Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Watts Bar Nuclear Plant, Unit 1
Facility Operating License NPF-90
NRC Docket No. 50-390

Subject: **Watts Bar Nuclear Plant Unit 1 - Resolution of Generic Safety Issue 191**

References:

1. SECY-12-0093, "Closure Options for Generic Safety Issue-191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," dated July 9, 2012
2. NRC Staff Requirements - SECY-12-0093, "Closure Options for Generic Safety Issue-191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," dated December 14, 2012
3. TVA letter to NRC, "Watts Bar Nuclear Plant, Unit 1 - Path Forward for Resolution of Generic Safety Issue (GSI)-191," dated May 16, 2013 [ML13142A199]
4. NRC letter to PWR Owners Group, "Final Safety Evaluation for Pressurized Water Reactor Owners Group Topical Report WCAP-16793-NP, Revision 2, 'Evaluation of Long-Term Cooling Considering Particulate Fibrous and Chemical Debris in the Recirculating Fluid' (TAC No. ME1234)," dated April 8, 2013
5. TVA Letter to NRC, "Watts Bar Nuclear Plant (WBN) Unit 1 -Supplemental Response to Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors (PWR) - Notice of Completion (TAC No. MC4730)," dated March 31, 2008 [ML081090500]

6. NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 - Request for Additional Information Regarding Generic Letter 2004-02, 'Potential Impact of Debris Blockage During Design-Basis Accidents at Pressurized-Water Reactors' (TAC No. MC4730)," dated September 29, 2009 [ML092650260]
7. Public Meeting Between the NRC and TVA on May 12, 2011 [ML110700031]
8. TVA letter to NRC, "Responses to Requests for Additional Information Related to NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage During Design Basis Accidents at Pressurized-Water Reactors," dated August 15, 2011 [ML11229A783]
9. NRC letter to NEI, "Revised Content Guide for Generic Letter 2004-02 Supplemental Responses," dated November 21, 2007 [ML073110269]

The purpose of this letter is to provide the Nuclear Regulatory Commission (NRC) with the approach that Tennessee Valley Authority (TVA) intends to pursue in order to resolve Generic Safety Issue (GSI)-191. There are three key aspects to this resolution that are addressed in this letter. First, TVA is advising the NRC of a change in a regulatory commitment for addressing in-vessel effects related to fibrous material inside containment. Second, the letter provides information to demonstrate that, following implementation of all modifications, Watts Bar Nuclear Plant (WBN) Unit 1 will meet the requirements for a low fiber plant. Finally, an updated response is provided for two specific items related to sump strainer structural analysis. The details of each of these three aspects are described below.

In Reference 1, the NRC developed options for closure of GSI-191. In Staff Requirements Memorandum (SRM) for SECY-12-0093 (Reference 2), the Commission approved the staff's recommendation to allow licensees the flexibility to choose any of the three options discussed in Reference 1 to resolve GSI-191. TVA submitted its approach to resolving GSI-191 for WBN Unit 1 in Reference 3, proposing to use the low fiber approach (Option 1) described in SECY-12-0093. In TVA's letter, a commitment was made to complete the resolution of the downstream in-vessel effects within two refueling outages after May 2013, which would be the Fall 2015 refueling outage for WBN Unit 1.

Enclosure 1 provides the basis for revising the existing regulatory commitment associated with meeting the requirements for a low fiber plant with respect to in-vessel effects. As described in Enclosure 1, additional time is required to complete all the modifications.

To fully address in-vessel effects, 3M Radiant Energy Shield (RES) material will be removed from inside the cranewall in lower containment resulting in a latent fiber load of less than 15 grams/fuel assembly. TVA proposes to complete the modifications on WBN Unit 1 by the conclusion of the Fall 2018 refueling outage. Work will be conducted during both the Fall 2015 and Spring 2017 refueling outages as well. As a parallel path, TVA plans to request approval of Appendix R deviations for two potential cable interaction scenarios inside lower containment. These Appendix R deviations would reduce the cost and dose impact of removing some 3M RES material, and are consistent with similar deviations approved for WBN Unit 2. Approval of these Appendix R deviations would eliminate the need for the final modifications in the Fall 2018 refueling outage. This would permit earlier resolution of GSI-191 during the Spring 2017 refueling outage.

Enclosure 2 to this letter provides information to demonstrate that, following implementation of all the modifications, WBN Unit 1 will meet the requirements for a low fiber plant. Enclosure 2 addresses the fourteen limitations identified in the NRC's Safety Evaluation (SE) of WCAP-16793-NP, Revision 2 (Reference 4).

By letter dated March 31, 2008 (Reference 5), TVA provided a supplemental response to Generic Letter (GL) 2004-02, consistent with the guidance provided in Reference 9. Enclosure 1 of Reference 5 provided supplemental information to address items 3.k.1 and 3.k.2 related to sump structural analysis. The supplemental information provided for items 3.k.1 and 3.k.2 was based, in part, on a head loss across the strainer of 3.5 feet of water.

By letter dated September 29, 2009 (Reference 6), the NRC issued a request for additional information to TVA to facilitate the completion of their review of GL 2004-02 responses for WBN Unit 1. During a public meeting between the NRC and TVA staff held on May 12, 2011 (Reference 7), TVA discussed the results of strainer testing and additional analysis work related to the closure of GSI-191. The discussions included descriptions of the strainer structural evaluation using a head loss of 6.0 feet of water. A summary of the discussions from the public meeting, including the structural considerations, was included with TVA's response to the Reference 6 request (see Reference 8, Attachment 4).

Enclosure 3 to this letter provides an updated response to items 3.k.1 and 3.k.2 previously provided in Reference 5, Enclosure 1, based on a head loss across the strainer of 6.0 feet of water, as agreed to in a telephone call between the NRC and TVA staff on March 26, 2015.

New regulatory commitments associated with this letter are provided in Enclosure 4.

If you have any questions or comments, please contact Gordon Arent at (423) 365-2004.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 17th day of April 2015.

Respectfully,

Eugene W. Cobey FOR

J. W. Shea
Vice President, Nuclear Licensing

Enclosures:

1. Watts Bar Nuclear Plant Basis for Commitment Change
2. Summary of Watts Bar Unit 1 LOCADM Results and Review of Limitations and Conditions to WCAP-16793-NP-A, Revision 2
3. Watts Bar Nuclear Plant Unit 1 Strainer Structural Evaluation Update
4. List of Commitments

cc: See Page 4

cc (Enclosures):

U. S. Nuclear Regulatory Commission, Region II
NRC Senior Resident Inspector - Watts Bar Nuclear Plant Unit 1
NRR Project Manager - Watts Bar Nuclear Plant

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT
BASIS FOR COMMITMENT CHANGE

Discussion

The Watts Bar Nuclear Plant (WBN) Unit 1 Fire Protection Report (FPR) currently credits 3M Radiant Energy Shield (RES) material in lieu of physical cable separation inside the WBN Unit 1 reactor building, as allowed by 10 CFR 50 Appendix R, Section III.G.2.f. The 3M RES material, being fibrous, poses some concern for in-vessel effects. Tennessee Valley Authority (TVA) has evaluated the feasibility of removing each credited RES application within primary containment. The results of the evaluations have identified cables that would need to be relocated.

The original commitment to address downstream in-vessel effects was within two outages from May 16, 2013, or the Fall 2015 outage. Following making this commitment in May 2013, additional testing of 3M material was unsuccessful in addressing in-vessel effects. Protection of 3M material and alteration of the sump design to capture 3M material were also determined to have significant implementation issues and were not pursued. As a result of these parallel investigations, an engineered design for removal of 3M material was not completed prior to the Spring 2014 refueling outage. The design to remove 3M RES is now sufficiently complete, and the modifications required are extensive.

Implementing the full scope of cable reroutes within a single refueling outage is not practicable. Because of the limited space inside containment, installation requires extensive field routing, and specific mounting details have to be developed as the raceway is installed. The limited space also impacts the number of craft individuals who can work in an area to support installation. For this reason, this effort is most appropriately performed over multiple outages. As a parallel path, TVA plans to request approval of Appendix R deviations for two potential cable interaction scenarios inside lower containment. These Appendix R deviations would reduce the cost and dose impact of removing some 3M RES material. A detailed implementation plan is described below.

Unit 1 Containment RES Removal Summary - Fall 2015 Outage

Cables (1PM777E and 1PM783E) supporting the Loop 3 Hot and Cold leg temperature elements (1-TE-68-43-E and 1-TE-68-60-E) are routed in conduit (1PM8026E) in close proximity to conduit containing cables for the Loop 1 Hot and Cold leg temperature elements inside the polar cranewall in fire zone RI. In order to remove the RES from conduit 1PM8026E serving the Loop 3 instruments, the cables and conduit for these instruments will need to be rerouted outside and around the polar cranewall and wrapped with RES in Fire Zones RF1 (Reactor Building Fan Room 1 [lower containment outside cranewall]) and RA1 (Reactor Building Accumulator Room 1 [lower containment outside cranewall]), which are outside the zone of influence of a postulated pipe break (see table below).

Cable	Raceway	Equipment	Estimated Linear Ft Raceway RES to be Removed	Estimated Linear Feet of New Raceway to Install
1PM777E 1PM783E	1PM8026E	1-TE-68-43-E 1-TE-68-60-E	90	212

In addition, during the Fall 2015 outage, detailed measurements will be taken to support submittal of Appendix R deviations for the following cables/equipment.

Cable	Equipment	Noun Name
1PM590D 1PM594D	1-TE-68-1-D 1-TE-68-18-D	Loop 1 Hot Leg Temperature Loop 1 Cold Leg Temperature
1V5663B	1-FSV-68-395-B 1-FSV-68-396-B	Reactor Pressure Vessel (RPV) Head Vent Valves

Unit 1 Containment RES Removal Summary - Spring 2017 Outage

The cable (1V1207A) providing power to the pressurizer power operated relief valve (PORV) 1-PCV-68-340A-A is routed with cable (1V4426A) for 1-FCV-62-69-A, the Reactor Coolant Loop 3 Letdown Flow Control Valve, in conduit 1VC4057A. The conduit and cables are protected with RES, because the cable serving 1-FCV-62-69-A contains energized conductors that, if shorted to conductors of the cable for PORV 1-PCV-68-340A-A, could spuriously open PORV 1-PCV-68-340A-A. In order to remove the RES from conduit 1VC4057A, the cable for 1-PCV-68-340A-A will be removed from conduit 1VC4057A and routed in a new dedicated conduit to penetration 8 (see table below).

The cable (1V5492B) supporting 1-FSV-68-395-B and 1-FSV-68-396-B (Reactor Pressure Vessel Head Vent Valves) is routed in conduits (1VC4431B and 1VC4432B) that are too close to the pressurizer PORVs located on top of the pressurizer. In order to remove the RES from conduit 1VC4431B and 1VC4432B, cable 1V5492B will be removed and relocated in a dedicated conduit outside and around the polar cranewall. The relocated cable will maintain at least 20 feet from the pressurizer enclosure inside the polar cranewall (see table below).

Cable	Raceway	Equipment	Estimated Linear Ft Raceway RES to be Removed	Estimated Linear Feet of New Raceway to Install
1V1207A	1VC4057A	1-PCV-68-340A-A	20	240
1V5492B	1VC4431B, 1VC4432B	1-FSV-68-395-B 1-FSV-68-396-B	65, 10	140
Total			95	380

Unit 1 Containment RES Removal Summary (Contingency) - Fall 2018 Outage

Cables (1PM590D and 1PM594D) for Loop 1 Hot and Cold leg temperature elements (1-TE-8-1-D and 1-TE-68-18-D) are routed in the same conduit (1PM8020D, 1PM8021D, and 1PM8022D) as cables (1PM685D and 1PM680D) for the Loop 2 Hot and Cold leg temperature elements inside polar cranewall fire zone RI. In order to remove the RES from conduit 1PM8020D, 1PM8021D, and 1PM8022D; cables (1PM590D and 1PM594D) for the Loop 1 Hot and Cold leg temperature elements will need to be relocated outside and around the polar cranewall and wrapped with RES in fire zone RF2 (Reactor Building Fan Room 2 [lower containment outside cranewall]) and RA2 (Reactor Building Accumulator Room 2 [lower containment outside cranewall]). See the table below.

Cable (1V5663B) for RPV Head Vent Valves 1-FSV-68-395-B and 1-FSV-68-396-B is routed in conduits (1VC4062B, 1VC4063B, and 1VC4064B) that are too close to the pressurizer PORVs. In order to remove the RES from these conduits (1VC4062B,

1VC4063B, and 1VC4064B), the cable (1V5663B) for the RPV head vent valves will need to be relocated to achieve 20 feet of separation from the pressurizer enclosure (see table below).

Cable	Raceway	Equipment	Estimated Linear Ft Raceway RES to be Removed	Estimated Linear Feet of New Raceway to Install
1PM590D 1PM594D	1PM8020D, 1PM8021D, 1PM8022D	1-TE-68-1-D 1-TE-68-18-D	25, 15, 30	156
1V5663B	1VC4062B, 1VC4063B, 1VC4064B	1-FSV-68-395-B 1-FSV-68-396-B	5, 5, 15	160
Total			95	316

Appendix R Deviation Requests

As a parallel path, TVA plans to request approval of Appendix R deviations. These Appendix R deviation requests will be submitted after critical measurements are made during the Fall 2015 refueling outage. The deviations will request relief from the 20 foot horizontal separation requirement of 10 CFR 50 Appendix R Section III.G.2.d based upon an analysis of the fire hazards inside WBN Unit 1 primary containment. The dual unit Fire Protection Report Part VII, Section 2.9.20 contains similar deviation requests, which were reviewed by the NRC Staff and documented in NUREG-0847 Supplement 26, Section 6.1.10.4. Approval of these Appendix R deviations would eliminate the need for the final modifications in the Fall 2018 refueling outage. This would permit earlier resolution of GSI-191 during the Spring 2017 refueling outage.

Summary

The full scope of effort to remove all remaining 3M RES material from inside the cranewall in lower containment has been identified and the path to complete is well understood. Following implementation of these modifications, WBN Unit 1 will fully meet the requirements for a low fiber plant.

ENCLOSURE 2

Summary of Watts Bar Unit 1 LOCADM Results and Review of Limitations and Conditions to WCAP-16793-NP-A, Revision 2

The following provides a response to each of the limitations and conditions with respect to the application of WCAP-16793-NP-A, Revision 2 which will apply to Watts Bar Nuclear Plant (WBN) Unit 1 following completion of all modifications to remove 3M Radiant Energy Shield (RES) material from inside the cranewall of lower containment.

1. *Licensees should confirm that their plants are covered by the PWROG sponsored fuel assembly tests by confirming that the plant available hot-leg break driving head is equal to or greater than that determined as limiting in the proprietary fuel assembly tests and that flow rate is bounded by the testing. Licensees should validate that the fuel types and inlet filters in use at the plant are covered by the test program (with the exception of LTAs). Licensees should limit the amount of fibrous debris reaching the fuel inlet to that stated in Section 10 of the WCAP (15 grams per fuel assembly for a hot-leg break scenario).*

   Alternately, licensees may perform plant specific testing and/or evaluations to increase the debris limits on a site-specific basis. The available driving head should be calculated based on the core exit void fraction and loop flow resistance values contained in their plant design basis calculations, considering clean loop flow resistance and a range of break locations. Calculations of available driving head should account for the potential for voiding in the steam generator tubes. These tests shall evaluate the effects of increased fiber on flow to the core, and precipitation of boron during a postulated cold-leg break, and the effect of p/f ratios below 1:1. The NRC staff will review plant specific evaluations, including hot- and cold-leg break scenarios, to ensure that acceptable justification for higher debris limits is provided. (Sections 3.1.2 (c), 3.1.2 (e), 3.3.1, 3.4.2, 3.8, 3.9 and 3.10 of this SE)

   TVA Response

   Tennessee Valley Authority (TVA) confirms that WBN Unit 1 is covered by the Pressurized Water Reactor Owners Group (PWROG) sponsored fuel assembly tests. WCAP-16793-NP-A, Revision 2, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," states that the maximum delta pressure (dP) due to the presence of 15 grams of fiber is small and **all plants** (emphasis added) have a driving head greater than this value, so core flow will not be impeded. This statement was based on the AREVA testing conducted in support of WCAP-17057-P, Revision 1, "GSI-191 Fuel Assembly Test Report for PWROG," that demonstrated that 15 grams of fiber per fuel assembly will not cause a blockage that can challenge long-term core cooling.

   WBN Unit 1 fuel is Westinghouse Robust Fuel (RFA-2) with the Debris Filter Bottom Nozzle (DFBN). Testing established that the RFA-2 fuel is bounded by the limiting fuel used to establish the allowable limits presented in WCAP-16793.

The WBN Unit 1 fuel fiber content is 14.8 grams per fuel assembly. This is calculated to be the worst possible debris case for the design basis. WBN Unit 1 will keep an inventory of fibrous debris in containment to ensure that both the known and latent fibrous debris remains below the design basis limit. Because the WBN Unit 1 fuel fiber content is less than the 15 grams per fuel assembly limit, WBN Unit 1 is bounded by the evaluations performed in the aforementioned WCAPs for low fiber plants and therefore, has adequate hot leg driving head to assure core cooling.

2. *Each licensee's GL 2004-02 submittal to the NRC should state the available driving head used in the evaluation of the hot-leg break scenario, the ECCS flow rates, and the results of the LOCADM calculations. Licensees should provide the type(s) of fuel and inlet filters installed in their plants, as well as the amount of fiber (gram per fuel assembly) that reaches the core. (Section 3.3.1 and 3.10 of this SE)*

TVA Response

The available driving head for the hot leg break scenarios was developed generically, and the results were provided in WCAP-16793-NP-A, Revision 2. The WCAP evaluated a minimum available driving head that was acceptable for the fleet of Pressurized Water Reactors.

The Loss of Coolant Accident Deposition Model (LOCADM) evaluation for WBN Unit 1 used plant-specific conditions and materials to determine a maximum cladding temperature and deposit thickness. The fuel cladding temperature and deposit thickness determined from the evaluation were compared to the conservative maximum deposition thickness of 50 mils (1,270 microns) and maximum acceptable temperature of 800°F as indicated in WCAP-16793-NP-A, Revision 2. The final calculated deposition thickness for WBN Unit 1 is 17.01 mils (432 microns), which is less than the recommended upper limit of 50 mils. The calculated maximum temperature of the fuel cladding during recirculation from the containment sump over the 30 days following the Loss-of-Coolant Accident (LOCA) is calculated to be 337.3°F, which is less than the recommended maximum cladding temperature of 800°F.

WBN Unit 1 fuel is Westinghouse RFA-2 with the Debris Filter Bottom Nozzle.

The low-fiber plant criterion from "Transmittal of GSI-191 Resolution Criteria for 'Low Fiber' Plants," Nuclear Energy Institute (NEI), dated December 22, 2011, has been used to establish that WBN Unit 1 is a low-fiber plant as shown below.

The WBN Unit 1 fuel fiber content is calculated assuming 14 pounds of fiber, 100% debris transport (which is more conservative than the NEI criteria), 45% strainer bypass and 193 fuel assemblies:

14 lb*1.0*.45*453.6 gm/lb/193 = 14.8 gm/assembly

The calculation uses NEI generic values adjusted for the plant-specific number of fuel assemblies as suggested by the NEI criteria. WBN Unit 1 has a fuel fiber content of 14.8 grams per fuel assembly. WBN Unit 1 will implement controls to ensure that latent fibrous debris is limited to no more than 14 pounds, as assumed in the calculation.

3. *Section 3.1.4.3 of the WCAP states that alternate flow paths in the RPV were not credited. The section also states that plants may be able to credit alternate flow paths for demonstrating adequate LTCC. If a licensee chooses to take credit for alternate flow paths, such as core baffle plate holes, to justify greater than 15 grams of bypassed fiber per fuel assembly, the licensee should demonstrate, by testing or analysis, that the flow paths would be effective, that the flow holes will not become blocked with debris during a LOCA, that boron precipitation is considered, and that debris will not deposit in other locations after passing through the alternate flow path such that LTCC would be jeopardized. (Sections 3.3.1 and 3.4.2 of this SE)*

   TVA Response

   Alternate flow paths were not credited in determining the acceptability of the WBN Unit 1 Emergency Core Cooling System (ECCS) design, as WBN Unit 1 does not have a need to justify a fiber load of > 15 grams fiber/fuel assembly.

4. *Sections 3.2 and 3.3 of the WCAP provide evaluations to show that even with large blockages at the core inlet, adequate flow will enter the core to maintain LTCC. The staff recognizes that these calculations show that significant head loss can occur while maintaining adequate flow. However, the analyses have not been correlated with debris amounts. Therefore, the analyses cannot be relied upon to demonstrate adequate LTCC. (Sections 3.3.3 and 3.4 of this SE)*

   TVA Response

   WBN Unit 1 core inlet head loss is bounded by the generic evaluations and test results for low fiber plants.

5. *In RAI Response number 18 in Reference 13, the PWROG states that numerical analyses demonstrated that, even if a large blockage occurs, decay heat removal will continue. The NRC staff's position is that a plant must maintain its debris load within the limits defined by the testing (e.g., 15 grams per assembly). Any debris amounts greater than those justified by generic testing in this WCAP must be justified on a plant-specific basis. (Sections 3.4.2 and 3.10 of this SE)*

   TVA Response

   The WBN Unit 1 fuel fiber content of 14.8 gm/assembly is less than the 15 gm/assembly target value. Thus, the WBN Unit 1 debris amounts are justified by generic testing in WCAP-16793-NP-A, Revision 2, consistent with the NRC's position. Because fiber loading is less than 15 gm/assembly, there is no need to justify debris amounts higher than assumed in the Westinghouse WCAP generic evaluations.

6. *The fibrous debris acceptance criteria contained in the WCAP may be applied to fuel designs evaluated in the WCAP. Because new or evolving fuel designs may have different inlet fittings or grid straps that could exhibit different debris capture characteristics, licensees should evaluate fuel design changes in accordance with 10 CFR 50.59 to ensure that new designs do not impact adequate long term core cooling following a LOCA. (Section 3.4.2 of this SE)*

TVA Response

WBN Unit 1 fuel is Westinghouse RFA-2 with the Debris Filter Bottom Nozzle. Testing and evaluations as required by this condition established that the WBN Unit 1 RFA-2 fuel is bounded by the limiting fuel configuration tested to establish the allowable limits presented in WCAP-16793-NP-A, Revision 2.

7. *Sections 2 and 4.3 of the WCAP establish 800 degrees Fahrenheit as the acceptance limit for fuel cladding temperature after the core has been re-flooded. The NRC staff accepts a cladding temperature limit of 800 degrees Fahrenheit as the long-term cooling acceptance basis for GSI-191 considerations. Each licensee's GL 2004-02 submittal to the NRC should state the peak cladding temperature predicted by the LOCADM analysis. If a licensee calculates a temperature that exceeds 800 degrees Fahrenheit, the licensee must submit data to justify the acceptability of the higher clad temperature. (Sections 3.2, 3.4.3, 3.4.4, and 3.10 of this SE)*

TVA Response

The peak cladding temperature calculated by LOCADM for WBN Unit 1 is 337.3°F, which is less than the acceptance criterion of 800°F.

8. *As described in the Limitations and Conditions for WCAP-16530-NP (ADAMS Accession No. ML073520891) (Reference 21), the aluminum release rate equation used in TR WCAP-16530-NP provides a reasonable fit to the total aluminum release for the 30-day ICET tests but under-predicts the aluminum concentrations during the initial active corrosion portion of the test. Actual corrosion of aluminum coupons during the ICET 1 test, which used sodium hydroxide (NaOH), appeared to occur in two stages; active corrosion for the first half of the test followed by passivation of the aluminum during the second half of the test. Therefore, while the 30-day fit to the ICET data is reasonable, the WCAP-16530- NP-A model under-predicts aluminum release by about a factor of two during the active corrosion phase of ICET 1. This is important since the incore LOCADM chemical deposition rates can be much greater during the initial period following a LOCA, if local conditions predict boiling. As stated in WCAP-16530-NP-A, to account for potentially greater amounts of aluminum during the initial days following a LOCA, a licensee's LOCADM input should apply a factor of 2 increase to the WCAP-16530-NP-A spreadsheet predicted aluminum release, not to exceed the total amount of aluminum predicted by the WCAP-16530-NP-A spreadsheet for 30 days. In other words, the total amount of aluminum released equals that predicted by the WCAP-16530-NP-A spreadsheet, but the timing of the release is accelerated. Alternately, licensees may choose to use a different method for determining aluminum release but licensees should not use an aluminum release rate equation that, when adjusted to the ICET 1 pH, under-predicts the aluminum concentrations measured during the initial 15 days of ICET 1. (Section 3.7 of this SE)*

TVA Response

The deposit thickness calculated for WBN Unit 1, 17.01 mils, accounts for the underpredicted aluminum release rate during the active corrosion phase by doubling the available aluminum surface area while maintaining the 30-day total aluminum release mass as recommended by WCAP-16793-NP-A, Revision 2.

9. *In the response to NRC staff RAIs, the PWROG indicated that if plant-specific refinements are made to the WCAP LOCADM base model to reduce conservatisms, the user should demonstrate that the results still adequately bound chemical product generation. If a licensee uses plant-specific refinements to the WCAP-16530-NP-A base model that reduces the chemical source term considered in the downstream analysis, the licensee should provide a technical justification that demonstrates that the refined chemical source term adequately bounds chemical product generation. This will provide the basis that the reactor vessel deposition calculations are also bounding. (Section 3.7 of this SE)*

   TVA Response

   No plant-specific refinements were made to the LOCADM base model to reduce conservatism for WBN Unit 1.

10. *The WCAP states that the material with the highest insulating value that could deposit from post-LOCA coolant impurities would be sodium aluminum silicate. The WCAP recommends that a thermal conductivity of 0.11 BTU/(h-ft-°F) be used for the sodium aluminum silicate scale and for bounding calculations when there is uncertainty in the type of scale that may form. If plant-specific calculations use a less conservative thermal conductivity value for scale (i.e., greater than 0.11 BTU/(h-ft-°F)), the licensee should provide a technical justification for the plant-specific thermal conductivity value. This justification should demonstrate why it is not possible to form sodium aluminum silicate or other scales with thermal conductivities less than the selected value. (Section 3.7 of this SE)*

    TVA Response

    No plant-specific refinements were made to the LOCADM base model to reduce conservatism for WBN Unit 1. A value of 0.1 BTU/(hr-ft-°F) was used in the WBN Unit 1 LOCADM deposition evaluation, which is bounded by the WCAP.

11. *Licensees should demonstrate that the quantity of fibrous debris transported to the fuel inlet is less than or equal to the fibrous debris limit specified in the proprietary fuel assembly test reports and approved by this SE. Fiber quantities in excess of 15 grams per fuel assembly must be justified by the licensee. Licensees may determine the quantity of debris that passes through their strainers by (1) performing strainer bypass testing using the plant strainer design, plant-specific debris loads, and plant-specific flow velocities, (2) relying on strainer bypass values developed through strainer bypass testing of the same vendor and same perforation size, prorated to the licensee's plant specific strainer area; approach velocity; debris types, and debris quantities, or (3) assuming that the entire quantity of fiber transported to the sump strainer passes through the sump strainer. The licensee's submittals should include the means used to determine the amount of debris that bypasses the ECCS strainer and the fiber loading expected, per fuel assembly, for the cold-leg and hot-leg break scenarios. Licensees of all operating PWRs should provide the debris loads, calculated on a fuel assembly basis, for both the hot-leg and cold-leg break cases in their GL 2004-02 responses. (Section 3.10 of this SE)*

TVA Response

The WBN Unit 1 fuel fiber content, calculated in accordance with the NEI low-fiber plant criteria, is 14.8 gm/assembly. This value is less than the 15 gm/assembly target value. Thus, no detailed WBN Unit 1 specific strainer performance bypass evaluations were required.

12. *Plants that can qualify a higher fiber load based on the absence of chemical deposits should ensure that tests for their conditions determine limiting head losses using particulate and fiber loads that maximize the head loss with no chemical precipitates included in the tests. (Section 3.3.1 of this SE.) Note that in this case, licensees must also evaluate the other considerations discussed in Item 1 above.*

TVA Response

WBN Unit 1 does not need to qualify a higher fiber load.

13. *Licensees should verify that the size distribution of fibrous debris used in the fuel assembly testing referenced by their plant is representative of the size distribution of fibrous debris expected downstream of the plant's ECCS strainer(s). (Section 3.4.2.1 of this SE)*

TVA Response

The size distribution of fibrous debris for WBN Unit 1 is appropriate. The only fiber available in the sump water is latent fiber. There is no fibrous insulation in the zone of influence. NUKON commercial fiberglass was assumed to be representative of latent fiber per NUREG/CR-6224. NUKON fiber was also used to represent fiber in WCAP-17057-P, Revision 1.

14. *The "Margin Calculator," referenced in References 11 and 12, has not been submitted to the NRC under formal letter, and NRC staff has not performed a detailed review of the document. Therefore, NRC staff expects licensees to base their GL 2004-02 in-vessel effects evaluations on the information provided in the proprietary test reports and associated RAI responses (References 8, 16, 17, 11 and 12), including the conditions and limitations stated in this SE, and existing plant design-basis calculations and analyses.*

TVA Response

WBN Unit 1 did not use the "Margin Calculator" to determine in-vessel effects. The "Margin Calculator" was available as a tool for use by the PWROG to perform a preliminary evaluation of debris effects on fuel. The WBN Unit 1 in-vessel effects were determined based on WCAP-17057-P-A, Revision 1, and the plant-specific LOCADM analysis.

ENCLOSURE 3

WATTS BAR NUCLEAR PLANT UNIT 1
STRAINER STRUCTURAL EVALUATION UPDATE

This enclosure provides an updated response to items 3.k.1 and 3.k.2 that was previously submitted to the NRC in TVA letter dated March 31, 2008 (Reference 1).

3.k.1. Summarize the design inputs, design codes, loads, and load combinations utilized for the sump strainer structural analysis.

TVA Response

The structural evaluations of the WBN sump strainers and flow plenum assembly were performed using a combination of manual calculations and finite element analyses using the GTSTRUDL Computer Program. The evaluations follow requirements imposed by the TVA Design Specification for the containment building sump strainers which are consistent with the plant design and licensing basis requirements. A summary of the design inputs, design codes, loads and load combinations used in the strainer/plenum structural analyses are as follows.

Design Input

The design inputs used in the structural analysis of the WBN sump strainers and plenum assembly consisted of the following.

1) Strainer/plenum arrangement and dimensional data from the appropriate component design and fabrication drawings.
2) Strainer/plenum material types from the appropriate component design and fabrication drawings.
3) Design and maximum operating temperatures from the strainer/plenum design specification.
4) WBN plant specific seismic acceleration response spectra from the strainer/plenum design specification.
5) Structural analysis load type, combinations, and acceptance criteria from the strainer/plenum design specification.

Design Codes

The WBN containment sump strainers and flow plenum assembly were designed, fabricated, and inspected in accordance with the following codes and standards. Unless otherwise stated, the standards were the latest in effect on the date of the purchase order.

1) American Institute of Steel Construction (AISC), "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," 7^{th} Edition, adopted February 12, 1969.
2) ASME Section II, "Material Specifications."
3) ASME Section III, Division 1, Subsection NF, "Supports," 2004 Edition thru July 2005 Addenda.
4) ASME Section V, "Non-Destructive Examination," 2004 Edition thru July 2005 Addenda.

5) ASME Section IX, "Welding and Brazing Qualification," 2004 Edition thru July 2005 Addenda.
6) AWS D1.6 – 1999, "Structural Welding Code – Stainless Steel."

The primary design and fabrication standard for the WBN strainer equipment was the AISC standard cited above. The equipment structural analysis acceptance criteria were primarily established in accordance with this standard. In circumstances where the AISC Code does not provide adequate guidance for a particular component, other codes or standards are used for guidance. These alternate codes are discussed briefly below.

The AISC Code does not provide any design guidelines for perforated plate. Therefore, the equations from Appendix A, Article A-8000 of the ASME B&PV Code, Section III, 1989 Edition, were used to calculate the perforated plate stresses. The acceptance criteria are also based on this code. In addition, the AISC Code does not specifically cover stainless steel materials. Since the strainers are fabricated entirely from stainless steel, the ANSI/AISC N690-1994, "Specification for the Design, Fabrication, and Erection of Steel Safety Related Structures for Nuclear Facilities" was used to supplement the AISC in any areas related specifically to the structural qualification of stainless steel. Only the basic acceptance criteria (allowable stresses) are used from the ASME Code. Load combinations and allowable stress factors for higher service level loads are not used.

The strainer also has several components made from thin gage sheet steel and cold formed stainless sheet steel. For these components, SEI/ASCE 8-02, "Specification for the Design of Cold-Formed Stainless Steel Structural Members" was used where rules specific to thin gage and cold formed stainless steel are applicable. The rules for Allowable Stress Design (ASD) as specified in Appendix D of this code were used. This is further supplemented by the AISI Code where the ASCE Code is lacking specific guidance. Finally, guidance is also taken from AWS D1.6, "Structural Welding Code - Stainless Steel" as it relates to the qualification of stainless steel welds.

Structural Analysis Loads, Load Combinations, and Acceptance Criteria

The structural analysis of the strainers and associated flow plenum considered the following design basis loads.

1) DW - Dead Weight Loads and forces.
2) TOL - Thermal Effect Loads during normal operation (loads imposed by normal operating temperatures, conservatively taken at 140°F).
3) OBE - Seismic Loads generated by the operating basis earthquake.
4) SSE - Seismic Loads generated by the safe shutdown earthquake.
5) TAL - Thermal Effect Loads during accident operation (loads imposed by accident operating temperatures, taken as the maximum water temperature of 190°F).
6) JIL - Jet Impingement equivalent static load (if applicable) (JIL = 0 for WBN).
7) DIL - Debris Impact equivalent static load (if applicable) (DIL = 0 for WBN).
8) DP - Differential pressure across perforated plates and other pressure boundaries.
9) DEB - Debris Weight.

These design basis loads were combined and confirmed to meet the indicated acceptance criteria as follows:

Load Combination 1 -	DW + DP + DEB $\leq$ S	Notes 1,4,5,7
Load Combination 2 -	DW + OBE $\leq$ S	Notes 1,6,7
Load Combination 3 -	DW + TOL + OBE $\leq$ 1.5 x S	Notes 1,6,7
Load Combination 4 -	DW + TOL + SSE $\leq$ 1.6 x S	Notes 1,6,7
Load Combination 5 -	DW + DP + DEB + TAL $\leq$ 1.6 x S	Notes 1,4,5,6,7
Load Combination 6 -	DW + JIL + DIL + SSE $\leq$ 1.6 x S	Notes 1,2,3,6,7

Notes

1) For structural steel, the "S" value is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC specification, Seventh Edition. The 33 percent increase in allowable stresses for steel due to seismic or wind loadings permitted by the AISC standard was not applied to this evaluation. When alternate standards were used to supplement the AISC specification as indicated below, the "S" value was consistent with the AISC definition except that the allowable stresses were taken from the alternate standard.

   For perforated plates, the "S" value was the allowable stress from the ASME Section III B&PV Code, 1989 Edition including Appendix A, Article A-8000 provisions for calculating perforated plate stresses.

   For concrete anchor bolts, the tensile and shear forces shall not exceed the allowable loads for the selected anchor bolts in TVA Design Standard No. DS-C1.7.1 Revision 11. TVA concurrence with anchor bolt selection is required. Thermal stresses on anchor bolts shall be considered and minimized by the design.

2) The AISC allowable load combination for Load Case 6 shall not exceed the following limits:

$0.9 \times F_y$	for Tension or Bending Stress
$(0.9 \times F_y) \div (3.0)^{0.5}$	for Shear Stress
$0.9 \times F_{critical\ buckling}$	for Compression Stress

   where F_y = minimum specified yield strength of the material, and
   $F_{critical\ buckling}$ = the compressive stress calculated by the AISC equations without the appropriate factor of safety

3) The jet impingement load (JIL) and debris impact load (DIL) are negligible for the final strainer design.

4) The differential pressure (DP) shall be considered 6.0 feet of water for Unit 1 or the design basis head loss determined by the AREVA Document 38-9164217-000 and 32-9154651-000.

5) Debris weight (DEB) shall be considered for Loading Combinations 1 and 5. The debris weight on the strainer structure shall be the larger of 25 pounds per square foot applied to the total strainer/flow plenum horizontal footprint area or the maximum calculated debris weight transported to the strainer under design basis operating conditions.

6) It is not necessary to consider hydrostatic or hydrodynamic loads for the load combinations which include OBE and SSE loads.

7) Since stainless steel does not display a single, well defined modulus of elasticity, the allowable compression stress equations from the AISC specification, Seventh Edition shall not be applied to stainless steel materials. For stainless steel materials, the allowable compression stress will be based on the lower allowable from ANSI/AISC N690-1994. The allowable stresses for tension, shear, bending and bearing for stainless steel materials shall be taken from the allowables provided for carbon steel in the AISC specification, Seventh Edition.

3.k.2. Summarize the structural qualification results and design margins for the various components of the sump strainer structural assembly.

TVA Response

The structural analysis of the strainer and flow plenum assemblies established that the assemblies meet the structural acceptance criteria for all applicable loadings. A summary of the limiting stress interaction ratios (i.e., calculated stress divided by allowable stress) is as follows:

WBN Containment Sump Strainer and Flow Plenum Structural Analysis Interaction Ratios

Strainer Component	Maximum Stress Ratio	Flow Plenum Component	Maximum Stress Ratio
Radial Stiffener (w/ Collar)	0.89	Support Beams	0.10
Tension Rods	0.46	Support Floor Beam Local Web	0.95
Edge Channels	0.89	Top Cover Plate	0.71
Cross Bracing	0.40	Lower Deck Plate	0.44
Hex Coupling	0.34	Plate Beam Over Pit	0.44
Core Tube	0.21	Hex Couplings	0.26
Radial Stiffeners (Bent Portion)	0.13	Plenum Box Channels	0.27
Spacer	0.87	Plenum Box Channel Local Web	0.29
Spacer Separation	0.89	Lower Deck Drainage Perforated Plate	0.71
Perforated Plate (DP Case)	0.36	Lower Deck Drainage Plate Openings	0.06
Perforated Plate (Seismic Case)	0.04	Top Strip to Hex Couple Bolts	0.52
Perforated Plate (Inner Gap)	0.25	Channel to Support Beam Bolts	0.36
Inner Gap Buckling	0.33	Channel Local Flange at Bolts	0.72
Wire Stiffener	0.89	Bottom Plates to Beam Bolts	0.28
Perforated Plate (Core Tube End Cover DP Case)	0.47	Channel Splice Plate Bolts	0.37
Radial Stiffening Spokes of the End Cover Stiffener	0.68	Channel to Channel Splice Welds	0.92
End Cover Sleeve	0.23	Channel Splice Plate Shear	0.66
Weld of End Cover Stiffener to End Cover Sleeve	0.19	Channel to Channel Welds at Curb Corner	0.67

Strainer Component	Maximum Stress Ratio	Flow Plenum Component	Maximum Stress Ratio
Weld of Radial Stiffener to Core Tube	0.10	Concrete Expansion Anchors	0.71
Edge Channel Rivets	0.08	Floor Beam Local Flange at Bolts	0.80
Inner Gap Hoop Rivets	0.06	Clip Angle to Sump Curb Weld	0.83
End Cover Rivets	0.00	Tube Steel Posts	0.36
Connecting Bolts	0.33	Strip Plate Local Stress at TS Connection	0.41
Collar	0.95	Lower Deck Plate Local Stress	0.30
N/A	N/A	Plate Beam to Top Plate and Hex Connector Plate	0.85
N/A	N/A	Plate Beam to TS Weld	0.28
N/A	N/A	Plate Beam to TS Bolts	0.44
N/A	N/A	TS to Strip Plate Weld	0.71

Reference

1. TVA letter to NRC, "Watts Bar Nuclear Plant (WBN) Unit 1 - Supplemental Response to Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors (PWR) - Notice of Completion (TAC No. MC4730)," dated March 31, 2008

ENCLOSURE 4

LIST OF COMMITMENTS

1. Watts Bar Nuclear Plant Unit 1 will complete the removal of 3M RES inside the cranewall from conduit 1PM8026E serving 1-TE-68-43-E and 1-TE-68-60-E and reroute these cables during the Fall 2015 outage.
2. Watts Bar Nuclear Plant Unit 1 will submit Appendix R deviation requests for raceway serving 1-TE-68-1-D, 1-TE-68-18-D, 1-FSV-68-395-B, and 1-FSV-68-396-B within 60 days of completion of the Fall 2015 outage.
3. Watts Bar Nuclear Plant Unit 1 will complete the removal of 3M RES inside the cranewall from conduits 1VC4057A, 1VC4431B, and 1VC4432B and reroute the associated cables during the Spring 2017 outage.
4. Watts Bar Nuclear Plant Unit 1 will remove 3M RES inside the cranewall from raceway serving Loop 1 Hot and Cold leg temperature elements and the RPV head vent valves during the Spring 2017 outage, if associated Appendix R deviation requests are approved prior to the commencement of the outage.
5. Watts Bar Nuclear Plant Unit 1 will reroute cables serving Loop 1 Hot and Cold leg temperature elements and the RPV head vent valves to allow removal of RES inside the cranewall during the Fall 2018 outage, if associated Appendix R deviations are not approved.
6. The Watts Bar Nuclear Plant Unit 1 Containment Debris Log will be revised to limit latent fibrous debris to no more than 14 pounds by October 31, 2015.