UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 8, 2012

Mr. Joseph W. Shea
Manager, Corporate Nuclear Licensing
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING THE PROPOPSED TECHNICAL SPECIFICATION CHANGES TO ALLOW USE OF AREVA ADVANCED W17 HIGH THERMAL PERFORMANCE FUEL (TAC NOS. ME6538 AND ME6539)

Dear Mr. Shea:

By letter dated June 17, 2011, and supplemented by letters dated July 27 and November 14, 2011, you submitted an application for license amendment to revise the Sequoyah Nuclear Plant, Units 1 and 2, Technical Specifications to allow the use of AREVA Advanced W17 high thermal performance fuel to address fuel assembly distortion and its resultant fuel handling issues.

The Nuclear Regulatory Commission staff is reviewing the submittal and has determined that additional information is required to complete its evaluation. This request was discussed with Mr. Clyde Mackaman of your staff on January 25, 2012, and it was agreed that a response would be provided within 30 days from the date of this letter.

If you have any questions regarding this matter, I can be reached at 301-415-1564.

Sincerely,

Siva P. Lingam

Siva P. Lingam, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-327 and 50-328

Enclosure: Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

REGARDING TECHNICAL SPECIFICATION CHANGES

TO ALLOW USE OF AREVA ADVANCED W17

HIGH THERMAL PERFORMANCE FUEL

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

By letter dated June 17, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML111720780), and supplemented by letters dated July 27 and November 14, 2011 (ADAMS Accession Nos. ML112101798 and ML113200023, respectively), the Tennessee Valley Authority (the licensee), submitted a license amendment request to revise the Sequoyah Nuclear Plant (SQN), Units 1 and 2, Technical Specifications (TSs) to allow the use of AREVA Advanced W17 high thermal performance (HTP) fuel to address fuel assembly distortion and its resultant fuel handling issues. In order to complete its review of the above documents, the U.S. Nuclear Regulatory Commission (NRC) staff needs the following additional information for the Reactor Systems Branch (SRXB) and the Nuclear Performance and Code Review Branch (SNPB):

SRXB

1. The AREVA Advanced W17 HTP fuel assembly design consists of standard uranium dioxide (UO_2) fuel pellets with gadolinium oxide (Gd_2O_3) burnable poison and M5^{TM} cladding. Please identify those plants used similar fuel and provide description for the discrepancy, if any, in the fuel design from the proposed Advanced W17 HTP fuel.

2. SQN plans to refuel and operate with AREVA Advanced W17 HTP fuel beginning with the cycles following the refueling outage in the fall of 2012 for SQN, Unit 2, and in the fall of 2013 for SQN, Unit 1. Please provide: (a) the core loading pattern with clearly specifying fuel types and quantities including their fresh or resident fuel with once-burned, twice-burned or thrice-burned for SQN, Units 1 and 2; (b) the guidelines or procedures used to generate the final core loading pattern; and (c) a detailed description of the impact on the departure from nucleate boiling ratio (DNBR) limit calculation due to the final selected core loading pattern.

3. The proposed TS changes for the DNBR limits for each fuel type are as follows:

   For the Advanced W17 HTP fuel design
   DNBR ≥ 1.132 for the BHTP correlation

   DNBR ≥ 1.21 for the BWU-N correlation

   For the Mark-BW fuel design
   DNBR ≥ 1.21 for the BWCMV correlation

   DNBR ≥ 1.21 for the BWU-N correlation

Enclosure

Please provide: (a) a detailed description with respect to the application of the DNBR correlations to the Advanced W17 HTP fuel and the Mark-BW fuel design; (b) rationale to apply two DNBR correlations to each fuel type; (c) operating procedures and method of any core monitoring system to assure that these two DNBR limits would not be violated.

4. The proposed Safety Limit 2.1.1.2 states, "The maximum local fuel pin centerline temperature shall be maintained ≤ 4901 °F[degrees Fahrenheit], decreasing by 13.7 °F per 10,000 MWD/MTU [megawatt-days/metric ton of uranium] of burnup for COPERNIC applications, and ≤ 4642 °F, decreasing by 58 °F per 10,000 MWD/MTU of burnup for TACO3 application." Please provide: (a) details of the reason why these two codes are needed to set the fuel pin centerline temperature limit; and (b) the means used to monitor the maximum local fuel pin centerline temperature.

5. In the proposed Figure 3.2-1, Flow versus Power for 4 Loops in Operation, there is a 3.5 percent measurement uncertainty for flow is included. Please provide the basis for a 3.5 percent flow measurement uncertainty and justify that this 3.5 percent is conservative.

6. Provide a flow chart or table to clearly demonstrate that all the approved methodologies listed in the proposed revision to TS 6.9.1.14.a are necessary to support the cycle-specific parameters listed in TS 6.9.1.14.

7. It appears that TACO3 code is applied to calculation for limitation of local fuel pin centerline temperature. Please justify that TACO3 code does not support the parameter listed in TS 6.9.1.14.1.

8. Provide a detailed description with respect to the applicability of all the approved methodologies listed in the proposed TS 6.9.1.14.a to the AREVA Advanced W17 HTP fuel design.

9. Please clarify that the analytical methods used to determine the reactor coolant pressure (RCS) pressure and temperature limits listed in TS 6.9.1.15.a is still applicable to AREVA Advanced W17 HTP fuel design.

10. Please describe in details the application of both proposed TS 6.9.1.14.a.4, "PWR [pressurized water reactor] Small Break LOCA [loss-of-coolant accident] Evaluation Model, March 2001," and TS 6.9.1.14.a.7, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors, April 2003," to the proposed fuel transition analyses.

SNPB

11. The references of Mark-BW fuel design leading to the Advanced W17 HTP fuel design in Sections 1 and 2 of ANP-2986(P), Rev. 2 are unclear. Please provide a complete list of references and/or the NRC staff safety evaluations showing how the Mark-BW fuel design evolves to the Advanced W17 HTP fuel design.

12. In responding to the NRC staff's request for additional information, the licensee stated (ADAMS Accession No. ML113200023, page 11):

    > If fuel rod burnups were to exceed 54 GWd/mtU [gigawatt-days/metric ton of uranium] and any pins exceed the LHGR [linear heat generation rate] of 6.3 kW/ft [kilowatts per foot], then the gap release fraction for the non-LOCA events would be conservatively doubled or evaluated using the ANSI/ANS-5.4 [American National Standards Institute/American Nuclear Society] methodology based on the maximum burnup.

    Please provide detailed description in (a) the criteria of selecting either analytical methodology in determining the gap release fraction, and (b) how the ANSI/ANS-5.4 methodology will be evaluated.

SMALL BREAK LOSS-OF-COOLANT ACCIDENT (SBLOCA) (ANP-2971)

13. Is the moderator density reactivity curve in Table 3-3 based on the most positive moderator temperature coefficient (MTC)? If not, please show the effect of the use of the most positive MTC for moderator density feedback. Please explain.

14. Please explain what is meant by the 166 seconds (sec) trip time for reactor coolant pumps (RCPs). Is this the delay time to RCP trip once the pressure set point for tripping RCPs has been reached? Please explain.

15. Does the high-pressure safety injection (HPSI) curves include allowance for pressure and flow measurement from the surveillance requirement on HPSI flow testing? Please explain.

16. Fig 3-3 shows the nodalization for the vessel with the core barrel leakage paths opened. Was this the leakage path at the top of the downcomer open during the SBLOCA analyses? Please explain. If so, please show the impact of the leakage path closed for the limiting breaks.

17. Please justify the accumulator temperature of 105 °F in the SBLOCA analyses. Does this temperature bound the highest accumulator temperature during the cycle? Please explain.

18. Does the model also account for residual water remaining in the horizontal section of the suction leg piping after the vertical section clears? Please explain. Please show the amount of residual water remaining in the loop seals for the 2.75-, 3.0-, and 9.76-inch diameter breaks.

19. Please show the heat transfer coefficient from 50 to 200 sec on an expanded scale for the 9.76-inch break in Fig 4-23.

20. Please show the sink temperature for the 9.76-inch break versus time. What causes the temperature to decrease at about 75 sec during the steam cooling phase of core

uncovery for the 9.76-inch break in Fig. 4-24? Are there entrained water droplets in the upward flowing vapor? Please explain. Is liquid allowed to downflow into the hot bundle from the upper plenum during periods of steam cooling? Please explain.

21. Please describe the junctions shown connecting the downcomer to the top of the baffle region in Fig. 3-3.

LARGE BREAK LOSS-OF-COOLANT ACCIDENT (LBLOCA) (ANP-2970)

22. Please present the decay heat multiplier chosen for the breaks shown in Fig. 3-8.

23. Please identify the values of the parameters in the sampled breaks in Fig. 3-9. Also provide the containment pressure at the time of peak cladding temperature (PCT) for these cases. The parameters are:

    Case number, PCT (°F), PCT time (sec), case end time (sec), PCT elevation (ft), assembly burnup (GWd/mtU), core power (MWt), planar linear heat generation rate, PLHGR (kW/ft), axial skew (top, bottom), axial shape index, ASI, break type (guillotine, split), one sided break size (ft^2), Tmin (°F), initial hot rod fuel stored energy (°F), decay heat multiplier, film boiling heat transfer coefficient (HTC), dispersed flow film boiling HTC (Btu/hr-ft^2-°F), condensation interphase HTC (Btu/hr-ft^2-°F), initial reactor coolant system (RCS) flow rate (M lb/hr), initial operating temperature(Tcold, °F), pressurizer pressure (psia), pressurizer level (%), containment volume (ft^3), containment temperature (°F), containment pressure at time of PCT (psia), safety injection tank (SIT) temperature (°F), SIT pressure (psia), SIT volume (ft^3), start of broken loop SIT injection (sec), start of intact loop SIT injection (sec), broken loop SIT empty time (sec), intact loop SIT empty time (sec), start of HPSI (sec), low-pressure safety injection (LPSI) available (sec).

24. Please show the lateral k-factors used in the downcomer model for downcomer boiling analyses. How are the k-factors computed? Please explain. What is the worst single failure for the limiting downcomer boiling case? Please explain? What is the maximum refueling water storage tank (RWST) temperature assumed for the limiting downcomer boiling case? How is condensation of emergency core cooling (ECC) in the cold legs and upper downcomer modeled and what is the sensitivity of downcomer boiling to the condensation coefficient? Please also show the downcomer fluid temperatures versus time compared to saturation for the limiting downcomer boiling case.

25. Please provide the decay heat multipliers for the breaks in Fig. 3-9.

26. Does the LBLOCA methods account for loop seal refilling? Please explain.

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