August 22, 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 — ISSUANCE OF AMENDMENT REGARDING ONE-TIME EXTENSION OF APPENDIX J, TYPE A, INTEGRATED LEAK RATE TEST INTERVAL TECHNICAL SPECIFICATION CHANGE REQUEST (TAC NO. MC9239)

Dear Mr. Singer:

The Commission has issued the enclosed Amendment No. 63 to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant, Unit 1. This amendment is in response to your application dated December 14, 2005 (TS-05-07), as supplemented by letter dated March 31, 2006.

The amendment revises Technical Specification Section 5.7.2.19, "Containment Leakage Rate Testing Program." The change will allow a one-time, 5-year extension to the current 10-year test interval for the containment integrated leakage rate test. The revision is based on the risk-informed approach developed using Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

A copy of the safety evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

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

Enclosures: 1. Amendment No. 63 to NPF-90
2. Safety Evaluation

cc w/enclosures: See next page

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WATTS BAR NUCLEAR PLANT

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-390

WATTS BAR NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 63
License No. NPF-90

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Tennessee Valley Authority (the licensee) dated December 14, 2005, as supplemented by letter dated March 31, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (I) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-90 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 63, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, and shall be implemented no later than 45 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Lakshminaras Raghavan, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Change to the Technical Specifications

Date of Issuance: August 22, 2006

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 63</u>

<u>FACILITY OPERATING LICENSE NO. NPF-90</u>

<u>DOCKET NO. 50-390</u>

Replace page 3 of Operating License No. NPF-90 with the attached page 3.

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

<u>REMOVE</u>	<u>INSERT</u>
5.0-28	5.0-28

(4) TVA, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis, instrument calibration, or other activity associated with radioactive apparatus or components; and

(5) TVA, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

TVA is authorized to operate the facility at reactor core power levels not in excess of 3459 megawatts thermal.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 63 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Safety Parameter Display System (SPDS) (Section 18.2 of SER Supplements 5 and 15)

Prior to startup following the first refueling outage, TVA shall accomplish the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to having the Watts Bar Unit 1 SPDS operational.

(4) Vehicle Bomb Control Program (Section 13.6.9 of SSER 20)

During the period of the exemption granted in paragraph 2.D.(3) of this license, in implementing the power ascension phase of the approved initial test program, TVA shall not exceed 50% power until the requirements of 10 CFR 73.55(c)(7) and (8) are fully implemented. TVA shall submit a letter under oath or affirmation when the requirements of 73.55(c)(7) and (8) have been fully implemented.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. NPF-90

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-390

1.0 INTRODUCTION

By application dated December 14, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML053540072), as supplemented by letter dated March 31, 2006 (ADAMS Accession No. ML060950497), Tennessee Valley Authority (TVA, the licensee) requested changes to the Technical Specifications (TSs) of the license of Watts Bar Nuclear Plant (WBN) Unit 1. The proposed amendment would revise TS 5.7.2.19, "Containment Leakage Rate Testing Program." The change will allow a one-time, 5-year extension to the current 10-year test interval for the containment integrated leakage rate test. The revision is based on the inservice inspection (ISI) requirements of the containment structure and the risk-informed approach developed using Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

Notice of this amendment was given in the *Federal Register* on February 28, 2006 (71 FR 10078). The letter dated March 31, 2006, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. WBN TS 5.7.2.19, "Containment Leakage Rate Testing Program," requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) report NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995.

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at WBN have been successful, so the current interval requirement is 10 years.

The licensee is requesting a change to TS 5.7.2.19 which would add an exception from the guidelines of RG 1.163 and NEI 94-01, Revision 0, regarding the Type A test interval. Specifically, the added sentence states, "The Fall 2007 end date for conducting the 10-year interval containment integrated leakage rate (Type A) test may be deferred up to 5 years but no later than Fall 2012."

The local leakage rate tests (Type B and Type C tests), including their schedules, are not affected by this request.

3.0 TECHNICAL EVALUATION

The most recent Type A test performed at Watts Bar was conducted in September 1997. Absent the requested extension, the next Type A test would have to be performed during the Fall 2006 refueling outage.

The leak rate testing requirements of Option B of Appendix J, and the containment ISI requirements mandated by 10 CFR 50.55a complement each other to ensure the continued leak-tightness and structural integrity of the containment. The staff's evaluation of the licensee's application is based upon a review of the risk assessment and a detailed evaluation related to the ISI of the containment and potential areas of weaknesses in containment.

3.1 Inservice Inspection for Primary Containment Integrity

WBN Unit 1 is a Westinghouse pressurized-water reactor. The primary structure consists of a freestanding steel containment vessel (SCV) with an ice condenser and a separate secondary reinforced concrete shield building. The SCV consists of a cylindrical wall (with access penetrations, process piping and electrical penetrations), a hemispherical dome, and a bottom liner plate encased in concrete. The integrity of the penetrations and containment isolation valves is verified through Type B and Type C local leak rate tests (LLRT), as required by 10 CFR Part 50, Appendix J, and the overall integrity of the containment structure is verified through a Type A test. These tests are performed to verify the essentially leak-tight characteristics of the containment at the design basis accident (DBA) pressure. The last integrated leak rate test for WBN's primary containment was performed in September 1997. With the extension of the Type A test interval, the licensee is committing to perform the next Type A test prior to the fall of 2012. The Type A test, the LLRTs, and ISI of the containment collectively ensure the continued leak-tight and structural integrity of the containment.

The licensee notes that the proposed TS change is based on performance history from previous Type A tests and WBN's American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, Subsection IWE examination and inspection program. The previous WBN Type A test results have shown leakage to be below the 1.0 La leakage limit. The performance leakage rates for the last two tests were 0.0143 percent/day which is equivalent to 0.0572 La (June 1994) and 0.0444 percent/day which is equivalent to 0.1776 La (September 1997). The licensee emphasizes that the margins to date from previous tests indicate at least 80 percent margin (worst case).

The licensee notes that the risk is further minimized by continued 10 CFR Part 50, Appendix J, Type B and Type C testing. WBN's ISI program provides additional confidence in containment structural integrity and leak tightness. Based on this assessment, the licensee notes that the proposed extension of the Type A test represents minimal risk for increased leakage.

Based on the susceptibility of penetration bellows to crack, the staff has focused its attention on mechanical bellows in steel containments. The licensee describes the results of containment penetrations with mechanical bellows as follows:

> The WBN containment penetration mechanical bellows are within the scope of containment inspection and Appendix J Type A, B, or C leak testing and are two-ply laminated testable bellows. Each bellow is local leak rate tested (Type B) by pressurizing between the two plies. These bellows incorporate a screen mesh between the inner and outer plies to ensure separation is maintained. This design prevents a "pinch" from occurring at the folds and ensures that the entire space between the plies is pressurized and leak tested during Type B testing. If the bellows test fails the Appendix J, Type B test, the bellows' sheet metal cover is removed, the bellows are pressurized to test pressure, and visually inspected for leakage using a bubble solution (snoop), lights, mirrors, etc. The bellows are repaired or replaced as necessary if the bellows are found to be leaking. A review of TVA records since startup in May 1996 has revealed no test failures of these bellows for WBN Unit 1.

The NRC staff finds the process used in identifying bellows degradation acceptable.

The licensee notes that Option B of 10 CFR 50, Appendix J allows extended test intervals up to 120 months for Type B components, based on acceptable performance. However, due to industry concerns, WBN has limited extended test intervals for bellows to 60 months. Additionally, penetrations with bellows are tested on a staggered basis such that a portion are tested each refueling outage.

The licensee, thus, states that the one-time 5-year (from 10 to 15 years) extension of the Type A test frequency has no effect on this testing since the frequency of inspection and testing of these bellows is limited to 60 months.

As the majority of Type A test failures were related to Type B and Type C tests, the NRC staff finds the licensee’s process and frequency for testing reasonable and acceptable.

The licensee has provided information related to containment examination pursuant to Subsection IWE of Section XI of the ASME Code, as follows:

> TVA performs inspection activities on the containment structure that also support performance of the required Type A test. WBN performs containment inspections in accordance with the ASME Section XI Subsection IWE ISI program. The IWE program will continue to perform inspection activities on WBN Unit 1 containment through the proposed Appendix J test extension interval. TVA's IWE program is based on the applicable portions of Subsections IWA and IWE of the 1992 Edition, Winter 1992 Addenda, of ASME Section XI. The first inspection interval for the containment ISI program began September 9, 1998, and ended September 8, 2001. The second inspection period ended September 8, 2005, and the third inspection period will end September 8, 2008,

in accordance with ASME Section XI. The second inspection interval for containment will begin September 9, 2008. Visual examinations of the WBN SCV have been performed in accordance with the IWE program. To date, no indications of containment degradation have been found. These periodic IWE examinations provide assurance that degradation of the containment structure will be detected and corrected before it can affect structural integrity or leak tightness.

A general visual examination was performed on the WBN SCV during the Cycle 3 and 6 Refueling Outages. These examinations were performed to meet the ASME Section XI, Subsection IWE, Table IWE-2500-1, Examination Category E-A, Item Number E1.11 requirements and the WBN TS 3.6.1.1 requirements. A general visual examination is required to be performed once per inspection period on the accessible exterior surface areas of the SCV per 10 CFR 50.55a (b)(2)(ix)(E). The TS general visual examination is performed on the accessible interior and exterior surface areas of the SCV prior to each 10 CFR 50, Appendix J, Type A, CILRT and during one other refueling outage if the Type A test has been extended to 10 years. There were no conditions identified during these general visual examinations that affected the leak tightness or structural adequacy of the SCV.

WBN issued an initial evaluation of potential areas for augmented containment inservice inspection (CISI) examination (areas likely to experience accelerated aging and degradation). This evaluation was updated based on completion of the WBN Unit 1, Cycle 3, general visual examination. This evaluation determined that there were no areas of WBN SCV which should be considered as requiring augmented examinations in accordance with IWE-1240 and IWE program.

The licensee, also, describes its plans to perform future examinations:

A VT-3 examination to meet ASME Section XI (i.e., Subsection IWE, Table IWE-2500-1, Examination Category E-A, Item Number E1.12) requirement to examine the accessible surface areas at the end of the interval from one side of the SCV is scheduled during the Unit 1 Cycle 8 Refueling Outage (third period of the first containment ISI interval).

The licensee provides an estimation of areas examined during general visual examination:

The total estimated area of the SCV from the base concrete floor slab to the top of the SCV on the exterior side is approximately 61,300 square feet. The inaccessible exterior surface area is estimated to be approximately 2800 square feet due to insulation on the exterior SCV surface and the area around the fuel transfer penetration. It is estimated that 95 percent of the SCV exterior side is assessable for general visual and VT-3 visual examination in accordance with ASME Section XI, Examination Category E-A. The area below the floor is not included in the area for examination because the embedded metal liner and concrete base slab are exempt from examination in accordance with

IWE-1220(b) and IWL-1220(b) of Subsections IWE and IWL of ASME Section XI.

Furthermore, the licensee states that TVA is not proposing any additional IWE examination or nondestructive examinations of the WBN SCV based on the following:

- WBN has no areas identified for augmented examinations in accordance with IWE-1240.
- The WBN SCV general visual examinations performed in accordance with the ASME Code Section XI did not identify any conditions that affected the leak tightness or structural adequacy of the SCV.

Moreover, the licensee states the WBN and Sequoyah Nuclear Plant (SQN) primary containment structures consist of a freestanding steel vessel with an ice condenser and a separate secondary containment that is a reinforced concrete Shield Building. The maximum internal pressure for WBN and SQN is 15.0 and 12.0 pounds per square inch, respectively. Review of the WBN and SQN SCV design drawings indicates that the WBN SCV is designed with greater wall thickness than the SQN SCV. This additional wall thickness for WBN SCV enhances the leak tightness or structural adequacy of the SCV. Based on the comparison of WBN and SQN SCV, the licensee emphasizes that TVA has no plans to perform any additional inspections in inaccessible areas to validate integrity of the SCV.

The SCV areas behind the ice-baskets inside containment are susceptible to corrosion degradation and are only accessible from outside the SCV. If corrosion degradation occurs in these inside areas, it cannot be evaluated unless there is a program to perform ultrasonic testing of these areas (as per IWE-1241(a)). In further discussion, the licensee described TVA's relief request CISI-05, where the staff has approved the use of ASME Code Case –605 for augmented examination. The Code Case provides an alternative to the ultrasonic testing requirements of the Code but does not relieve the licensee from performing augmented examination. The staff notes that if the licensee finds evidences of water seepage in this gap, the licensee must perform the augmented examination as required by the ASME Code.

Based on past performance of the Type A tests, the licensee's procedures to examine and monitor potential age-related and environmental degradations of the pressure retaining components of the WBN primary containment, and programs to conduct leak rate tests of penetrations and bellows, the staff finds that granting the requested Type A test extension will not adversely affect the leak tight integrity of the primary containment.

3.2 <u>Risk Impact of Extending the Type A Test Interval</u>

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. The risk assessment was provided in the December 14, 2005, application for license amendment. Additional analysis and information was provided by the licensee in its letter dated March 31, 2006. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during the development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," provided the technical basis to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement this basis, industry undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The Appendix J, Option A, requirements that were in effect for Watts Bar early in the plant's life required a Type A test frequency of three tests in 10 years. The EPRI study estimated that relaxing the test frequency from three tests in 10 years to one test in 10 years would increase the average time that a leak, that was detectable only by a Type A test, goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the pressurized water reactor and boiling water reactor representative plants in the EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three tests in 10 years to one test in 20 years leads to an "imperceptible" increase in risk that is on the order of 0.2 percent and a fraction of one person-rem per year in increased public dose.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem per year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking was completed in 1995, the staff has issued RG 1.174 on the use of probabilistic risk assessment in evaluating risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 guidance to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking.

RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per year and increases in large early release frequency (LERF) less than 10^{-7} per year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original frequency of three tests in a 10-year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided analyses, as discussed below. The following comparisons of risk are based on a change in test frequency from three tests in 10 years (the test frequency under Appendix J, Option A) to one test in 15 years. This bounds the impact of extending the test frequency from one test in 10 years to one test in 15 years. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

1. Given the change from a three in 10-year test frequency to a one in 15-year test frequency, the increase in the total integrated plant risk is estimated to be about 0.01 person-rem per year. This increase is comparable to that estimated in NUREG-1493, where it was concluded that a reduction in the frequency of tests from three in 10 years to one in 20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test frequency from the original three in 10 years to one in 15 years is estimated to be about 3.7 x 10^{-7} per year based on consideration of internal events and external events (i.e., fire and seismic events.) There is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the IWE/IWL visual examination of the containment surfaces (as identified in American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Subsections IWE/IWL). Visual inspections are expected to be effective in detecting large flaws in the visible regions of containment, and this would reduce the impact of the extended test interval on LERF. The licensee's risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment shell on the proposed change. The increase in LERF associated with corrosion events is included in the above LERF estimate.

   When the calculated increase in LERF is in the range of 10^{-7} per year to 10^{-6} per year, applications are considered if the total LERF is less than 10^{-5} per year. The licensee estimates that the total LERF including the requested change is 2.78 x 10^{-6} per year, which meets the total LERF criteria. The staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved between prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimates the change in the conditional containment failure probability to be an increase of approximately one percentage point for the cumulative change of going from a test frequency of three in 10 years to one in 15 years. The staff finds that the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability for the proposed amendment.

3.3 Summary

The NRC staff finds that the licensee has adequate procedures to examine and monitor potential age-related and environmental degradations of the pressure retaining components of the WBN primary containment. In addition, the staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines while maintaining the defense-in-depth philosophy of RG 1.174. Therefore, based upon the past performance of Type A tests, the ISI procedures and risk assessment described above, the staff concludes that granting a one-time extension for performing the Type A test as proposed by the licensee is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 [and changes surveillance requirements]. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (71 FR 10078). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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