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March 26, 2003

BFN-TS-405

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop OWFN, P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 1, 2, AND 3 - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATING TO TECHNICAL SPECIFICATIONS (TS) CHANGE NO. TS-405 - ALTERNATIVE SOURCE TERM (AST) (TAC NOS. MB5733, MB5734, MB5735)

This letter provides a response to requests for additional information in support of TS-405. TS-405 was submitted to NRC on July 31, 2002, and requested a license amendment and TS changes for a full scope application of AST methodology for BFN Units 1, 2, and 3. Enclosures 1 and 2 provide the TVA responses along with the NRC questions.

Enclosure 1 provides a response to two questions, which were provided on February 4, 2003. These questions were further discussed with NRC Staff in a February 20, 2003, telephone conference.

Previously, on October 15, 2002, NRC provided a request for information on TS-405, which was answered in a letter dated December 9, 2002. In that letter, TVA replied to requests 1 through 14 and request 16. During a teleconference on

November 4, 2002, TVA indicated that a reply to request 15 would be held pending further NRC clarification, which was subsequently provided in a teleconference on December 12, 2002. Enclosure 2 provides TVA's reply to request 15. This completes the response to the October 15, 2002, information request.

Also on February 4, 2003, TVA and NRC staff participated in a telephone conference discussion concerning the review of the new main steam line break (MSLB) instantaneous ground level puff release dispersion χ/Q model that was presented for use in support of the July 31, 2002, TS-405 license amendment as shown in Safety Assessment Table 3-2. As a result of this discussion, TVA has decided not to seek approval of the new MSLB puff dispersion χ/Q model. Instead, TVA will continue to use the current licensing basis χ/Q values presented in Table 14.6-8 of the Updated Final Safety Analysis Report (UFSAR) for the MSLB analysis.

The control room dose result (0.409 rem total effective dose equivalent) provided in the TS-405 dose tables for the MSLB accident, which was calculated based on the new puff release dispersion χ/Q (4.60E-4 sec/m^3), is conservative compared to the result that would be calculated based on the current UFSAR χ/Q (3.22E-4 sec/m^3). Since the puff release χ/Q dose result is bounding for the existing UFSAR χ/Q, no recalculation will be performed at this time. In Enclosure 5 of the July 31, 2002, TS-405 submittal, TVA proposed changes to the UFSAR associated with the implementation of AST, which included the use of the puff model χ/Q. Based on the above discussion, following approval of AST by NRC, the final changes to the UFSAR will not include discussions of the new MSLB puff release dispersion χ/Q values, but rather will retain the current UFSAR χ/Q values.

If you have any questions about this response, please telephone me at (256) 729-2636. Pursuant to 28 U. S. C. § 1746 (1994), I declare under penalty of perjury that the foregoing is true and correct. Executed on this day 24th day of March, 2003.

Sincerely,

original signed by:

T. E. Abney
Manager of Licensing
and Industry Affairs
U.S. Nuclear Regulatory Commission

Enclosures:

1. Response To February 4, 2003, Request For Additional Information (RAI) Relating To Technical Specifications Change No. TS-405 - Alternative Source Term (AST)

2. Response To October 15, 2002, Request For Additional Information (RAI) Relating To Technical Specifications Change No. TS-405 - Alternative Source Term (AST) (Request 15)

Enclosures
cc (Enclosures):
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s:lic/submit/techspec 405 supplement alternative source term

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

RESPONSE TO THE FEBRUARY 4, 2003, REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATING TO TECHNICAL SPECIFICATIONS CHANGE NO. TS-405 ALTERNATIVE SOURCE TERM (AST)

NRC Request 1

Please provide details of the qualification of the ORIGEN code for use with blended low enriched uranium fuel since the down blended High Enriched Uranium is an off-spec fuel material.

TVA Reply 1

The ORIGEN-Scale (S) code was used to calculate the plant-specific fission product inventories for ATRIUM-10 fuel bundles including analyses for both commercial grade uranium and Blended Low Enriched Uranium (BLEU). This is the SCALE version of ORIGEN issued in September 1998. The code was developed for the NRC and is recommended for use in NRC Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. " The code was obtained by Framatome Advanced Nuclear Power, installed and qualified (i.e., test cases run) per applicable code quality assurance procedures in April 2000.

The code was designed to handle light water reactor fuel and allows input of individual isotopes that make up the fuel assemblies (U-236 is included in the library). Accordingly, BLEU fuel is modeled by its individual uranium constituents, i.e., U-236 is added to the normal beginning-of-life fuel makeup input of U-234, U-235, and U-238.

The only significant difference between BLEU and commercial grade uranium is in the isotopic content of the uranium (typically around 0.09% U-234 and 1.6% U-236 is contained in the BLEU fuel). All other isotopic differences are on the trace level and would have no impact on source term calculations. Otherwise, there is no difference between BLEU and commercial uranium fuel relative to use of the ORIGEN code. Therefore, the ORIGEN code is qualified to calculate the plant-specific fission product inventories for BLEU fuel.

NRC Request 2

Please provide additional details of the projected fuel requirements for full Extended Power Uprate conditions, including the projected number of each type of fuel and fuel enrichments.

TVA Reply 2

Typical fuel cycle analysis results for the Extended Power Uprate (EPU) cycles are shown in the following tables:

GE-14 Fuel Requirements for EPU[1]		
Commercial Uranium EPU	320 bundles of GE-14	@ 4.16 % U-235
Transition Cycle	16 bundles of GE-14	@ 3.67 % U-235

1 Provided for information, but current plans do not include use of fresh GE14 fuel for EPU cycles.

ATRIUM-10 Fuel Requirements For EPU[2]		
BLEU EPU Equilibrium Cycle	380 bundles of A-10	@ 4.27% U-235

2 This loading assumed a higher U-236 content of 1.8%. Preliminary results indicate that an optimized loading using a more realistic 1.6% U-236 content is on the order of 368 bundles at 4.29% U-235.

Fuel enrichments do not change significantly for EPU conditions since the enrichment licensing limit is expected to remain at the current 5% U-235. Boiling Water Reactor assemblies cannot be designed with a radially uniform enrichment distribution due to the need to control rod-to-rod peaking. The presence of water gaps around the assembly results in higher moderation at the edge of the assembly. This requires the use of lower enrichment pins in these locations, which limits the bundle average enrichment to a value below the licensing limit.

The fuel enrichments chosen for the AST analysis were based upon the desire to bound future operations. The bundle average enrichments for the commercial grade uranium cases bound any possible bundle designs for the respective product lines that still meet the licensing limit for the peak enrichment pins. The higher enrichment for the BLEU bundles is based upon selecting a bundle of equivalent reactivity to the ATRIUM-10 commercial grade uranium case. BLEU bundles contain approximately 1.6% U-236, which acts as a neutron poison and reduces the reactivity of the bundle. This approach provides a bounding case for BLEU fuel in case credit may be allowed in the future for an equivalent enrichment limit (equivalent to the 5% current limit). The increase in enrichment limit is not being requested, but is being included in the AST analysis in the event it is needed in the future.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

RESPONSE TO THE OCTOBER 15, 2002, REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATING TO TECHNICAL SPECIFICATIONS CHANGE NO. TS-405 ALTERNATIVE SOURCE TERM (AST) (REQUEST 15)

NRC Request 15

The full implementation of AST analyses will modify the licensing bases by adopting AST methodology which replaces the current accident source term with an alternative source term as prescribed in 10 CFR 50.67 and establishes the 10 CFR 50.67 total effective dose equivalent (TEDE) dose limits as a new acceptance criteria. Provide a discussion of the impact on environmental qualification (EQ) based on the doses using the AST. In addition, discuss the impact of the postulated increase in the cesium concentration 30 days following the an accident with regard to the calculated dose and the component's qualification dose.

TVA Reply 15

RG 1.183, Section 6, indicates that licensees may use either the AST or the TID-14844 assumptions for performing the required EQ analyses.

Fission product source terms for both the initial BFN power uprate (from 3293 to 3458 MWt) and for EPU (to 3952 MWt) were determined for their respective power levels using the ORIGEN computer code. The radiation doses calculated at the original licensed thermal power (3293 MWt) using TID-14844 assumptions were adjusted for a 5% increase in core thermal power by comparing the ORIGEN source terms at the increased power level to the original fission product source terms. The resulting adjusted radiation doses were then used for EQ analyses. The same methodology was applied to adjust the radiation doses for a 20% increase in thermal power (to 3952 MWt) for the EQ analyses at the EPU power level. Therefore, the use of TID-14844 assumptions is being continued in the BFN EQ program with radiation doses originally determined by the usage of TID-14844 assumptions updated via comparison of increased power level source terms to the original source terms to reflect the effect of the increased power level. This methodology is summarized in Section 2.1.5 of the Safety Assessment (Enclosure 4) in the July 31, 2002, TS-405 submittal and is consistent with RG 1.183, Section 6.