November 3, 2000

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

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - REVIEW OF HOLTEC TOPICAL REPORTS HI-992349 AND HI-992302 REGARDING BORON CREDIT IN SPENT FUEL POOL (TAC NOS. MA9101 AND MA9102)

Dear Mr. Scalice:

The U.S. Nuclear Regulatory Commission (NRC) has reviewed the Holtec International topical reports submitted by Tennessee Valley Authority on April 21, 2000, to allow the use of credit for soluble boron in spent fuel pool criticality safety analyses of the Sequoyah Nuclear Plant, Units 1 and 2. The first report (HI-992349) provides the criticality safety analysis. The second report (HI-992302) provides the boron dilution analysis. These reports were provided in advance of a Sequoyah Technical Specification (TS) change request for soluble boron credit. The reports are based on the NRC-approved Westinghouse Owners Group generic methodology for crediting soluble boron given in Westinghouse Topical Report WCAP-14416-NP-A.

The NRC finds reports HI-992349 and HI-992302 to be acceptable. Attached is our Safety Evaluation. The NRC staff is in the process of reviewing the associated TS change request submitted on August 31, 2000.

Sincerely,

/RA/
Ronald W. Hernan, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and50-328

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO HOLTEC INTERNATIONAL TOPICAL REPORTS HI-992349 AND HI-992302

BORON CREDIT IN THE SPENT FUEL POOL

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

In a letter of April 21, 2000 (Reference 1), supplemented by letter of September 29, 2000 (Reference 2), Tennessee Valley Authority (TVA) requested U.S. Nuclear Regulatory Commission (NRC) review of two topical reports which support the Sequoyah Nuclear Plant (SQN) Unit 1 and 2 long-term spent fuel storage program. The first report (HI-992349) provides the criticality safety analysis (Reference 3) and the second report (HI-992302) provides the boron dilution analysis (Reference 4) to allow the use of credit for soluble boron in the spent fuel pool criticality safety analyses. The criticality safety analysis (Reference 3) was performed using the methodology developed by the Westinghouse Owners Group and described in WCAP-14416-NP-A, Rev. 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology" (Reference 5). A boron dilution evaluation was performed (Reference 4) to confirm that design features, instrumentation, administrative procedures, and sufficient time are available to detect and mitigate boron dilution in the spent fuel pool before the boron concentration is reduced below the value assumed in the criticality safety analysis.

2.0 EVALUATION

Criticality Safety Analyses

NRC General Design Criterion (GDC) 62 requires prevention of criticality in fuel handling and storage areas outside the reactor. The SQN spent fuel storage racks were analyzed using the Westinghouse methodology which has been reviewed and approved by the NRC (Reference 5). This methodology takes partial credit for soluble boron in the fuel storage pool criticality analyses and proposes conformance with the following NRC acceptance criteria to meet GDC 62 requirements for preventing criticality outside the reactor:

1) k_{eff} shall be less than 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties at a 95% probability, 95% confidence (95/95) level as described in WCAP-14416-NP-A, Rev 1; and

2) k_{eff} shall be less than or equal to 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level as described in WCAP-14416-NP-A, Rev. 1.

The primary analysis of the reactivity effects of fuel storage in the SQN spent fuel racks was performed with the three-dimensional Monte Carlo code, KENO5a, with the 238-group SCALE-4.3 neutron cross section library developed by the Oak Ridge National Laboratory. An independent method of analysis using the MCNP Monte Carlo code was used to verify the reference KENO5a calculations. Since the KENO5a code package does not have burnup capability, depletion analyses and the determination of small reactivity increments due to manufacturing tolerances were made with the two-dimensional transport theory code, CASMO4. The analytical methods and models used in the reactivity analysis have been benchmarked against experimental data for fuel assemblies similar to those for which the SQN racks are designed and have been found to adequately reproduce the critical values. This experimental data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include close proximity storage and strong neutron absorbers. Validity of the CASMO4 code was also established by comparison with KENO5a and MCNP calculations for a comparable storage rack configuration. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the SQN storage racks with a high degree of confidence.

The SQN spent fuel pools are each divided in three regions. Region 1 is analyzed to store high reactivity (fresh or low burnup) fuel with a maximum enrichment of 5.00 weight percent (w/o) U-235 in a checkerboard pattern alternating with burned fuel (one fresh assembly out of four burned assemblies). Region 2 is analyzed to store lower reactivity (lower enrichment or higher burnup) fuel. Region 3 is analyzed to store fresh fuel alternating with water-filled cells. Each storage cell is composed of a single Boral neutron absorber panel positioned between two 8.75-in. I.D., 0.060-in. thick stainless steel boxes. Peripheral cells use a 0.060-in. stainless steel sheathing on the outside supporting the Boral panel. The nominal spacing between fuel assemblies is 8.97 in. The Boral absorber has a nominal thickness of 0.102 in. and an as-built nominal areal density of 0.03388 $g/cm.^2$

Calculations for both Westinghouse 17x17 fuel and Framatome Mark-BW17 fuel indicate that the Westinghouse 17x17 fuel exhibits a slightly higher reactivity worth and was therefore used in the analyses.

The moderator was assumed to be pure water at a temperature of 20^{o}C and the array was assumed to be infinite in lateral (x and y) extent. Uncertainties due to tolerances in fuel enrichment and density, stainless steel thickness, lattice spacing, assembly position, Boral width, boron-10 (B-10) loading, calculational uncertainty, and methodology bias uncertainty were accounted for. These uncertainties were appropriately determined at the 95/95 probability/confidence level. A methodology bias (determined from benchmark calculations) as well as a reactivity bias to account for the effect of a reduction in pool water temperature to 4°C were included. An uncertainty was also applied to the depletion calculations. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

For the Region 1 racks with a checkerboard pattern of one fresh assembly and three spent fuel assemblies (in four assembly groups), minimum burnup values were determined that result in a maximum k_{eff} of less than 1.0, including all biases and uncertainties, with no soluble boron in the pool water. In the base case, no credit was taken for the presence of integral fuel burnable absorbers (IFBAs) or gadolinia rods in the fresh fuel. The analysis also included spent fuel decay time (cooling time) credit, which results from the radioactive decay of isotopes in the spent fuel. Credit is taken only for the decay of actinides, the major contributor being the decay of Plutonium-241 to Americium-241. The loss in reactivity due to the radioactive decay of the spent fuel results in reducing the minimum burnup needed to meet the reactivity requirements. The results of these analyses for Region 1 were put in the form of curves (Figure 1-2 of Reference 3) showing the minimum acceptable burnup for fuel of various initial enrichments along with the required cooling times for the spent fuel assemblies in the pool. All points on the curves have the same maximum 95/95 k_{eff} of approximately 0.9975, thus meeting the NRC criterion of k_{eff} less than 1.0 for precluding criticality with no credit for soluble boron.

The effect of IFBAs and gadolinia present in some of the fresh fuel rods was also analyzed for the Region 1 checkerboard configuration with no credit for soluble boron. No credit for cooling time was taken in these latter calculations. The minimum acceptable burnup for fuel of various initial enrichments, as a function of the number of IFBA rods in the fresh fuel assemblies, is shown in Figure 1-4 of Reference 3. Likewise, the minimum acceptable burnup for fuel of various initial enrichments, for varying numbers of gadolinia rods in the fresh fuel assemblies, is shown in Figure 1-3 of Reference 3. All points on these curves have a maximum calculated 95/95 k_{eff} of less than 1.0.

Calculations were done for storage of spent fuel assemblies that are face adjacent to each other (designated as Region 2) with no credit for soluble boron. “Face adjacent” is defined to mean that the flat surface of a fuel assembly in one cell faces the flat surface of the fuel assembly in the next cell. For each initial enrichment and spent fuel cooling time, minimum burnup values were determined that resulted in a maximum k_{eff} of less than 1.0, thus meeting the NRC criterion for precluding criticality with no soluble boron. Figure 1-1 of Reference 3 summarizes the results of these analyses, showing the minimum acceptable burnup for fuel of various initial enrichments along with the cooling time of the spent fuel in the pool. Again, all points on these curves have a maximum calculated 95/95 k_{eff} of less than 1.0. Additional calculations for spent fuel currently in storage indicate that almost all can be safely stored in this face adjacent configuration.

The third configuration analyzed was that of fresh fuel assemblies alternating with empty (water-filled) cells (designated as Region 3). This checkerboard configuration resulted in a maximum calculated 95/95 k_{eff} of less than 0.95 even without credit for any soluble boron. Additional analyses for this configuration showed that 75% of the water can be displaced from empty cells by non-fuel bearing assembly components such as thimble plugs and boral coupon trees.

Soluble boron credit is used to provide safety margin by maintaining k_{eff} less than or equal to 0.95 including 95/95 uncertainties. The minimum soluble boron required to maintain k_{eff} less than or equal to 0.95 including 95/95 uncertainties was determined to be 300 ppm for the Region 1 and Region 2 storage configurations. The required amount of soluble boron is well below the minimum spent fuel pool boron concentration value of 2000 ppm required by plant technical specifications (TS) and is, therefore, acceptable.

The potential effects of several different fuel misloading events were evaluated. The misloading of a fresh Westinghouse 17x17 fuel assembly of maximum enrichment of 5.0 w/o U-235 into a cell intended for spent fuel in the above-analyzed storage configurations results in the highest reactivity increase. However, the minimum spent fuel pool boron concentration value of 2000 ppm required by TS is more than sufficient to maintain k_{eff} less than or equal to 0.95 for this reactivity increase. In fact, a minimum boron concentration of 700 ppm is sufficient to maintain k_{eff} less than or equal to 0.95 for the worst misloading event. By virtue of the double contingency principle, which has been endorsed by the staff, two unlikely, independent, concurrent events are beyond the scope of the required analysis. Therefore, credit for the presence of the entire amount of soluble boron above that required to maintain k_{eff} less than or equal to 0.95 under normal conditions (300 ppm) may be assumed in evaluating other accident conditions such as a fuel misloading since the causes of fuel misloading and boron dilution are unrelated and not assuming its presence would be a second unlikely event. A boron dilution event from the minimum TS value of 2000 ppm to 300 ppm is evaluated below.

Boron Dilution Event

The NRC safety evaluation report for crediting soluble boron (Reference 5) states that potential events which could dilute the spent fuel pool soluble boron to the concentration required to maintain the 0.95 k_{eff} limit should be identified. In addition, the available time span of these dilution events should be quantified to show that sufficient time is available to enable adequate detection and suppression of any dilution event.

Deterministic plant-specific dilution event calculations were performed for SQN by Holtec International and presented in HI-992302 in order to define the dilution times and volumes necessary to dilute the spent fuel pool from the minimum TS boron concentration of 2000 ppm to a soluble boron concentration where a k_{eff} of 0.95 would be approached (300 ppm). The water volume in the pool is approximately 265,200 gallons, assuming the maximum number of allowed stored fuel assemblies are in place and excluding the cask handling area and transfer canal. In order to dilute 265,200 gallons in the spent fuel pool from the TS limit of 2000 ppm to 300 ppm, 547,000 gallons of unborated water is needed. The various initiating events considered included dilution from the chemical and volume control system holdup tanks, demineralized water system, component cooling water system, drain systems, fire protection system, and spent fuel pit demineralizer, as well as other events that may affect the boron concentration of the pool, such as seismic events, random pipe breaks, and loss of offsite power.

The Holtec evaluation of these sources concluded that the addition of unborated water from the fire header break results in the shortest boron dilution time. A hypothetical maximum flow of 5,000 gpm (same as the capacity of two fire pumps) can be postulated but the flow will spill on the floor and escape through the floor drains and the stairwell openings. Also, there is a 2-in. curb around the pool which would help to prevent any significant flow into the pool. However, if it is postulated that somehow the break flows into the spent fuel pool, a high-level alarm would be generated within a minute as well as a fire protection system pressure drop alarm, both in the control room, alerting operators to stop the fire protection pumps and isolate the break. If the flow continued, the pool could be diluted from 2000 ppm to 300 ppm in less than 2 hours. However, since this would require the flooding of the auxiliary building with 517,000 gallons of water on the floor to go unnoticed, this scenario is not considered to be credible.

There are two fire hose stations located in the vicinity of the spent fuel pool. The 2.5-in. supply pipe is about 20 ft. from the pool boundary and is about 6 ft. high. Ignoring the reduction that would be afforded due to the presence of a 3 ft. high plexiglass skirt at the pool boundary, a split rupture of the fire hose supply line could result in a flow of 2100 gpm. More than 4 hours would be required to dilute the pool below 300 ppm under these conditions. However, based on control room alarms (initiated in about 1 minute) which would result from (1) start of the fire protection pumps due to low pressure in the fire protection piping, and (2) high level in the spent fuel pool, the dilution would be detected and terminated well before a boron concentration of 300 ppm could be reached.

The scenario yielding the next shortest dilution time is postulated as the continuous flow of 300 gpm of non-borated water into the pool resulting from two fire hoses with 150 gpm flow each left unattended. The high level alarm would be initiated in 7.8 minutes. If the flow is not isolated, the 300 ppm boron concentration would be reached in over 30 hours. This would probably be detected by an operator on rounds within 12 hours.

The next shortest dilution time and the worst-case scenario for normal operation is due to the continuous drawdown of unborated water from the demineralized water makeup sources in the case of a manual isolation valve inadvertently left open. If the 250 gpm flow were not terminated by a high level alarm initiated at 9.4 minutes, the 300 ppm boron concentration would be reached in 36.5 hours. This would also probably be detected by an operator on rounds within 12 hours.

Holtec has stated that an event which could dilute the spent fuel pool boron concentration from 2000 ppm to 300 ppm is not a credible event. The NRC staff concludes that the combination of the large volume of water required for a dilution event, TS required minimum boron concentration, weekly sampling requirement, spent fuel pool alarms, plant personnel rounds, and other administrative controls and procedures, should adequately detect a dilution event prior to k_{eff} reaching 0.95 (300 ppm).

Additionally, the criticality analyses for the spent fuel pool show that k_{eff} remains less than 1.0 at a 95/95 probability/confidence level even if the pool were completely filled with unborated water. Therefore, even if the spent fuel storage pool were diluted to 0 ppm, the stored fuel is expected to remain subcritical, thereby meeting the requirements of GDC 62.

3.0 CONCLUSION

The staff has reviewed the criticality safety analysis presented in Holtec report HI-992349and the boron dilution event analysis presented in HI-992302. The analysis conforms to NRC staff guidance (Reference 6) and demonstrates that the spent fuel storage racks in SQN can safely accommodate fuel with initial nominal enrichments up to 4.95 ±0.05 w/o U-235, with assurance that under normal and accident conditions the maximum reactivity, including calculational and manufacturing uncertainties and credit taken for soluble boron, will be less than 0.95, with 95 percent probability at the 95 percent confidence level, provided the fuel conforms to the enrichment, burnup limits, cooling time and loading patterns given in the report. This meets the staff’s subcriticality margin guidelines as well as the subcriticality requirements of GDC 62.

4.0 REFERENCES

1. P. Salas, TVA, letter to NRC, "Sequoyah Nuclear Plant (SQN) - Holtec International Topical Reports,” April 21, 2000.

2. P. Salas, TVA, letter to NRC, "Sequoyah Nuclear Plant (SQN) Units 1 and 2 - Response to Request for Additional Information on the Holtec International Reports (TAC NOS. MA9101 and MA9102),” September 29, 2000.

3. “Criticality Safety Analyses of Sequoyah Spent Fuel Racks with Alternative Arrangements,” HI-992349, Holtec International, April 20, 2000

4. “Boron Dilution Analysis Sequoyah Nuclear Plant Units 1 and 2,” HI-992302, Holtec International, April 20, 2000.

5. W. D. Newmyer, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," Westinghouse Electric Corporation, WCAP-14416-NP-A, Rev. 1, November 1996.

6. L. Kopp, Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants, NRC Memorandum to T. Collins, August 19, 1998.

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Date: November 3, 2000