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

December 18, 2008

Mr. David A. Christian
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION, UNIT NO. 2, ISSUANCE OF AMENDMENT REGARDING THE REVISION OF TECHNICAL SPECIFICATION 4.4, PERTAINING TO THE CONTAINMENT LEAKAGE RATE TESTING PROGRAM (TAC NO. MD7535)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 263 to Renewed Facility Operating License No. DPR-37 for Surry Power Station, Unit No. 2. The amendment changes the Technical Specifications (TSs) in response to your application dated December 17, 2007, as supplemented by letters dated April 24, 2008, and June 27, 2008.

The amendment revises TS 4.4, pertaining to the containment leak rate testing program. The TS change would permit a one-time 5-year extension to the once per 10-year frequency of the performance-based leakage rate testing program for Type A tests, which are done in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program." This one time exception to the RG 1.163 requirement would allow the next Type A test to be performed no later than October 26, 2015.

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,

John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-281

Enclosures:
1. Amendment No. 263 to DPR-37
2. Safety Evaluation

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 263
Renewed License No. DPR-37

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated December 17, 2007, as supplemented April 24, 2008, and June 27, 2008, 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 Renewed Facility Operating License No. DPR-37 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 263, are hereby incorporated in the renewed 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

FOR

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes License No. DPR-37
and the Technical Specifications

Date of Issuance December 18, 2008

ATTACHMENT

TO LICENSE AMENDMENT NO. 263

RENEWED FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Replace the following pages of the License and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
License	License
License No. DPR-37, page 3	License No. DPR-37, page 3
TS	TS
TS 4.4-1	TS 4.4-1

E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2546 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 263 , are hereby incorporated in this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. Deleted by Amendment 54

F. Deleted by Amendment 59 and Amendment 65

G. Deleted by Amendment 227

H. Deleted by Amendment 227

4.4 CONTAINMENT TESTS

Applicability

Applies to containment leakage testing.

Objective

To assure that leakage of the primary reactor containment and associated systems is held within allowable leakage rate limits; and to assure that periodic surveillance is performed to assure proper maintenance and leak repair during the service life of the containment.

Specification

A. Periodic and post-operational integrated leakage rate tests of the containment shall be performed in accordance with the requirements of 10 CFR 50, Appendix J, "Reactor Containment Leakage Testing for Water Cooled Power Reactors."

B. Containment Leakage Rate Testing Requirements

1. The containment and containment penetrations leakage rate shall be demonstrated by performing leakage rate testing as required by 10 CFR 50 Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September, 1995 as modified by the following exception:

   NEI 94-01-1995, Section 9.2.3: The first Unit 2 Type A test performed after the October 26, 2000 Type A test shall be performed no later than October 26, 2015.

2. Leakage rate acceptance criteria are as follows:

   a. An overall integrated leakage rate of less than or equal to La, 0.1 percent by weight of containment air per 24 hours, at calculated peak pressure (Pa).

   b. A combined leakage rate of less than or equal to 0.60 La for all penetrations and valves subject to Type B and C testing when pressurized to Pa.

   Prior to entering an operating condition where containment integrity is required the as-left Type A leakage rate shall not exceed 0.75 La and the combined leakage rate of all penetrations subject to Type B and C testing shall not exceed 0.6 La.

3. The provisions of Specification 4.0.2 are not applicable.

Basis

The leak tightness testing of all liner welds was performed during construction by welding a structural steel test channel over each weld seam and performing soap bubble and halogen leak tests.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 263

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NO. 2

DOCKET NO. 50-281

1.0 INTRODUCTION

By letter dated December 17, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML073511682), as supplemented by letters dated April 24, 2008, and June 27, 2008 (ADAMS Accession No. ML081200869 and ML081790666, respectively), Virginia Electric and Power Company (the licensee) submitted a request for a change to the Surry Power Station, Unit No. 2 (Surry 2), Technical Specifications (TSs). The requested change would allow a one-time 5-year extension of the containment integrated leak rate test (ILRT) interval from 10 years to 15 years, allowing the licensee to perform its next ILRT no later than October 26, 2015. The Surry 2 current 10-year interval for Type A test ends on October 26, 2010. The supplements dated April 24, 2008, and June 27, 2008, provided clarifying information that did not change the scope of the original application and the initial proposed no significant hazards consideration determination.

The reactor containment leakage test program requires the licensee to perform an ILRT, also termed as a Type A test; and local leakage rate test (LLRT) termed as Type B and Type C tests. The Type A test measures the overall leakage rate of the primary reactor containment. Type B tests are primarily intended to detect leakage paths and measure leakage rates for primary reactor containment penetrations. Type C tests are intended to measure containment isolation valve leakage rates. The local leakage rate tests (Type B and Type C tests), including their schedules, are not affected by this request.

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 2 TS 4.4.B.1 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" (Reference 2). RG 1.163 endorses, with certain exceptions, NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," (Reference 3).

RG 1.163, Section C, "Regulatory Position" states "licensees intending to comply with the Option B in the amendment to Appendix J should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 (Reference 3) rather than using test intervals specified in ANSI/ANS-56.8-1994" (Reference 5). The industry guidelines in NEI 94-01, Rev. 0, 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 two, most recent Type A tests at Surry 2 have been successful, thus the current interval requirement is 10 years. The license amendment request would change the 10-year ILRT interval to a 15-year interval based on the performance of the historical plant specific Type A tests and the Containment In-Service Inspection (CISI) results, supported by risk-informed analysis performed in accordance with the guidelines in RG 1.174 (Reference 6).

3.0 TECHNICAL EVALUATION

3.1 Current Containment Integrity

This evaluation addresses the current condition of structural and leak-tight integrity of the containment structure and the adequacy of the licensee's LLRT program and In-service Inspection (ISI) Program to detect and manage aging degradation of the containment so that the structural and leak-tight integrity of the containment will be maintained, if the ILRT test interval is extended as proposed by the licensee.

As described in the licensee's application (Reference 1) and the Updated Final Safety Analysis Report (UFSAR), Surry 2 is a pressurized-water reactor with a steel-lined reinforced concrete primary containment structure with vertical cylindrical wall and hemispherical dome, supported on a flat 10 feet- thick base mat. The base of the foundation mat is located approximately 66 feet below finished ground grade. A waterproof membrane was placed below the containment structural mat and carried up the containment wall to above ground level. The containment wall steel liner is 3/8-inch thick. The steel liner for the mat consists of a 1/4-inch plate and 3/4-inch plate. The 1/4-inch mat (floor) liner plate is overlaid with a reinforced-concrete slab 1 1/2 to 2 feet thick. The steel liner for the dome is 1/2-inch thick. The primary function of the steel liner is to provide a leak-tight barrier against release of radioactive material. No credit has been taken for the presence of the steel liner in the design of the containment structure to resist seismic forces or other design loads. During power operation, Surry 2 is maintained at a sub-atmospheric condition.

The containment pressure boundary consists of the steel liner, containment access penetrations, and penetrations for process piping and electrical services. The integrity of the penetrations and containment isolation valves is verified through Type B and Type C tests as required by 10 CFR 50, Appendix J, and the overall integrity of the containment structure is verified through Type A tests. The tests are performed to verify the leak tight integrity of the containment structure at the design basis accident (DBA) pressure. The leakage rate testing requirements of 10 CFR Part 50, Appendix J, Option B (Type A, Type B and Type C tests), and the Containment In-Service Inspection (CISI) requirements mandated by 10 CFR 50.55a, together ensure the continued leak-tight and structural integrity of the containment during its service life.

3.1.1 Type A Tests

In Reference 1, the licensee stated, that based on the May 1991 and October 2000 Type A tests, the current Type A test interval for Surry 2 is once every 10 years. With the requested 5-year extension of the ILRT interval, the licensee proposed that the next overall verification of the containment leak-tight integrity will be performed no later than October 26, 2015.

In Reference 1, and in response to the NRC staff's RAI (Reference 4), the licensee provided the results of the last three Type A ILRT results as:

Test Date	November 1986	May 1991	October 2000	Acceptance Criteria
Total As-Found Leakage	0.728 La	0.452 La	0.061 La	1.0 La
Total As-Left Leakage	0.638 La	0.418 La	0.06 La	0.75 La

"La" (percent/24 hours), as defined in 10 CFR 50, Appendix J, means the maximum allowable leakage rate at pressure Pa (calculated peak containment internal pressure) as specified for preoperational tests in the TSs. As stated in the Surry TSs, the leakage rate acceptance criteria is defined as: La = 0.1 percent by weight of containment air per 24 hours at calculated peak pressure (Pa).

The results of the Type A ILRT show containment leakage within the established acceptance limit and adequate margin indicating leak-tightness of the Surry 2 containment structure.

Regulatory Position C.3 of RG 1.163 recommends that a visual examination of accessible interior and exterior surfaces of the containment structure should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test based on a 10-year ILRT interval. The NRC staff's RAI requested that the licensee describe the plan to supplement the 10-year interval-based visual inspection requirement to accommodate the requested 15-year ILRT interval. In response to the NRC staff's RAI, the licensee stated (Reference 4) that for the current 10-year ILRT interval, two visual examinations have been completed in October 2003 and October 2006. For the 15-year extended ILRT interval, the licensee stated that one visual examination will be performed prior to the Type A test during the October 2015 outage and an additional visual examination of the containment structure will be performed during one of the prior outages (October 2009 or October 2012). In addition to these examinations, a general visual examination of the containment liner will be performed as part of the CISI.

In summary, for the 15-year extended ILRT interval the containment structure will have at least three visual examinations (two already performed in 2003 and 2006, and one planned for upcoming outages in 2009 or 2012) prior to performance of pre-ILRT visual examination in 2015.

The NRC staff finds the licensee's plan to perform three additional visual examinations prior to pre-ILRT visual examination for a 15-year interval acceptable. It consistent with the intent of Regulatory Position C.3 of RG 1.163 to perform visual examinations of accessible interior and exterior surfaces of the containment system for structural problems prior to initiating a Type A test, and during two other refueling outages before the next Type A, to allow for early uncovering of evidence of structural deterioration.

3.1.2 Type B and C Tests

As stated in the licensee's application, the current interval for Type B and Type C testing of containment penetrations and isolation valves will not be affected by extension of the Type A test interval. In response to the NRC staff's RAI (Reference 4), the licensee provided the as-found Type B and Type C test results (from 1995 to 2006) for the containment electrical and mechanical penetrations. Reference 4 indicates that currently, the electrical penetrations are on a 60-month test interval. The licensee stated, in Reference 4, that Surry 2 has no electrical penetrations with an unacceptable Type B performance history. The mechanical penetrations (including containment isolation valves) are also on a 60-month (maximum) test interval. If a mechanical penetration fails a test, it will be returned to the short interval (every outage) until it passes two consecutive refueling outage tests. Reference 4 indicates that approximately 10 percent of mechanical penetrations tested between 2003 and 2006 had unacceptable as found test results and were placed on the short interval test schedule. Considering the 60-month (maximum) interval for Type B and Type C tests currently implemented for Surry 2, the containment electrical and mechanical penetrations will be tested at least once during the requested 5-year extension period of the ILRT interval.

As described in the UFSAR, and based on the response to the NRC staff's RAI, the fuel transfer tube connects the refueling canal in the containment structure and the spent-fuel pool in the fuel building. The penetration consists of a stainless steel pipe installed inside a sleeve, as shown in Figure 15.5-10 of the UFSAR. The outer pipe of the fuel transfer tube is welded to the containment liner and consists of sections of cylinder connected by bellows. These bellows are designed to accommodate any differential movement between the spent-fuel pool in the fuel building and the refueling canal in the containment structure. The containment boundary is the welded connection at the containment liner to the inner and outer pipes and the double O-ring blind flange on the inner tube. The blind flange is Type B tested every refueling outage. The operating experience at Surry 2 has not identified bellows leakage. The overall penetration integrity is verified during the performance of Type A test.

The local leak rate testing interval for the equipment hatch, escape air lock, and the personnel air lock are performed in accordance with Surry 2 TS requirements and are not affected by this license amendment request.

Based on the above review, the NRC staff agrees that the integrity of the containment pressure boundary penetrations and isolation valves will be effectively monitored through Type B and C testing, as required by 10 CFR Part 50 Appendix J, with the proposed new TS.

3.1.3 Containment In-Service Inspection

In Reference 1, the licensee stated that the first 10-year concrete containment IWL examinations were completed by August 31, 2007 in accordance with the requirements of the 1992 Edition through the 1992 addenda of the American Society of Mechanical Engineers, *Boiler and Pressure Vessel* Code (ASME Code), Section XI, Subsection IWL. The second 10-year interval IWL examinations will be performed in accordance with the 2001 Edition through 2003 Addenda of ASME Section XI. The next two 5-year IWL interval dates have been scheduled for August 31, 2011 and August 31, 2016.

In Reference 1, the licensee stated that the first 10-year interval (October 20, 1997 to May 21, 2008) IWE examinations of containment liner and penetration liner were performed in accordance with the 1992 Edition with the 1992 Addenda of ASME Code Section XI, Subsection IWE. In response to the NRC staff's RAI, the licensee stated that the Surry 2 IWE examinations of the containment liner performed in the first 10-year interval did not identify any areas that required augmented Category E-C examinations. The licensee also stated that the second 10-year interval IWE examinations will be performed in accordance with the 2001 Edition through 2003 Addenda of ASME Section XI as modified by the 10 CFR 50.55a(b).

Under its CISI program, as required by 10 CFR 50.55a(b)(2)(viii)(E) and 10 CFR 50.55a(b)(2)(ix)(A), the licensee evaluates the acceptability of inaccessible areas of the containment structure and metallic liner if conditions exist in the accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas. In response to the NRC staff's RAI, the licensee stated that based on the conditions found during examination of accessible areas, no evaluation of inaccessible areas was required.

In response to the NRC staff's RAI relative to the inspection of containment moisture barrier between the containment wall liner and containment concrete floor, the licensee responded that the Surry 2 containment design does not include a moisture barrier at the interface of the containment wall liner and concrete floor. The licensee stated that visual inspection of the Surry 2 containment liner in 2000 revealed that the majority (97 percent) of the containment perimeter at the interface of the wall liner and the concrete floor was free from defects, or had only minor surface cracks and blister of the coating system. The remaining 3 percent of the perimeter had longer cracks, approximately 6 to 18 inches, in the coating system. There was no indication of distress in the concrete and no moisture was evident in or around the interface joint. The licensee also stated that, in the 2003, Fall outage, further visual inspection of the interface joint at the containment liner and the concrete floor revealed that the small gap at this interface was filled with silicone caulk around the entire perimeter of the containment. After visual inspection of the existing caulk and the liner area behind the caulk, it was concluded that the liner was in good condition with only minor surface rust behind the caulk. The licensee later removed the caulk and approximately 2 inches by 2 inches of concrete at the interface joint around the perimeter of the containment. Subsequently, the liner area was cleaned and coated and the concrete floor was restored.

The licensee stated that in the 2002 IWL inspections of containment concrete surfaces, embedded wood (approximately 2 inches by 12 inches) pieces were identified in the containment dome area. The licensee evaluated the condition and concluded that the containment structural integrity and the liner plate were not adversely affected. The concrete repair was performed in accordance with the ISI and maintenance programs, as applicable. During the 2006 IWL inspection, the licensee discovered other embedded materials in the containment dome area and identified areas of minor surface defects (e.g., pop outs, abandoned anchors, small hole) on the exterior surface of the containment. All identified embedded debris was located above the first layer of reinforcing steel within the concrete cover. After removal of all debris and abandoned items, the concrete void areas were repaired in accordance with the ISI and maintenance programs, as applicable.

3.1.4 Containment Integrity Summary

On the basis of its review of the information provided in the licensee's application and responses to the NRC staff's RAI, the NRC staff finds that: (1) the results of the past ILRT demonstrate that the leak-tight integrity of the containment structure has been adequately managed; (2) the structural integrity of the containment vessel is verified through periodic ISIs conducted as required by Subsections IWE and IWL of the ASME Code, Section XI; (3) the integrity of the penetrations and containment isolation valves are periodically verified through Type B and Type C tests as required by 10 CFR Part 50, Appendix J, and the Surry 2 TSs; (4) the licensee is employing a CISI program that requires evaluation of any potential degradation of accessible and inaccessible areas of the containments, and (5) the containment liner protective coating is inspected visually every refueling outage and repair of any identified damage is adequately managed. Based on these findings, the NRC staff concludes that the licensee has an adequate ISI program and procedures in place to examine, monitor and correct potential age-related and environmental degradations of the pressure retaining components of the Surry 2 containment structure.

3.2 Risk Assessment

The licensee performed a risk impact assessment of extending the Type A test interval to 15 years. Additional analysis and information was provided by the licensee in a letter dated June 27, 2008 (Reference 7). In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in the August 1994, Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing", the NEI Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Surveillance Intervals, and RG 1.174.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, 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 TR-104285 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 initially in effect for Surry 2 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 months to 60 months. Since Type A tests only detect about 3 percent of leaks (others 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 and the NEI Interim Guidance, the licenseeassessed 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. After the Option B rulemaking was completed in 1995, the NRC staff issued RG 1.174 on the use of probabilistic risk assessment (PRA) 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 provides risk-acceptance guidelines for assessing the increases in core damage frequency (CDF) and large early release frequency (LERF) for risk-informed license amendment requests. 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 based on 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 how it meets the defense-in-depth philosophy.

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. These bound 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 licensee analyses 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 less than 0.1 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 3.2×10^{-7} per year based on the plant-specific internal events PRA, and 6.2×10^{-7} per year when external events are included. 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 ASME 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 estimated to be less than 1×10^{-8} per year.

   Pursuant to RG 1.174, 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. Based on information provided by the licensee, the total LERF for internal and external events, including the requested change, is about 1.5×10^{-6} per year, which meets the total LERF criteria. The NRC 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 allows 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. Based on application of the NEI Interim Guidance, the maximum increase in the conditional containment failure probability for the cumulative change of going from a test frequency of three in 10 years to one in 15 years would be approximately one percentage point. The NRC 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.

Based on the above review, the NRC staff finds that the increase in predicted risk impact due to the proposed change is within the acceptance guidelines of RG 1.174, and maintains the defense-in-depth philosophy. Therefore, the proposed change is acceptable from a risk perspective.

3.3 Summary

Based on the technical evaluation above, the NRC staff finds the licensee's proposed one-time extension of the ILRT interval from 10 to 15 years, for Surry 2, acceptable. The Type A test shall be performed no later than October 26, 2015.

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 comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (73 FR 2551). 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from William R. Matthews, Virginia Electric and Power Company (Dominion), to NRC Document Control Desk, "Surry Power Station Unit 2 Proposed License Amendment Request, One-Time Five-Year Extension to Type A Test Interval", dated December 17, 2007 (ML073511682).

2. Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," September 1995 (ML003740058).

3. NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995.

4. Letter form Gerald T. Bischof, Virginia Electric and Power Company (Dominion), to NRC Document Control Desk, "Surry Power Station Unit 2, Response to Request for Additional Information Regarding Proposed License Amendment Request, One-Time Five-Year Extension to Type A Test Interval", dated April 24, 2008 (ML081200869).

5. ANSI/ANS-56.8-1994, "Containment System Leakage Testing Requirements."

6. Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis," November 2002 (ML023240437).

7. Letter form Gerald T. Bischof, Virginia Electric and Power Company (Dominion), to NRC Document Control Desk, "Surry Power Station Unit 2, Response to Request for Additional Information Regarding Proposed License Amendment Request, One-Time Five-Year Extension to Type A Test Interval," dated June 27, 2008 (ML081790666).

Principal Contributors: Hans Ashar, NRR
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Date: December 18, 2008

Mr. David A. Christian
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION, UNIT NO. 2, ISSUANCE OF AMENDMENT REGARDING THE REVISION OF TECHNICAL SPECIFICATION 4.4, PERTAINING TO THE CONTAINMENT LEAKAGE RATE TESTING PROGRAM (TAC NO. MD7535)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 263 to Renewed Facility Operating License No. DPR-37 for Surry Power Station, Unit No. 2. The amendment changes the Technical Specifications (TSs) in response to your application dated December 17, 2007, as supplemented by letters dated April 24, 2008, and June 27, 2008.

The amendment revises TS 4.4, pertaining to the containment leak rate testing program. The TS change would permit a one-time 5-year extension to the once per 10- year frequency of the performance-based leakage rate testing program for Type A tests, which are done in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program." This one time exception to the RG 1.163 requirement would allow the next Type A test to be performed no later than October 26, 2015.

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/
John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-281

Enclosures:
1. Amendment No. 263 to DPR-37
2. Safety Evaluation

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Amendment No.: ML083180243 * SE Input provided by memo dated **SE Input provided by memo dated

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