December 16, 2002

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

SUBJECT: SURRY UNIT 1 - ISSUANCE OF AMENDMENT RE: ONE-TIME EXTENSION OF APPENDIX J TYPE A INTEGRATED LEAKAGE RATE TEST INTERVAL (TAC NO. MB3199)

Dear Mr. Christian:

The Commission has issued the enclosed Amendment No. 233 to Facility Operating License No. DPR-32 for the Surry Power Station, Unit No. 1. The amendment changes the Technical Specifications (TS) in response to your application dated October 15, 2001, as supplemented by letters dated November 8, 2001, and June 28 and July 25, 2002.

This amendment revises the TS to allow a one-time change in the Appendix J Type A containment integrated leakage rate test interval from the required 10 years to a test interval of 15 years at Surry Power Station, Unit No. 1.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Christopher Gratton, Sr. Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-280

Enclosures:
1. Amendment No. 233 to DPR-32
2. Safety Evaluation

cc w/encls: See next page

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Distribution: See attached list

ADAMS ACCESSION NUMBER: **ML023510006** *See previous concurrence

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DATE	12/9/2002	12/9/2002	12/10/2002	11 /25/2002	12/10/2002

OFFICIAL RECORD COPY

DATED: December 16, 2002

AMENDMENT NO. 233 TO FACILITY OPERATING LICENSE NO. DPR-32 - SURRY UNIT 1
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VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 233
License No. DPR-32

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated October 15, 2001, as supplemented November 8, 2001, June 28, 2002, and July 25, 2002, 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 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 233, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 16, 2002

ATTACHMENT TO

LICENSE AMENDMENT NO. 233 TO FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

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 vertical lines indicating the area of change.

Remove Page	Insert Page
TS 4.4-1	TS 4.4-1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 233 TO FACILITY OPERATING LICENSE NO. DPR-32

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NO. 1

DOCKET NO. 50-280

1.0 INTRODUCTION

By letter dated October 15, 2001, as supplemented by letters dated November 8, 2001, and June 28 and July 25, 2002, Virginia Electric and Power Company (Dominion), the licensee for Surry Power Station, Unit 1 (Surry 1), requested a technical specification change that would allow a one-time change in their Appendix J Type A test (containment integrated leakage rate test) interval from the required 10 years to a test interval of 15 years.

The November 8, 2001, and June 28 and July 25, 2002, supplements contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

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. Surry 1 Technical Specification 4.4 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) 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 Surry 1 have been successful, resulting in the current extended interval of 10 years.

The licensee is requesting a modification to Technical Specification (TS) 4.4, “Containment Tests,” which would allow the licensee to take an exception from the guidelines of RG 1.163 regarding the Type A test interval. Specifically, the proposed TS states that the first Unit 1 Type A test performed after the April 23, 1992, Type A test shall be performed no later than April 22, 2007.

The licensee is using the 1992 Edition and the 1992 Addenda of Subsections IWE and IWL of Section XI of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (the Code), as amended by the modifications and limitations of 10 CFR 50.55a, for conducting the inservice inspection (ISI) of the Surry 1 containment with approved relief from certain Code requirements. The start date for the current containment ISI 10-year interval for examination of steel liner and penetrations (per Subsection IWE) is April 26, 1997, and that for the concrete examination (per Subsection IWL) is September 1, 1996.

3.0 TECHNICAL EVALUATION

3.1 Risk Evaluation

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. The assessment was provided to the NRC staff in the October 15, 2001, application for license amendment. Additional analysis and information were provided by the licensee in letters dated November 8, 2001, June 28, 2002, and July 25, 2002. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in 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 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,” September 1995, provided the technical basis to support rulemaking 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 the NRC’s rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (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 EPRI study estimated that relaxing the test frequency from 3 in 10 years to 1 in 10 years will increase the average time that a leak 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, in percent of person-rem/year, for the pressurized-water reactor (PWR) and boiling-water reactor representative plants confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 per 10 years to 1 per 10 years leads to an “imperceptible” increase in risk ranging from 0.02 to 0.14 percent.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/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 in 1995, the NRC staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 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 reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor 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 3 in 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 an analysis that estimated all of these risk metrics and whose methodology is consistent with previously approved submittals. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

1. A slight increase in risk is predicted when compared to that estimated from current requirements. Given the change from a 10-year test interval to a 15-year test interval, the increase in the total integrated plant risk, in person-rem/year, is estimated to be 0.001 percent. The increase in the total integrated plant risk, given the change from a 3 in 10-year test interval to a 15-year test interval, was 0.004 percent. NUREG-1493 concluded that a reduction in the frequency of tests from 3 per 10 years to 1 per 10 years leads to an "imperceptible" increase in risk, ranging from 0.02 to 0.14 percent. 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 interval from the original 3 in 10 years to 1 in 15 years is estimated to be 1.2×10^{-7}/year.

   However, there is some likelihood that the undetected flaw in the containment liner estimated as part of the Class 3b frequency would be detected as part of the IWE visual examination process of the containment liner. The containment was visually inspected in 1998 and 2000. The next scheduled IWE containment liner inspection for Unit 1 is during the Spring 2003 refueling outage. Eighty-five percent of the inner containment liner can be visually inspected. Assuming the visual inspections are capable of detecting large flaws in the visible regions of the containment, then the increase in LERF would go from 1.2×10^{-7}/year to 1.8×10^{-8}/year. Therefore, increasing the Type A interval to 15 years is considered to be a very small change in LERF when using the guidelines of RG 1.174.

   The licensee performed additional risk analysis to consider the impact of hypothetical corrosion in inaccessible areas of the containment liner on the proposed change. The inaccessible areas included the back side of the containment liner. The risk analysis considered the likelihood of an age-adjusted liner flaw that would lead to a breach of the containment. The risk analysis also considered the likelihood that the flaw was not visually detected but could be detected by a Type A test. When possible corrosion of the containment liner is considered, the increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be 2.3×10^{-8}/year. This additional risk analysis provides added assurance that increasing the Type A interval to 15 years is a very small change in LERF.

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 among prevention of core damage, prevention of containment failure, and consequence mitigation. Based on information provided by the licensee, the NRC staff estimates the change in the conditional containment failure probability to be an increase of 0.001 for the proposed change and 0.0031 for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the change in the conditional containment failure probability for the proposed amendment.

Based on these conclusions, the NRC 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 and, therefore, is acceptable.

3.2 Mechanical Engineering Evaluation

This Safety Evaluation addresses the potential impact of the licensee's existing programs to inspect and monitor aging degradation of the containment pressure boundary on extending the Containment Integrated Leak Rate Test (ILRT) to no later than April 22, 2007.

Surry 1 is a Westinghouse three-loop PWR housed in a subatmospheric containment. The containment consists of a steel-lined reinforced concrete cylinder with hemispherical dome and a reinforced concrete basemat. The containment is penetrated by access penetrations and other process piping and electrical penetrations. The integrity of the penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) as required by 10 CFR Part 50, Appendix J, and the overall leaktight integrity of the containment is verified through an ILRT. These tests are performed to verify the essentially leaktight characteristics of the containment at the design-basis accident (DBA) pressure. The last ILRT for Surry 1 was performed on April 23, 1992, and the proposed TS change request commits to perform the next ILRT no later than April 22, 2007. Because the ILRTs, the LLRTs, and ISI of the containment collectively ensure the leaktight and structural integrity of the containment, the NRC staff requested additional information regarding the licensee's program for managing the containment degradation through ISI. Some of the ISI-related information has been provided in the October 15, 2001, letter. The NRC staff requested additional information related to the potential areas of weakness in its containment that might not be apparent in the risk assessment. The following is a discussion of the licensee's June 28, 2002, responses to the NRC staff's questions.

In response to the NRC staff's question on the examination of the primary containment pressure boundary seals and gaskets, the licensee states that some portion of Type B tests is performed each refueling outage, staggering the testing to balance the outage work scope. Currently these tests are completed in approximately 60 months.

The NRC staff considers the performance tests (Type B tests) performed by the licensee acceptable for detecting degradation of pressure-retaining seals and gaskets.

For the examination of the pressure-retaining containment bolts, the licensee follows the examination (VT-1) requirement of Examination Category E-G of Subsection IWE of the Code, and the bolted connections are pressure tested during Type B pressure testing as required by Examination Category E-P, Item E9.40. Based on the frequency of Type B testing and the VT-1 examination as required by the Code, the NRC staff concludes that the pressure-retaining bolts will be maintained to perform their function during the ILRT extension period.

In response to a question on degradation of stainless steel bellows, the licensee states that (1) there is a stainless steel bellows inside the containment of the outer tube of the fuel transfer tube, and it does not form the containment pressure boundary; and (2) there are four Inconel bellows between the service water system (SWS) discharge piping and the recirculation spray heat exchangers. They are located inside the containment. However, they are tested as part of the containment boundary during the ILRTs, and also as part of the closed system boundary. Furthermore, the licensee asserts that even if a flaw exists in the bellows that went undetected during the last leakage test, it would most likely not have a driving mechanism to propagate through the bellows since the sections of SWS piping containing the bellows are maintained in a dry condition during plant operation to optimize performance of the heat exchangers. The licensee has experienced pitting due to microbiologically induced corrosion (not transgranular stress-corrosion cracking, as stated in the NRC's question). As a result of this concern, the licensee has completed the replacement of four Inconel 600 bellows with more corrosion-resistant Inconel 625 bellows. The licensee mentions that there are procedures in place that would alert it in case a leak occurs when the system is in service. Based on this explanation, the NRC staff concludes that the containment pressure boundary bellows degradation will not compromise the containment integrity during the extended ILRT interval.

In response to a question related to the effects of degradations in uninspectable areas of the liner, the licensee described the methods used as follows:

- The licensee emphasized the effectiveness of the containment ISI program in detecting significant degradation of the containment liner and the concrete wall and dome.

- The licensee performed a sensitivity analysis considering (a) historical data of liner flaw from the uninspectable side of the liner, (b) fragility analysis to estimate the probability of breaching the containment at the ILRT pressure, (3) the likelihood of visual detection failure to detect such flaws.

- The licensee described the product of the above terms as the likelihood of non-detected containment leakage, which was calculated for the containment cylinder, the dome and the basemat.

- The product of this likelihood and the non-large early release frequency was considered as the increase in LERF due to non-detected containment leakage.

The NRC staff review of the licensee's quantitative sensitive analysis indicated that most of the quantities considered are reasonable. However, the NRC staff does not consider the use of the likelihood of breach above ILRT pressure for the basemat consistent with the potential physical response of the basemat. However, the quantities above the ILRT pressure do not contribute to the LERF calculations related to the ILRT extension.

The NRC staff finds the logic used by the licensee in its sensitivity analysis acceptable.

Based on the licensee's procedures discussed above to preclude excessive degradation of the primary containment components, and incorporation of certain degradation in the risk analysis, the NRC staff finds that granting the requested ILRT extension will not adversely affect the leaktight integrity of the primary containment. It should be noted that Subarticle IWE-5000 of the ASME Code, Section XI requires leak rate testing following repair, modification, or replacement of containment components. An ILRT might be required to confirm that these activities are adequate, and that further degradation does not exist in other areas of the containment. The licensee is required to report serious degradation of the containment pressure boundary pursuant to 10 CFR 50.72 or 10 CFR 50.73.

Based on our review, 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 Surry, Unit 1 containment. For the cases where the degradations could occur in the uninspectable portions of the containment, the licensee has factored in such conditions into its risk assessment. Therefore, granting a one-time extension for performing the ILRT as proposed by the licensee in Section 4.4B of the TS change request is acceptable.

Based on these conclusions, the NRC staff finds that the interval until the next Type A test at Surry 1 may be extended to 15 years, and that the proposed changes to TS Section 4.4 are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comment.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes an inspection or surveillance requirement. The NRC staff has determined that the amendments involve 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (66 FR 64309). 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 amendment.

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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Date: December 16, 2002

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