves

   a). The TVA response to NRC GL 2004-02 dated September 1, 2005, indicated that an updated evaluation will be performed following final selection of strainer design and that the conclusions will be provided in a supplemental response. Describe the approach, including testing program, and schedule to finalize throttle valve positions/openings.

   b). Explain how NRC Information Notice 96-27, and the recent NRC Throttle Valve Testing (NUREG/CR-6902), when available, will be considered in the throttle valve evaluation.

4. Methodology

   a). The TVA response to GL 2004-02 dated September 1, 2005, indicated that the evaluation of downstream effects is consistent with the Westinghouse Commercial Atomic Power (WCAP) Report, WCAP-16406-P, and during the audit the licensee confirmed that they are not taking any exceptions to the WCAP-16406-P methodology. The NRC staff has outstanding questions (NRC letter dated October 27, 2005) on the WCAP-16406-P methodology, and has recently been requested by the Westinghouse Owners Group to formally review WCAP-16406-P as a topical report. Explain how you plan to address comments that result in a revision or addendum to the methodology for topics such as:

- Validation of potential non-conservative assumptions,
- Conservatism to account for uncertainties,
- Wear rates correlated to testing data,
- Debris adhesion to solid surfaces, and
- Downstream matting effect.

Sump Structure

1. Please provide the strainer final design and structure analyses report. If it is not available now, please indicate when it will be available.