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U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: VIRGIL C. SUMMER NUCLEAR STATION (VCSNS) UNIT 1
DOCKET NO. 50-395
OPERATING LICENSE NO. NPF-12
LICENSE AMENDMENT REQUEST - LAR (13-00705)
Technical Specification Change to Extend
Integrated Leak Rate Test Frequency to 15 Years

In accordance with the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, requests Nuclear Regulatory Commission (NRC) review and approval to amend Operating License NPF-12 for Virgil C. Summer Nuclear Station (VCSNS). The proposed change would allow for the extension of the approximate 130-months (10.9 years) frequency of the VCSNS containment leakage rate test (e.g., Integrated Leak Rate Test [ILRT] or Type A test) that is required by Technical Specification (TS) 6.8.4(g) to 15 years on a permanent basis.

This proposed change has been evaluated in accordance with 10 CFR 50.91 (a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that this change involves no significant hazards consideration. The bases for this determination are included in Attachment I, which provides an analysis of proposed TS change. Attachment II provides the revised TS pages reflecting the proposed changes. Attachment III provides the annotated TS pages showing the proposed changes. Attachment IV is the Post Outage Appendix J Activity Report Type B and Type C Containment Penetration Leakage Assessment Record. Attachment V is a summary table of LLRT results of those containment penetrations (including their test schedule intervals) that have not demonstrated acceptable performance history in accordance with the Containment Leakage Rate Program with a discussion of the causes and corrective actions taken. Attachment VI provides an assessment on the risk impact of extending the ILRT interval. Attachment VII provides a commitment list.

SCE&G requests approval of the proposed amendment by April 4, 2014. Once approved, the amendment shall be implemented within 60 days.

A001
NRR

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated South Carolina Official.

This proposed change has been reviewed and approved by both the VCSNS Plant Safety Review Committee and the VCSNS Nuclear Safety Review Committee.

The proposed change includes three new commitments. There are no revisions to existing commitments. The new commitments are summarized in Attachment VII.

If you have any questions or require additional information, please contact Bruce Thompson at (803) 931-5042.

I certify under penalty of perjury that the information contained herein is true and correct.

4/3/13
Executed on

Thomas D. Gatlin

BJQ/TDG/wm

Attachments:
- I. Analysis of Proposed Technical Specification Change
- II. Marked Up TS Page
- III. Retyped TS Page
- IV. Post Outage Appendix J Activity Report Type B and Type C Containment Penetration Leakage Assessment Record.
- V. Summary Table of LLRT Results Not Demonstrating Acceptable Performance History
- VI. SCE&G ILRT Interval Extension Risk Analysis
- VII. Commitment Page

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ATTACHMENT I

Analysis of Proposed Technical Specification Change

Technical Specification Change to Integrated Leak Rate Test Interval Permanent 15-Year

1.0 DESCRIPTION

This letter is a request to amend Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station (VCSNS), Unit 1.

The proposed change would allow for the extension of the 130-month frequency of the VCSNS containment leakage rate test (e.g., Integrated Leak Rate Test [ILRT] or Type A test) that is required by Technical Specification (TS) 6.8.4(g) to 15 years on a permanent basis. With the approval of the proposed change, the existing ILRT frequency would be revised from 10.9 years to 15 years.

The technical analysis for the proposed license amendment is based on risk related and non-risk related considerations. A risk analysis performed by SCE&G concluded that the increases in estimated person-rem population dose and large early release frequency (LERF) are consistent with the guidance provided in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.174, NUREG-1493, and EPRI Report No. 1009325, Revision 2-A. The technical analysis provides the basis for the determination that the proposed amendment does not involve a significant hazards consideration as described in 10 CFR 50.92.

A one-time 15-month extension to the ten-year ILRT frequency was approved, by the NRC, in license amendment no. 189, issued May 1, 2012. With the current 15-month extension, the next ILRT would be performed during the Spring of 2014 refueling outage.

The proposed amendment would allow the next ILRT for VCSNS to be performed within 15 years from the last ILRT (i.e., by October 15, 2018), as opposed to the current interval. The change would allow successive ILRTs to be performed at 15-year intervals (assuming acceptable performance history). The performance of fewer ILRTs would result in significant savings in radiation exposure to personnel, cost, and critical path time during future refueling outages.

2.0 PROPOSED CHANGE

VCSNS TS 6.8.4(g), "Containment Leakage Rate Testing Program," states,

> "A program shall be established to implement leakage rate testing of the containment system as required by 10 CFR 50.54(o) and 10 CFR 50. Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995; NEI 94-01, "Industry Guideline for Performance-Based Option of 10 CFR 50, Appendix J," Revision 0; ANSI/ANS-56.8-1994, "Containment System Leakage Testing Requirements"; as modified by approved exceptions that the next Type A test performed after the October 15, 2003 Type A test shall be performed no later than August 15, 2014."

The proposed amendment would utilize the latest revisions of the applicable NEI and ANSI/ANS standards and would revise VCSNS TS 6.8.4(g), "Containment Leakage Rate Testing Program," to state,

> "A program shall be established to implement leakage rate testing of the containment system as required by 10 CFR 50.54(o) and 10 CFR 50. Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995; NEI 94-01, "Industry Guideline for Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, July 2012; ANSI/ANS-56.8-2002, "Containment System Leakage Testing Requirements"; as modified by approved exceptions that the next Type A test performed after the October 15, 2003 Type A test shall be performed no later than October 15, 2018."

Revision 3-A of NEI 94-01 describes an approach for implementing the optional performance-based requirements of Option B, including provisions for extending primary containment integrated leak rate test (ILRT) intervals to 15 years, and incorporates the regulatory positions stated in RG 1.163. In the safety evaluation (SE) issued by NRC letter dated June 8, 2012, the NRC concluded that NEI 94-01, Revision 3, describes an acceptable approach for implementing the optional performance-based requirements of Option B of 10 CFR 50, Appendix J, and found that NEI 94-01, Revision 3, is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SE.

In accordance with the guidance in NEI 94-01, Revision 3-A, SCE&G proposes to extend the interval for the primary containment ILRT to no longer than 15 years from the last ILRT. The last ILRT was completed on October 15, 2003. The ILRT is currently required to be performed at a ten years and 10 months interval and is due no later than August 15, 2014, as required by TS 6.8.4(g). Using the proposed interval of no longer than 15 years, the next ILRT will be due no later than October 15, 2018.

Attachment II contains the existing TS page 6-12b marked-up to show the proposed changes to TS 6.8.4(g).

3.0 BACKGROUND

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the TS, and that periodic surveillance of containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment and the systems and components penetrating containment. The limitation on containment leakage provides assurance that the containment would perform its design function. Appendix J identifies three types of required tests: (1) Type A tests, intended to measure the containment overall integrated leakage rate; (2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for containment penetrations; and (3) Type C tests, intended to measure containment isolation valve leakage. Type B and C tests identify the vast majority of potential containment leakage paths. Type A tests identify the overall (integrated) containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those structural parts of the containment not covered by Type B and C testing.

The purpose of the tests are to assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the Technical Specifications (TS) or associated bases; and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment.

In 1995, the NRC amended 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to provide a performance-based Option B for the containment leakage testing requirements. Option B requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Performance-based test intervals are based on consideration of the operating history of the component and resulting risk from its failure. The amendment is also aimed at improving the focus of the body of regulations by eliminating prescriptive requirements that are marginal to safety and by providing licensees greater flexibility for cost-effective implementation methods for regulatory safety objectives.

Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, was developed as a method acceptable to the staff for implementing Option B. This RG states that the Nuclear Energy Institute (NEI) guidance document, NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the staff for complying with Option B. RG 1.163 specifies an extension in Type A frequency to at least one test in ten years based upon two consecutive successful tests and risk history. This relaxation was based on an NRC risk assessment contained in NUREG-1493,"Performance-Based Containment Leak-Test Program,"

and Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," both of which illustrated that the risk increase associated with extending the ILRT surveillance interval was very small.
By letter dated April 16, 1996, South Carolina Electric & Gas (SCE&G) submitted a TS change request concerning the implementation of 10 CFR 50, Appendix J, Option B. In the Safety Evaluation (SE) approving this request (letter dated October 2, 1996), the NRC noted the proposed TS changes are in compliance with the requirements of 10 CFR 50, Appendix J, Option B, and are consistent with the guidance in RG 1.163.

With the approval of the TS change request, VCSNS transitioned to a performance based ten year frequency for the Type A tests. SCE&G submitted a LAR to extend the ILRT interval from ten years (120 months) to approximately 130 months via letter dated October 25, 2011. This one-time extension was approved by the NRC, as license amendment no. 189, issued May 1, 2012.

By letters dated August 31, 2007 and June 9, 2011, EPRI Report No. 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," and NEI document NEI 94-01, Revision 3, were submitted to the NRC Staff for review.

EPRI Report No. 1009325, Revision 2, provides a risk impact assessment for optimized ILRT intervals of up to 15 years, using current industry performance data and risk informed guidance, primarily Revision 1 of RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The NRC's final SE issued by letter dated June 25, 2008, documents the evaluation and acceptance of EPRI Report No. 1009325, Revision 2, subject to the specific limitations and conditions listed in Section 4.2 of the SE. An accepted version of EPRI Report No. 1009325 has subsequently been issued as Revision 2-A (also identified as Technical Report TR-1018243) dated October 2008.

NEI 94-01, Revision 3-A, describes an approach for implementing the optional performance-based requirements of Option B, which includes provisions for extending Type A intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163. It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. This method uses industry performance data, plant-specific performance data, and risk insights in determining the appropriate testing frequency. NEI 94-01, Revision 3-A, also discusses the performance factors that licensees must consider in determining test intervals. However, the NEI guideline does not address how to perform the tests because these details are included in other industry documents (e.g., American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-2002). The NRC final SE issued by letter dated June 8, 2012, documents the evaluation and acceptance of NEI 94-01, Revision 3-A, subject to the specific limitations and conditions listed in Section 4.0 of the SE.

4.0 TECHNICAL ANALYSIS

Primary containment provides an essentially leak-tight barrier against the uncontrolled release of radioactivity into the environment following a design basis accident. The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the primary containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the TS.

The proposed change to extend the ILRT surveillance interval through the end of the RF-24 refueling outage is justified based on the results of previous ILRTs and containment inspection programs.

The below limitation and condition questions come from section 4.1, Limitations and Conditions for NEI TR 94-01 Revision 2.

Limitation/Condition (From Section 4.1 of SE)	SCE&G Response
1. For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002).	VCSNS uses the definition found in Section 5.0 of NEI 94-01, Revision 3-A, for calculating the Type A leakage rate (GTP-315 Step 4.1.4.A). NEI 94-01, Revision 3-A contains the same definition. (For proper and effective implementation of the Type B and Type C local leak rate testing program, please see the Post Outage Appendix J Activity Report Type B and Type C Containment Penetration Leakage Assessment Record Attachment IV.)
2. The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests.	A schedule of containment inspections is provided in Section 4.2 below.
3. The licensee addresses the areas of the containment structure potentially subjected to degradation.	General visual examination of accessible interior and exterior surfaces of the containment system for structural problems is conducted in accordance with the VCSNS Inservice Inspection Plan which implements the requirements of the ASME, Section XI, Subsections IWE and IWL, as required by 10 CFR 50.55a(g). A summary table of LLRT results of those containment penetrations (including their test schedule intervals) that have not demonstrated acceptable performance history in accordance with the Containment Leakage Rate Program with a discussion of the causes and corrective actions taken are listed in Attachment V.

4. The licensee addresses any test and inspections performed following major modifications to the containment structure, as applicable.	VCSNS replaced the steam generators in RF8 (1994). No modifications to the containment structure were required.
5. The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provisions of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition.	VCSNS acknowledges and accepts this NRC staff position, as communicated to the nuclear industry in Regulatory Issue Summary (RIS) 2008-27 dated December 8, 2008.
6. For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data.	Not applicable. VCSNS is not licensed pursuant to 10 CFR Part 52.

4.1 Previous ILRT Results

The two most recent ILRT test results confirm that the VCSNS containment structure leakage is acceptable, demonstrating considerable margin with respect to the TS acceptance criterion of 0.20% of containment air weight per 24 hours at the design basis loss of coolant accident pressure.

The performance leakage rates are calculated in accordance with NEI 94-01, Section 9.1.1. The performance leakage rate includes the Type A Upper Confidence Limit (UCL) at 95% plus the as-left minimum pathway leakage rate for all Type B and C pathways not in service, isolated, or not lined up in their test position. In addition, leakage pathways that were isolated during the performance of the test because of excessive leakage are included in the test results by adding the as-found minimum pathway leakage rate to the Type A test 95% UCL. The performance leakage rate does not include leakage savings (i.e., improvements to Type B and C components made prior to the Type A test).

The VCSNS April 1993 periodic Type A test using the Mass Point method calculated at the 95% UCL resulted in a leakage rate of 0.1298 %wt / day. The minimum pathway leakage rate for Type B and C pathways not in service (considering water level corrections) was 0.0070 %wt / day. Therefore, the performance leakage rate was 0.1298 + 0.0070 = 0.1368 %wt / day.

The VCSNS October 2003 periodic Type A test using the Total Time method calculated at the 95% UCL resulted in a leakage rate of 0.0581 %wt / day. The minimum pathway leakage rate for Type B and C pathways not in service and water level corrections was 0.0173 %wt / day. The Volume Change Correction factor is -0.00051 %wt / day. Therefore, the performance leakage rate was 0.0581+ 0.0173-.00051 = 0.0749 %wt / day.
These results show that there is considerable margin compared to the maximum allowable leakage rate for V. C. Summer of 0.20 %wt / day at a pressure of 45.1 psig. Based upon these two consecutive successful tests, and the approval of this License Amendment, the current ILRT interval requirement for VCSNS will be 15 years.

No modifications that require a Type A test are planned prior to RF-24, when the next Type A test will be performed under this proposed change. Any unplanned modifications to the containment prior to the next scheduled Type A test would be subject to the special testing requirements of Section IV.A of 10 CFR 50, Appendix J. There have been no pressure or temperature excursions in the containment which could have adversely affected containment integrity. There is no anticipated addition or removal of plant hardware within containment which could affect leak-tightness.

4.2 Containment Inspection Programs and Results

VCSNS has established procedures for performing visual examinations of the accessible surfaces of the containment for detection of structural problems. RG 1.163, Regulatory Position C.3 specifies that these examinations should be conducted prior to initiating a Type A test and during two other outages before the next Type A test if the interval for the Type A test has been extended to ten years, in order to allow for early detection of evidence of structural deterioration. These visual examinations have been completed, with no significant defects noted to date.

The ASME Section XI Program requires that the steel containment vessel be examined in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE, and associated modifications and limitations imposed by 10 CFR 50.55a(b)(2). Details of the containment inservice inspection program are described in the VCSNS Containment Inservice Inspection (CISI) found in Technical Specification Bases 3/4 6-2.

The 8th Period (30th year) IWL surveillance including the tendon pre-stress system and containment concrete inspections were completed during RF-19 in March-April 2011. The inspections were done in accordance with ASME Section XI and the results of the inspections confirm that there is no abnormal degradation of the containment and that the vertical, horizontal, and dome tendon groups are conservatively projected to maintain more than the required minimum average tendon force for the particular group until the next regularly scheduled surveillance. The VCSNS program to perform these surveillance examinations at a 5 year or less inspection interval continues to be an acceptable and appropriate program to monitor and maintain these structures to ensure their capability to perform the safety related design basis containment function.

The testing frequency for Type B and C tests is not affected by this requested amendment to permanently extend the Type A test interval from 130 months (10.9 years) to 15 year. The following provides an approximate schedule for the containment surface examinations assuming the Type A test frequency is extended to 15 years.

Table 4.2-1
Containment Inservice Inspection Periods (IWE/IWL)

Calendar Year/Refueling Outage	Type A Test (ILRT)	General Visual Examination of Accessible Exterior Surface	General Visual Examination of Accessible Interior (Liner) Surface
S2011/RF 19		X (Note 1)	
F2012/RF 20			X (Note 3)
S2014/RF 21			
F2015/RF 22		X (Note 1)	X (Note 3)
S2017/RF 23			
F2018/RF 24	X	X (Note 2)	X (Note 3)
S2020/RF 25		X (Note 1)	
F2021/RF 26			X (Note 3)
S2023/RF 27			
F2024/RF 28		X (Note 1)	X (Note 3)
S2026/RF 29			
F2027/RF 30			X (Note 3)
S2029/RF 31		X (Note 1)	
F2030/RF 32			X (Note 3)
S2032/RF 33			
F2033/RF 34	X	X (Note 1) X (Note 2)	X (Note 3)
S2035/RF 35			
F2036/RF 36			X (Note 3)
S2038/RF 37		X (Note 1)	
F2039/RF 38			X (Note 3)
S2041/RF 39			
F2042/RF 40		X (Note 1)	X (Note 3)
S2044/RF 41			
F2045/RF 42			X (Note 3)
S2047/RF 43		X (Note 1)	
F2048/RF 44	X	X (Note 2)	X (Note 3)

NOTES:

1. IWL pre-stress tendon surveillance and IWL concrete exterior inspection are done on a 54 month interval corresponding to every third refueling outage.
2. General containment visual inspections are conducted in accordance with General Test Procedure per 10CFR50 App. J prior to performing the Type A ILRT.
3. STP 207.002, "Inspection of Containment," or an equivalent inspection, is performed every other refueling outage (36 month frequency). This inspection verifies the structural integrity of the exposed accessible interior and exterior surfaces of the containment by a visual examination.

4.3 Supplemental Inspection Requirements

General visual examination of accessible interior and exterior surfaces of the containment system for structural problems are conducted in accordance with the VCSNS Inservice Inspection Plan which implements the requirements of the ASME, Section XI, Subsections IWE and IWL, as required by 10 CFR 50.55a(g).

4.4 Deficiencies Identified

A summary table of LLRT results of those containment penetrations (including their test schedule intervals) that have not demonstrated acceptable performance history in accordance with the Containment Leakage Rate Program with a discussion of the causes and corrective actions taken are listed in Attachment V.

4.5 Plant-Specific Confirmatory Analysis

4.5.1 Methodology

An evaluation has been performed to assess the risk impact of extending the VCSNS ILRT interval from the current ten years to fifteen years. This plant-specific risk assessment follows the guidance in NEI 94-01, Revision 2-A, the methodology described in EPRI TR-1009325, Revision 2-A and the NRC regulatory guidance outlined in RG 1.174 on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change plant licensing bases. In addition, the methodology used for Calvert Cliffs Nuclear Power Plant to estimate the likelihood and risk implication of corrosion-induced leakage of steel containment liners going undetected during the extended ILRT interval was also used in the analysis.

The VCSNS Level 1/simplified Level 2 internal events PRA model was used to perform the plant-specific risk assessment. The analysis includes evaluation for the dominant external events using information from the VCSNS Individual Plant Examination of External Events (IPEEE). The VCSNS IPEEE event models have not been updated since the original IPEEE. While a Fire PRA model has been developed to support transition to NFPA 805, this Fire PRA model corresponds to the post-transition plant and does not match the current licensing basis for fire response. While the IPEEE models have not been updated, they include pertinent information and insights and have, therefore, been used to estimate the effect on total LERF of including external events in the ILRT interval extension risk assessment.

The below limitation and condition questions come from section 4.2 Limitations and Conditions for EPRI Report No. 1009325, Revision 2.

Limitation/Condition (From Section 4.2 of SE)	SCE&G Response
1. The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension.	Refer to Section 4.5.2.
2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small, and consistent with the clarification provided in Section 3.2.4.5 of this SE. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in CCFP should be defined as a value marginally greater than that accepted in a previous one-time ILRT extension request. This would require that the increase in CCFP be less than or equal to 1.5 percentage points.	EPRI Report No. 1009325, Revision 2-A, incorporates these population dose and Conditional Containment Failure Probability (CCFP) acceptance guidelines and these guidelines have been used for the VCSNS Unit 1 plant specific risk assessment.
3. The methodology in EPRI Report No. 1009325 Revision 2, is acceptable except for the calculation of the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate for the pre-existing containment large leak accident case (accident case 3b) used by the licensees shall be 100 La instead of 35 La.	EPRI Report No. 1009325, Revision 2-A, incorporated the use of 100 La as the average leak rate for the pre-existing containment large leakage rate accident case (accident case 3b), and this value has been used in the VCSNS plant specific risk assessment.
4. A LAR is required in instances where containment over-pressure is relied upon for ECCS performance.	VCSNS does not credit containment overpressure for the mitigation of design basis accidents as documented in the VCSNS plant specific risk assessment.

4.5.2 PRA Quality

The VCSNS Unit 1 Internal Events PRA is based on a detailed model of the plant developed from the Individual Plant Examination for Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities." The model is maintained and updated in accordance with VCSNS procedures, and has been updated to meet the ASME PRA Standard and Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

The model has been reviewed and assessed on several occasions. In August 2002, the VCSNS Internal Events PRA was peer reviewed in accordance with guidance in NEI 00-02, Industry PRA Peer Review Process. All A & B level Findings and Observations (F&O) from the WOG Internal Events PRA Peer Review have been addressed. Although all C & D level findings have not been incorporated, all of the items that had the potential to significantly impact model results have been resolved. Following completion of sufficient work to address the Peer Review comments, a 2005 gap assessment of the VCSNS Internal Events PRA was performed to determine the scope of work required to ensure the VCSNS Internal Events PRA meets Regulatory Guide 1.200, Revision 1. The results of this review indicated that VCSNS had resolved most of the issues identified in the original peer review, but the review identified some F&Os that needed additional work, as well as several new issues. Additionally (in this 2005 review) the VCSNS PRA was found to meet Capability Categories(CC)-II or better for 211 of the 271 Supporting Requirements (SRs) from the ASME PRA Standard, but 45 of the elements were found to either not meet the requirement or to meet the requirements at a CC-I level. Following work at VCSNS to address the findings and to increase the capability category ratings of the elements that needed an upgrade to allow use of the model in risk informed applications, a focused review was performed as required by the ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" (and 2007 addenda ASME RA-Sc-2007, Appendix A). All SRs were judged to be CC-II or better, with the exception of thirteen SRs that were rated at CC-I based on the VCSNS simplified NUREG/CR-6595 (An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events) compliant Large Early Release Frequency (LERF) model. While these 13 SRs specifically define a simplified NUREG/CR-6595 LERF models as CC-I, it was noted that use of the NUREG model is an acceptable means of calculating LERF for applications. The conclusion of the 2007 focused review was that the model is of sufficient quality for use in risk-informed applications.

VCSNS has recently developed a simplified Level 2 PRA for the internal events model. The method used to develop the simplified Level 2 model is based on WCAP-16341-P "Simplified Level 2 Modeling Guidelines," which is an extension of NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," and includes realistic quantification of containment threats resulting from high pressure failure of the Reactor Vessel and additional detail on the treatment of Interfacing System LOCA and Induced Steam Generator Tube Rupture. The combined Level I/simplified Level 2 PRA for internal events was utilized in the risk analysis to support extending the VCSNS Containment Integrated Leak Rate Test interval to fifteen years.

Based on the above, the VCSNS PRA Model is acceptable for use in this License Amendment Request.

4.5.3 Summary of Plant-Specific Risk Assessment Results

Based on the results from the risk assessment and the associated sensitivity calculations, the following conclusions regarding the assessment of the plant risk are associated with permanently extending the Type A ILRT test frequency to once in fifteen years:

- Regulatory Guide 1.174 provides guidance for determining the risk impact of plant specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10E-06/yr and increases in LERF below 10E-07/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is conservatively estimated as 4.08E-08/yr. As such, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Regulatory Guide 1.174.

- Regulatory Guide 1.174 also states that when the calculated increase in LERF is in the range of 1.00E-07 per reactor year to 1.00E-06 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.00E-05 per reactor year. An additional assessment of the impact from External Events was also made. In this case, the total LERF including External Events was conservatively estimated as 1.21E-06/yr for VCSNS. This is below the RG 1.174 acceptance criteria for total LERF of 1.00E-05/yr and therefore this change satisfies both the incremental and absolute expectations with regard to the Regulatory Guide 1.174 LERF metric.

- The change in Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 8.76E-03 person-rem/yr. Note that this value is based on internal events only and does not consider external events. EPRI Report No. 1009325, Revision 2-A states that a very small population dose is defined as an increase of ≤ 1.0 person-rem per year or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This is consistent with the NRC Final Safety Evaluation for NEI 94-01 and EPRI Report No. 1009325. Moreover, the risk impact when compared to other severe accident risks is negligible.

- The increase in the conditional containment failure probability from the three in ten year interval to a permanent one time in fifteen year interval is 1.06%. EPRI Report No. 1009325, Revision 2-A states that increases in CCFP of ≤1.5 percentage points are very small. This is consistent with the NRC Final Safety Evaluation for NEI 94-01 and EPRI Report No. 1009325. Therefore this increase is judged to be very small.

Therefore, permanently increasing the ILRT interval to once in fifteen years is considered to be a very small change to the V.C. Summer Nuclear Station risk profile.

Previous Assessments

The NRC in NUREG-1493 has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from three per ten years to one per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.

- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage rate tests is possible with minimal impact on public risk. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment structure.

The findings for VCSNS confirm these general findings on a plant specific basis considering the severe accidents evaluated for VCSNS, the VCSNS containment failure modes, and the local population surrounding VCSNS.

4.6 Conclusion

NEI 94-01, Revision 3-A, describes an NRC-accepted approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years. NEI 94-01, Revision 3-A delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. SCE&G is proposing to adopt the guidance of NEI 94-01, Revision 3-A for the VCSNS 10 CFR 50, Appendix J, testing program plan.

Based on the previous ILRT tests conducted at VCSNS, it may be concluded that extension of the containment ILRT interval from 130-months to 15 years represents minimal risk to increased leakage. The risk is minimized by continued Type B and Type C testing performed in accordance with Option B and inspection activities performed as part of the VCSNS IWE/IWL ISI program.

This experience is supplemented by risk analysis studies, including the VCSNS risk analysis provided in Attachment VI. The findings of the VCSNS risk assessment confirm the general findings of previous studies, on a plant-specific basis, that extending the ILRT interval from ten to 15 years results in a small change to the VCSNS risk profile.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements / Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met. 10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J to 10 CFR 50, "Leakage Rate Testing of Containment of Water Cooled Nuclear Power Plants." Appendix J specifies containment leakage testing requirements, including the types of tests required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. In addition, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing and reporting requirements for each type of test.

As discussed earlier, RG 1.163 endorses NEI 94-01, Revision 3-A with certain modifications and additions.

The adoption of the Option B performance-based containment leakage rate testing for Type A testing did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, B, and C containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, test frequency is based upon an evaluation that reviews "as-found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The change to the Type A test frequency did not directly result in an increase in containment leakage.

EPRI TR-1009325, Revision 2, provides a risk impact assessment for optimized Integrated Leak Rate Test (ILRT) intervals up to 15 years, utilizing current industry performance data and risk informed guidance. NEI 94-01, Revision 3-A, states that a plant-specific risk impact assessment should be performed using the approach and methodology described in TR-1009325, Revision 2, for a proposed extension of the ILRT interval to 15 years. In the safety evaluation (SE) issued by NRC letter June 25, 2008, the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their TS to extend the ILRT surveillance interval to 15 years, subject to the limitations and conditions noted in Section 4.0 of the SE.

NEI 94-01, Revision 3-A, describes an approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. The document incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years. NEI 94-01, Revision 3-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate test frequencies. In the SE issued by NRC letter dated June 8, 2012, the NRC concluded that NEI 94-01, Revision 3, describes an acceptable approach for implementing the optional performance-based requirements of 10 CFR 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions, noted in Section 4.2 of the SE.

Based on the considerations above, (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 continue to be conducted in accordance with the site licensing basis, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

5.2 No Significant Hazards Consideration

A change is proposed to the Virgil C. Summer Unit 1 Technical Specifications to extend the Type A test required by TS 6.8.4(g) to a permanent 15-year interval. SCE&G has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as described below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed exemption involves a permanent 15-year extension to the current interval for Type A containment testing. The current test interval of 130 months (10.9 years) would be extended to a permanent 15-year frequency from the last Type A test. The proposed extension does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the reactor containment itself and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. Therefore, this proposed extension does not involve a significant increase in the probability of an accident previously evaluated nor does it create the possibility of a new or different kind of accident.

   The integrity of the reactor containment is subject to two types of failure mechanisms which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as configuration management and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the containment itself combined with the containment inspections performed in accordance with ASME, Section XI, the Maintenance Rule, and Licensing commitments serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by a Type A test. Based on the above, the proposed extension does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed revision to the TS involves a 15-year permanent extension to the current interval for Type A containment testing. The reactor containment and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed TS change does not involve a physical change to the plant or the manner in which the plant is operated or controlled. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change to the TS involves a 15-year permanent extension to the current interval for Type A containment testing. The proposed TS change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The specific requirements and conditions of the Primary Containment Leak Rate Testing Program, as defined in the TS, exist to ensure that the degree of reactor containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by TS is maintained. The proposed change involves only the extension of the interval between Type A containment leak rate tests. The proposed surveillance interval extension is bounded by the 15-year permanent extension currently authorized within NEI 94-01, Revision 3-A. Type B and C containment leak rate tests will continue to be performed at the frequency currently required by TS. Industry experience supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME, Section XI and the Maintenance Rule serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. The combination of these factors ensures that the margin of safety that is in plant safety analysis is maintained. The design, operation, testing methods and acceptance criteria for Type A, B, and C containment leakage tests specified in applicable codes and standards will continue to be met, with the acceptance of this proposed change, since these are not affected by changes to the Type A test interval. Therefore, the proposed TS change does not involve a significant reduction in a margin of safety.

Based on the above, SCE&G concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

6.0 ENVIRONMENTAL CONSIDERATIONS

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 PRECEDENCE

This request is similar in nature to the license amendment authorized by the NRC for Arkansas Nuclear One, Unit No. 2 on April 7, 2011 (ADAMS Accessions Number ML101680380), Palisades on April 6, 2011 (ADAMS Accessions Number ML110970616), and the Nine Mile Point Unit 2 on March 30, 2010 (ADAMS Accession Number ML100730032).

VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)

ATTACHMENT II

PROPOSED TECHNICAL SPECIFICATION CHANGE (MARK-UP)

f. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measures of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM;

2) A Land Use Census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of the census; and

3) Participation in an Inter-laboratory Comparison Program to ensure that independent checks on the precision and accuracy of measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

g. Containment Leakage Rate Testing Program

A program shall be established to implement leakage rate testing of the containment system as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995; NEI 94-01, "Industry Guideline for Performance-Based Option of 10 CFR 50, Appendix J," Revision ~~0,~~ 3-A, July 2012; ANSI/ANS-56.8-~~1994~~ 2002 "Containment System Leakage Testing Requirements"; as modified by approved exceptions that the next Type A test performed after the October 15, 2003 Type A test shall be performed no later than ~~August 15, 2014.~~ October 15, 2018.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 45.1 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.20 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)

ATTACHMENT III

PROPOSED TECHNICAL SPECIFICATION CHANGE (RETYPED)

Attachment to License Amendment No.XXX
To Facility Operating License No. NPF-12
Docket No. 50-395

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

Remove Page	Insert Page
6-12b	6-12b

f. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measures of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM;

2) A Land Use Census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of the census; and

3) Participation in an Inter-laboratory Comparison Program to ensure that independent checks on the precision and accuracy of measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

g. Containment Leakage Rate Testing Program

A program shall be established to implement leakage rate testing of the containment system as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995; NEI 94-01, "Industry Guideline for Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, July 2012; ANSI/ANS-56.8-2002, "Containment System Leakage Testing Requirements"; as modified by approved exceptions that the next Type A test performed after the October 15, 2003 Type A test shall be performed no later than October 15, 2018.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 45.1 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.20 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)

ATTACHMENT IV

Post Outage Appendix J Activity Report Type B and Type C Containment Penetration Leakage Assessment Record

VCSNS Type B and C Frequencies

Equipment ID	Description	Penetration Number	Type	Interval (Months)	Comments
FUEL-XFERTUBE	Fuel Transfer Tube	XRP0107	Flange	18	Manually set to 18 months to meet every outage commitment
HATCH-EQUIP	Containment Equipment Hatch	HATCH-01	Hatch	18	Manually set to 18 months to meet every outage commitment
HATCH-ESCAPE	Personnel Escape Airlock	HATCH-02	Hatch	30	Manually set to 30 months to meet GTP-315 requirements
HATCH-PERSON	Personnel Airlock	HATCH-03	Hatch	30	Manually set to 30 months to meet GTP-315 requirements
MVB-3106A-SW	SWBP A Discharge	XRP0304	Check	60	
MVB-3106B-SW	SWBP B Discharge	XRP0403	Check	60	
MVB-3110A-SW	Ind Clg To RBCU 64A/65A	XRP0304	Check	60	
MVB-3110B-SW	Ind Cooling To RBCU 64B/65B	XRP0403	Check	30	Manually set to base interval Post LLRT exceeded Surveillance limit (RF-20) CR-12-05319
MVG-3003A-SP	Spray Header Isolation Loop A	XRP0401	Gate	60	
MVG-3003B-SP	Spray Header Isol Loop B	XRP0303	Gate	60	
MVG-3004A-SP	Sump Isolation Loop A	XRP0327	Gate	30	Manually set to base interval, Post LLRT exceeded Surveillance limit (RF-19) CR-11-02700
MVG-3004B-SP	Sump Isolation Loop B	XRP0328	Gate	60	
MVG-3103A-SW	RBCU 64A/65A Out Hdr Isol	XRP0305	Gate	60	
MVG-3103B-SW	RBCU 64B/65B Out Hdr Isol	XRP0102	Gate	60	
MVG-6797-FS	Fire Service Cntmt Isol	XRP0427	Gate	60	
MVG-7501-AC	TO CRDM Cooler Isolation	XRP0208	Gate	60	
MVG-7502-AC	TO CRDM Cooler Isolation (IRB)	XRP0208	Gate	0	Auto set to base interval, Pre LLRT exceeded Surveillance limit (RF-20) CR-12-04732
MVG-7503-AC	From CRDM Cooler Isol (IRB)	XRP0209	Gate	60	
MVG-7504-AC	From CRDM Cooler Isol (ORB)	XRP0209	Gate	30	Manually set to base interval, could not perform LLRT (RF-19) CR-11-02192
MVG-8107-CS	Charging Line Isolation	XRP0409	Gate	60	
MVG-8701A-RH	RCS Loop A To Pump A	XRP0316	Gate	60	30 Day Water Seal
MVG-8701B-RH	RCS Loop C To Pump B	XRP0226	Gate	0	(Auto set to base interval) Failed PRE LLRT (RF-20) CR-12-04956 (30 Day Water Seal)
MVG-8801A-SI	Hi Head To Cold Leg Injection	XRP0426	Gate	0	Removed from program (LAR-11-04821)
MVG-8801B-SI	Hi Head To Cold Leg Injection	XRP0426	Gate	0	Removed from program (LAR-11-04821)
MVG-8811A-SI	RHR Sump A to RHR Pump A	XRP0329	Gate	30	Manually set to base interval Pre LLRT not performed (RF-20) CR-12-12-04858
MVG-8811B-SI	RHR Sump B to RHR Pump B	XRP0425	Gate	30	Manually set to base interval, PRE LLRT not performed (RF-20) CR-12-04825
MVG-8884-SI	Charging Loop A To Hot Legs	XRP0415	Gate	0	Removed from program (LAR-11-04821)
MVG-8885-SI	Charging Loop A To Cold Legs	XRP0222	Gate	0	Removed from program (LAR-11-04821)
MVG-8886-SI	Charging Loop B To Hot Legs	XRP0412	Gate	0	Removed from program (LAR-11-04821)
MVG-8888A-SI	RHR LP A to Cold Legs	XRP0322	Gate	60	
MVG-8888B-SI	RHR LP B to Cold Legs	XRP0227	Gate	60	
MVG-8889-SI	RHR LP A&B to Hot Legs	XRP0325	Gate	60	

VCSNS Type B and C Frequencies

Equipment ID	Description	Penetration Number	Type	Interval (Months)	Comments
MVG-9568-CC	TO RB Load	XRP0312	Gate	30	Manually set to base interval, Pre LLRT not performed (RF-20) CR-12-04859
MVG-9600-CC	TO Thermal Barrier Isolation	XRP0204	Gate	60	
MVG-9605-CC	From RB Load Isolation (IRB)	XRP0330	Gate	60	
MVG-9606-CC	From RB Load Isolation (ORB)	XRP0330	Gate	60	
MVT-8100-CS	Seal Water Rtn Isol	XRP0410		60	
MVT-8102A-CS	A Seal Water Injection Isol	XRP0408		60	
MVT-8102B-CS	B Seal Water Injection Isol	XRP0229		60	
MVT-8102C-CS	C Seal Water Injection Isol	XRP0221		60	
MVT-8112-CS	Seal Water Rtn Isol	XRP0410		60	
NOZZLE: 0211	RB Leak Rate Test Blowdown	XRP0211	Flange	60	
NOZZLE: 0212	RB Leak Rate Test Blowdown	XRP0212	Flange	60	
NOZZLE: 0216	RB Leak Rate Test Pressurize	XRP0216	Flange	60	
NOZZLE: 0327	Prot Chamb RB Recirc Smp 3004A	XSM0004A	Flange	60	
NOZZLE: 0328	Prot Chamb RB Recirc Smp 3004B	XSM0004B	Flange	0	Auto set to base interval, exceeded Surveillance limit (RF-19) CR-11-01932
NOZZLE: 0329	Prot Chamb RB Recirc Smp 8811A	XSM0005A	Flange	60	
NOZZLE: 0425	Prot Chamb RB Recirc Smp 8811B	XSM0005B	Flange	60	
NOZZLE: 0501	Channel B	XRP0007	O-Ring Seal	60	
NOZZLE: 0502	Electrical Penetration	XRP0052	O-Ring Seal	60	
NOZZLE: 0503	Misc. 480V Power	XRP0016	O-Ring Seal	60	
NOZZLE: 0504	Reactor Coolant Pump C	XRP0001	O-Ring Seal	60	
NOZZLE: 0601	Reactor Coolant Pump B	XRP0003	O-Ring Seal	60	
NOZZLE: 0604	Instrument Group 3	XRP0048	O-Ring Seal	60	
NOZZLE: 0605	Instrument Channel 1	XRP0043	O-Ring Seal	60	
NOZZLE: 0606	Instrument Channel III	XRP0044	O-Ring Seal	60	
NOZZLE: 0607	Instrument Misc.	XRP0042	O-Ring Seal	60	
NOZZLE: 0700	Control Rod Drive Power	XRP0023	O-Ring Seal	60	
NOZZLE: 0701	Control Rod Drive Power	XRP0024	O-Ring Seal	60	
NOZZLE: 0702	Instrument Group 1	XRP0051	O-Ring Seal	60	
NOZZLE: 0704	MPWR	XRP0017	O-Ring Seal	60	
NOZZLE: 0705	MPWR/Control (Channel A)	XRP0028	O-Ring Seal	60	
NOZZLE: 0706	Control Rod Drive Power	XRP0054	O-Ring Seal	60	
NOZZLE: 0707	Control Rod Drive Power	XRP0025	O-Ring Seal	60	
NOZZLE: 0708	Control Rod Drive Power	XRP0026	O-Ring Seal	60	
NOZZLE: 0710	Reactor Coolant Pump A	XRP0006	O-Ring Seal	60	
NOZZLE: 0711	Ctrl Rod Pos. & Incore Temp.	XRP0037	O-Ring Seal	60	
NOZZLE: 0712	Incore Detector Drive	XRP0031	O-Ring Seal	60	
NOZZLE: 0714	Incore Detector Position Ind.	XRP0032	O-Ring Seal	60	
NOZZLE: 0715	Instrument Group 4	XRP0036	O-Ring Seal	60	
NOZZLE: 0716	Instrument Misc.	XRP0039	O-Ring Seal	60	
NOZZLE: 0717	Misc. 480V Power	XRP0019	O-Ring Seal	60	
NOZZLE: 0718	Instrument Misc.	XRP0040	O-Ring Seal	60	
NOZZLE: 0720	Thermocouple Elect. Pen.	XRP0104	O-Ring Seal	60	Pressurize with dry nitrogen only.
NOZZLE: 0721	Instrument Channel II	XRP0045	O-Ring Seal	60	
NOZZLE: 0722	Misc. 480V Power	XRP0018	O-Ring Seal	60	
NOZZLE: 0723	MPWR/Control (Channel B)	XRP0030	O-Ring Seal	60	
NOZZLE: 0725	Misc. Control & Comm.	XRP0033	O-Ring Seal	60	
NOZZLE: 0726	Channel A	XRP0010	O-Ring Seal	60	
NOZZLE: 0727	Channel A	XRP0011	O-Ring Seal	60	
NOZZLE: 0728	Control Rod Drive Power	XRP0055	O-Ring Seal	60	
NOZZLE: 0800	Instrument Group 2	XRP0035	O-Ring Seal	60	
NOZZLE: 0802	Electrical Penetration	XRP0053	O-Ring Seal	60	
NOZZLE: 0803	Control Rod Drive Power	XRP0056	O-Ring Seal	60	

VCSNS Type B and C Frequencies

Equipment ID	Description	Penetration Number	Type	Interval (Months)	Comments
NOZZLE: 0804	Control Rod Drive Power	XRP0057	O-Ring Seal	60	
NOZZLE: 0805	RB Polar Crane 480V Panel	XRP0015	O-Ring Seal	60	
NOZZLE: 0806	Thermocouple Elect. Pen.	XRP0106	O-Ring Seal	60	Pressurize with dry nitrogen only.
NOZZLE: 0808	Channel B	XRP0014	O-Ring Seal	60	
NOZZLE: 0809	Presurizer Heaters	XRP0020	O-Ring Seal	60	
NOZZLE: 0810	Presurizer Heaters	XRP0021	O-Ring Seal	60	
NOZZLE: 0812	Misc. Control & Comm.	XRP0034	O-Ring Seal	60	
NOZZLE: 0814	Instrument Channel IV	XRP0046	O-Ring Seal	60	
NOZZLE: 0815	Presurizer Heaters	XRP0022	O-Ring Seal	60	
NOZZLE: 505-18	Outage Penetration	XRP0505	Flange	18	Manually set to perform every outage (outage pen)
NOZZLE: 600-12	Outage Penetration	XRP0600	Flange	18	Manually set to perform every outage (outage pen)
NOZZLE: 602-12	Outage Penetration	XRP0602	Flange	18	Manually set to perform every outage (outage pen)
PVA-9311A-SS	Inlet To RM-A2 (IRB)	XRP0407A		60	
PVA-9311B-SS	Inlet To RM-A2 (ORB)	XRP0407A		60	
PVA-9312A-SS	RM-A2 To RB (IRB)	XRP0407B		60	
PVA-9312B-SS	RM-A2 To RB (ORB)	XRP0407B		60	
PVD-6242A-ND	RB Sumps Discharge Hdr Drn Vlv	XRP0424		60	
PVD-6242B-ND	RB Sumps Discharge Hdr Drn Vlv	XRP0424		60	
PVG-6056-HR	Alt Pur Supp Isol Vlv	XRP0103		60	
PVG-6057-HR	Alt Pur Supp Isol Vlv	XRP0103		60	
PVG-6066-HR	Cntmt Purge Exh Isol Vlv	XRP0302		60	
PVG-6067-HR	Cntmt Purge Exh Isol Vlv	XRP0302		60	
PVT-2660-IA	Instrument Air Supply To RB	XRP0311		60	
PVT-2662A-IA	RB Air Cmpr Suct Isol (ORB)	XRP0319		18	Manually set to 18 to meet every outage commitment.
PVT-2662B-IA	RB Air Cmpr Suct Isol (IRB)	XRP0319		18	Manually set to 18 to meet every outage commitment
PVT-8149A-CS	Letdown Orifice "A" Isolation	XRP0318		60	
PVT-8149B-CS	Letdown Orifice "B" Isolation	XRP0318		60	
PVT-8149C-CS	Letdown Orifice "C" Isolation	XRP0318		60	
PVT-8152-CS	Letdown Line Isolation	XRP0318		60	
PVT-8860-SI	Hydro Pump Disharge	XRP0317		60	
PVT-8871-SI	SI Test Line To RWST (IRC)	XRP0321		60	
PVT-8880-SI	SI Accumulator N2 Supply	XRP0320		60	
PVT-8961-SI	SI Test Line To RWST (ORC)	XRP0321		60	
RBLEAKRATEPRES	RB Leak Rate Press Sense Line	XRP0210	Flange	60	
RBLEAKRATETEST	RB Leak Rate Test Flow Test Ln	XRP0201	Flange	60	
SVX-6050A-HR	Post Accid H2 Loop A (IRB)	XRP0301B		60	
SVX-6050B-HR	Post Accid H2 Loop B (IRB)	XRP0105B		60	
SVX-6051A-HR	Post Accid H2 Loop A (IRB)	XRP0301A		60	
SVX-6051B-HR	Post Accid H2 Loop B (IRB)	XRP0105A		60	
SVX-6051C-HR	Post Accid H2 Loop A (DOME)	XRP0301A		60	
SVX-6052A-HR	Post Accid H2 Loop A (ORB)	XRP0301B		60	
SVX-6052B-HR	Post Accid H2 Loop B (ORB)	XRP0105B		60	
SVX-6053A-HR	Post Accid H2 Loop A (ORB)	XRP0301A		60	
SVX-6053B-HR	Post Accid H2 Loop B (ORB)	XRP0105A		60	
SVX-6054-HR	RB NR Press Cntmt Isol	XRP0301B		60	
SVX-9339-SS	PRT Sample Isolation Vlv	XRP0417		60	
SVX-9341-SS	PRT Sample Isolation Vlv	XRP0417		60	
SVX-9356A-SS	Pzr Steam Sample Isol	XRP0405		60	
SVX-9356B-SS	Pzr Liquid Sample Isol	XRP0405		60	
SVX-9357-SS	Pzr Sample Isol	XRP0405		60	
SVX-9364B-SS	RCS Loop B Sample Isol	XRP0314		60	

VCSNS Type B and C Frequencies

Equipment ID	Description	Penetration Number	Type	Interval (Months)	Comments
SVX-9364C-SS	RCS Loop C Sample Isol	XRP0223		60	
SVX-9365B-SS	RCS Loop B Sample Isol	XRP0314		60	
SVX-9365C-SS	RCS Loop C Sample Isol	XRP0223		60	
SVX-9387-SS	Accumulator Sample Isol	XRP0323		60	
XVB00001A-AH	RB Purge Sup Hdr Vlv	XRP0402		18	Manually set to 18 to meet every outage commitment
XVB00001B-AH	RB Purge Supply Isol Vlv (IRC)	XRP0402		18	Manually set to 18 to meet every outage commitment
XVB00002A-AH	Reactor Bldg Purge Exh Valve	XRP0101		18	Manually set to 18 months to meet every outage commitment
XVB00002B-AH	RB Purge Exh Isol Vlv (IRC)	XRP0101		18	Manually set to 18 months to meet every outage commitment
XVC02661-IA	RB Instr Sup Hdr Chk Vlv (IRC)	XRP0311	Check	18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC02913-SA	Reactor Bldg SA Hdr Chk (IRC)	XRP0310	Check	18	ASME Check Valve - Manual Set to 18 Mo (Refuel) Could not pressurize (RF-19) CR-11-01760
XVC03009A-SP	RB Spray Nozzle Inlet Hdr Chk	XRP0401		18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC03009B-SP	RB Spray Nozzle Inlet Hdr Chk	XRP0303		18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC06588-NG	S/G N2 Sup Hdr Check (IRC)	XRP0313	Check	18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC06799-FS	Fire Service Header Chk (IRC)	XRP0427	Check	18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC07541-AC	AC Sup Hdr Cntmt Isol Byp Chk	XRP0208	Check	18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC07544-AC	AC Ret Hdr Cntmt Isol Byp Chk	XRP0209	Check	18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC08046-RC	PRT Spray Header Check Valve	XRP0422	Check	18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC08103-CS	RCP Seal Wtr Ret Isol Bypass C	XRP0410	Check	18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC08368A-CS	RC Pump A Seal Sup Hdr Chk Vlv	XRP0408	Check	18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC08368B-CS	RC Pump B Seal Sup Hdr Chk Vlv	XRP0229	Check	18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC08368C-CS	RC Pump C Seal Sup Hdr Chk	XRP0221	Check	18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC08381-CS	CVCS Charging Hdr Chk Vlv	XRP0409	Check	18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC08861-SI	SI Accum Fill Line Chk Vlv	XRP0317		18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC08947-SI	SI Accum N2 Supply Check Vlv	XRP0320		18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC09570-CC	Excess Letdown Hx CC Sup Hdr	XRP0312		18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC09602-CC	RC Pumps CC Sup Hdr Chk Vlv	XRP0204		18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVC09689-CC	RB CC Ret Hdr Isol Vlv Byp Chk	XRP0330	Check	18	ASME Check Valve - Manual Set to 18 Months (Refuel)
XVD06671-SF	Refuel Cavity SF Pur Suct Isol	XRP0419		60	
XVD06672-SF	Refuel Cavity SF Pur Suct Isol	XRP0419		60	
XVD06697-SF	SF Pur Hdr Refuel Cavity Isol	XRP0421		60	
XVD06698-SF	Refuel Cavity SF Pur Hdr Isol	XRP0421		60	
XVD07126-WL	RCDT Vent To WGPS	XRP0418		60	

VCSNS Type B and C Frequencies

Equipment ID	Description	Penetration Number	Type	Interval (Months)	Comments
XVD07136-WL	RCDT To WPS (ORC)	XRP0423		60	
XVD07150-WL	RCDT Vent To WGPS	XRP0418		60	
XVD07170-WL	RCDT To WPS (IRC)	XRP0423		60	
XVD08028-RC	PRT RMWST Makeup	XRP0422		60	
XVD08033-RC	PRT N2 Vent	XRP0420		60	
XVD08047-RC	PRT N2 & Vent	XRP0420		30	Manually set to base interval, Diaphragm replaced without Pre LLRT (RF-19) CR-11-02214
XVD08767-DN	Demin Water Hdr Sup Isol (ORC)	XRP0231		60	
XVD08768-DN	Demin Water Sup Isol Vlv (IRC)	XRP0231		60	
XVG06772-FS	Fire Service Sup Hdr to RB Iso	XRP0404		60	
XVG06773-FS	Reactor Bldg FS Hdr Isol (IRC)	XRP0404		60	
XVR08117-CS	Letdown Flow Ctrl Hdr Relief	XRP0318	Relief	60	
XVT02679-IA	Out Cntmt Breathing Air Isol	XRP0324		60	
XVT02680-IA	RB Breathing Air Sup Hdr Isol	XRP0324		60	
XVT02912-SA	Reactor Bldg SA Hdr Iso Vlv	XRP0310		60	
XVT06587-NG	S/G Hi Press N2 Hdr Sup Isol	XRP0313		60	

VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)

ATTACHMENT V

Summary Table of LLRT Results Not Demonstrating Acceptable Performance History

VCSNS Failed LLRTs 2003-2012

Test Type	Date	Pass/ Fail	Equuipment ID	Leakage	Units	Acceptance Criteria	Penetration Number	Comments
As Left	5/16/2005	FAIL	PVG-6056-HR	2870	SCCM	2720	XRP0103	POST
As Found	10/27/2003	FAIL	XVC09602-CC	999999	SCCM	2015	XRP0204	Could not pressurize check valve. CER 03-3671 RF-14.
As Found	10/22/2012	FAIL	MVG-7502-AC	6300	SCCM	2720	XRP0208	PRE
As Found	10/22/2012	FAIL	XVC07541-AC	6300	SCCM	2720	XRP0208	PRE
As Found	11/15/2009	FAIL	MVG-7502-AC	999999	SCCM	2720	XRP0208	Could not pressurize pen
As Found	11/15/2009	FAIL	XVC07541-AC	999999	SCCM	2720	XRP0208	Could not pressurize pen
As Left	11/16/2009	FAIL	MVG-7502-AC	3340	SCCM	2720	XRP0208	Post LLRT
As Left	11/16/2009	FAIL	XVC07541-AC	3340	SCCM	2720	XRP0208	Post LLRT
As Found	5/2/2011	FAIL	MVG-7502-AC	6600	SCCM	2720	XRP0208	PRE
As Found	5/2/2011	FAIL	XVC07541-AC	6600	SCCM	2720	XRP0208	PRE
As Found	5/20/2008	FAIL	MVG-7502-AC	7954	SCCM	2720	XRP0208	
As Found	5/20/2008	FAIL	XVC07541-AC	7954	SCCM	2720	XRP0208	
As Left	5/21/2008	FAIL	MVG-7502-AC	3400	SCCM	2720	XRP0208	Rework
As Left	5/21/2008	FAIL	XVC07541-AC	3400	SCCM	2720	XRP0208	Rework
As Found	11/14/2009	FAIL	MVG-7503-AC	999999	SCCM	2720	XRP0209	
As Found	11/14/2009	FAIL	XVC07544-AC	999999	SCCM	2720	XRP0209	
As Found	11/1/2009	FAIL	MVT-8102C-CS	8450	SCCM	2015	XRP0221	
As Found	5/6/2008	FAIL	MVG-8885-SI	2251	SCCM	2015	XRP0222	Pre
As Found	10/31/2012	FAIL	MVG-8701B-RH	15600	SCCM	3625	XRP0226	PRE LLRT, 30 DAY WATER SEAL TEST FAILED, 1112172-001
As Found	4/17/2011	FAIL	XVC02913-SA	999999	SCCM	2015	XRP0310	Could not pressurize the pen
As Found	4/24/2005	FAIL	XVC02913-SA	999999.99	SCCM	2015	XRP0310	PRE, 2913 stuck open could not pressurize pen.
As Found	10/19/2006	FAIL	MVG-9568-CC	5120	SCCM	2920	XRP0312	
As Left	5/14/2011	FAIL	MVG-9568-CC	3830	SCCM	2920	XRP0312	POST
As Found	10/24/2003	FAIL	XVC08861-SI	999999	SCCM	2015	XRP0317	Could not presurize penetration. 999999 false number.
As Found	10/17/2006	FAIL	PVT-8149A-CS	3990	SCCM	2015	XRP0318	
As Found	10/22/2009	FAIL	PVT-8149A-CS	6350	SCCM	2015	XRP0318	Pre LLRT
As Found	11/9/2003	FAIL	PVT-8149C-CS	2380	SCCM	2015	XRP0318	
As Left	11/9/2006	FAIL	PVT-8149A-CS	3220	SCCM	2015	XRP0318	
As Found	4/26/2005	FAIL	PVT-8149C-CS	2150	SCCM	2015	XRP0318	PRE
As Found	4/29/2008	FAIL	PVT-8149A-CS	3860	SCCM	2015	XRP0318	Pre
As Left	5/14/2008	FAIL	PVT-8149A-CS	3350	SCCM	2015	XRP0318	Post
As Left	5/28/2008	FAIL	PVT-8149A-CS	3580	SCCM	2015	XRP0318	Post
As Found	10/30/2009	FAIL	PVT-8880-SI	2400	SCCM	2015	XRP0320	
As Left	11/12/2006	FAIL	MVG-3004B-SP	4380	SCCM	3625	XRP0328	
As Found	5/10/2005	FAIL	MVG-3004B-SP	4310	SCCM	3625	XRP0328	
As Found	10/21/2009	FAIL	MVG-9605-CC	999999	SCCM	2920	XRP0330	Pre LLRT. Could not pressurize pen due to excessive leakage.
As Found	10/21/2009	FAIL	XVC09689-CC	999999	SCCM	2920	XRP0330	Pre LLRT. Could not pressurize pen due to excessive leakage.
As Found	10/26/2003	FAIL	MVG-9605-CC	3140	SCCM	2920	XRP0330	CER 03-3647
As Found	10/26/2003	FAIL	XVC09689-CC	3140	SCCM	2920	XRP0330	
As Found	10/27/2006	FAIL	MVG-9605-CC	999999	SCCM	2920	XRP0330	
As Found	10/27/2006	FAIL	XVC09689-CC	999999	SCCM	2920	XRP0330	Could not pressurize pen
As Found	5/18/2008	FAIL	MVG-9605-CC	7500	SCCM	2920	XRP0330	
As Found	5/18/2008	FAIL	XVC09689-CC	7500	SCCM	2920	XRP0330	
As Found	11/2/2009	FAIL	XVC03009A-SP	4300	SCCM	3225	XRP0401	
As Left	11/11/2012	FAIL	MVB-3110B-SW	7250	SCCM	3626	XRP0403	POST
As Found	11/4/2003	FAIL	MVB-3106B-SW	0	SCCM	0	XRP0403	False record, 3106B removed from program.
As Found	6/1/2008	FAIL	MVB-3106B-SW	0	SCCM	0	XRP0403	False record, 3106B removed from the program
As Found	10/20/2009	FAIL	XVG06773-FS	8290	SCCM	2015	XRP0404	
As Found	10/16/2003	FAIL	PVA-9312B-SS	2240	SCCM	2015	XRP0407B	CER 03-3365 then 03-3801 1st CER written against wrong valve
As Left	11/3/2003	FAIL	PVA-9312B-SS	2420	SCCM	2015	XRP0407B	
As Found	5/18/2005	FAIL	PVA-9312B-SS	2160	SCCM	2015	XRP0407B	
As Found	11/1/2006	FAIL	MVG-8886-SI	2480	SCCM	2015	XRP0412	
As Found	5/6/2008	FAIL	MVG-8884-SI	2462	SCCM	2015	XRP0415	Pre
As Found	11/10/2003	FAIL	XVD06671-SF	5290	SCCM	2015	XRP0419	Pre llrt exceeded Surveillance limit CER 03-3915
As Found	11/10/2003	FAIL	XVD06672-SF	5290	SCCM	2015	XRP0419	Pre LLRT exceeded Surveillance limit CER 03-3915
As Found	10/30/2009	FAIL	MVG-8801A-SI	3010	SCCM	2015	XRP0426	
As Found	10/30/2009	FAIL	MVG-8801B-SI	3010	SCCM	2015	XRP0426	
As Found	10/25/2003	FAIL	XVC06799-FS	2320	SCCM	2015	XRP0427	CER 03-3624
As Found	4/21/2011	FAIL	NOZZLE: 0328	6342	SCCM	2015	XSM0004B	PRE