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SOUTHERN COMPANY
Energy to Serve Your World℠

NL-03-0257

February 26, 2003

Docket Nos.: 50-424
50-425

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Vogtle Electric Generating Plant
Technical Specification Revision Request
<u>Integrated Leakage Rate Testing Interval Extension</u>

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) is proposing a change to the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 Technical Specifications (TS). This proposed change will revise TS section 5.5.17, "Containment Leakage Rate Testing Program," to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT). The ten (10) year interval between integrated leakage rate tests is to be extended to fifteen (15) years from the previous integrated leakage rate tests, which were completed in March 2002 (Unit 1) and March 1995 (Unit 2).

This proposed change is based on and has been evaluated using the "risk informed" guidance in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The "Risk Assessment for Vogtle Electric Generating Plant Regarding the ILRT (Type A) Extension Request" is provided as an attachment to this letter.

Enclosure 1 provides a description of the proposed change and an explanation of the basis for the change. Enclosure 2 details the basis for SNC's determination that the proposed change does not involve a significant hazards consideration. Enclosure 3 provides page change instructions for incorporating the proposed change along with the revised Technical Specification page and the corresponding marked-up page.

Southern Nuclear Operating Company requests the proposed amendment be approved by July 31, 2003, to support the planning activities for the Unit 2 outage scheduled in March 2004.

A similar request was approved for Indian Point 3 in a letter dated April 17, 2001, Crystal River 3 in a letter dated August 30, 2001, and Peach Bottom 3 in a letter dated October 4, 2001.

A001

This letter contains no new commitments.

In accordance with the requirements of 10 CFR 50.91, a copy of this letter and all applicable enclosures will be sent to the designated state official of the Environmental Protection Division of the Georgia Department of Natural Resources.

Mr. J. T. Gasser states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company, and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

Jeffrey T. Gasser

Jeffrey T. Gasser

Sworn to and subscribed before me this 26th *day of* February*, 2003.*

Notary Public

My commission expires: 11/10/06

JTG/DRG

Enclosures:
1. Basis for Change Request
2. 10 CFR 50.92 Evaluation
3. VEGP Technical Specification Changed Page List, Marked-up Pages and Typed Pages

Attachment: Risk Assessment for Vogtle Electric Generating Plant Regarding the ILRT (Type A) Extension Request

cc: Southern Nuclear Operating Company
Mr. J. D. Woodard, Executive Vice President
Mr. G. R. Frederick, General Manager – Plant Vogtle
Mr. M. Sheibani, Engineering Supervisor – Plant Vogtle
Document Services RTYPE: CVC7000

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. F. Rinaldi, NRR Project Manager – Vogtle
Mr. J. Zeiler, Senior Resident Inspector – Vogtle

State of Georgia
Mr. L. C. Barrett, Commissioner – Department of Natural Resources

Vogtle Electric Generating Plant
Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension

Enclosure 1

Basis for Change Request

Vogtle Electric Generating Plant
Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension

Enclosure 1

Basis for Change Request

Proposed Change

Southern Nuclear Operating Company (SNC) is proposing a change to the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 Technical Specifications (TS). This proposed change will revise TS section 5.5.17, "Containment Leakage Rate Testing Program," to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT). The ten (10) year interval between integrated leakage rate tests is to be extended to fifteen (15) years from the previous integrated leakage rate tests, which were completed in March 2002 (Unit 1) and March 1995 (Unit 2).

The proposed change involves a one-time exception to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J." The current ten (10) year ILRT for VEGP is due in March 2012 (Unit 1) and March 2005 (Unit 2), which would require the test to be performed during Refueling Outage 1R16 (March 2011) and Refueling Outage 2R10 (March 2004). The proposed exception would allow the next ILRTs for VEGP to be performed within fifteen (15) years (Unit 1 - March 2017 and Unit 2 - March 2010) from the last ILRTs as opposed to the current ten (10) year frequency.

The proposed change would revise Section 5.5.17, "Containment Leakage Rate Testing Program," of the VEGP Technical Specifications by modifying the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995 with the following one-time exception to NEI 94-01, Rev. 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

> **Section 9.2.3: The next Type A test, after the March 2002 test for Unit 1 and the March 1995 test for Unit 2, shall be performed within 15 years. This is a one-time exception.**

This one-time exception will result in the following:

- For Unit 1, the Type A Containment ILRT will be performed during Refueling Outage Unit 1R20, currently scheduled for March 2017.

- For Unit 2, the Type A Containment ILRT will be performed during Refueling Outage Unit 2R14, currently scheduled for March 2010.

- A substantial cost savings will be realized, and unnecessary personnel radiation exposure will be avoided by deferring the Type A test for an additional five (5) years. Cost savings have been estimated for each outage at approximately $1.95 million, which includes labor, equipment, and critical path outage time needed to perform the test. Personnel radiation exposure reduction for each outage is estimated at 750 mrem.

Basis for Proposed Change

a. 10 CFR 50, Appendix J, Option B

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 Technical Specifications. The limitation on containment leakage provides assurance that the containment will perform its design function following plant design basis accidents.

10 CFR 50, Appendix J was revised, effective October 26, 1995, to allow licensees to perform containment leakage testing in accordance with the requirements of Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." Amendment 96 was issued for VEGP Unit 1 (dated September 25, 1996) and Amendment 74 was issued for VEGP Unit 2 (dated September 25, 1996) to replace the current Technical Specifications and associated Bases with a set based on Revision 1 to NUREG-1431, "Standard Technical Specifications Westinghouse Plants." The Improved Technical Specifications implemented by the amendments permit implementation of 10 CFR 50, Appendix J, Option B. These amendments revised Technical Specifications to require Type A, B, and C testing in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program." RG 1.163 specifies a method acceptable to the NRC for complying with 10 CFR 50, Appendix J, Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8-1994, subject to several regulatory positions in the guide.

Exceptions to the requirements of RG 1.163 are permitted by 10 CFR 50, Appendix J, Option B, as discussed in Section V.B, "Implementation." Therefore, this application does not require an exemption from 10 CFR 50, Appendix J, Option B.

Adoption of the Option B performance-based containment leakage rate testing program 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 and maintenance history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained.

The allowed frequency for Type A testing, as documented in NEI 94-01, is based, in part, upon a generic evaluation documented in NUREG-1493. The evaluation documented in NUREG-1493 included a study of the dependence of reactor accident risks on containment leak-tightness for five reactor/containment types including a Westinghouse designed pressurized water reactor in a large, dry containment building (VEGP Unit 1 and 2 are large, dry containment buildings). NUREG-1493 made the following observations with regard to decreasing the test frequency.

- Reducing the Type A Integrated Leak Rate Test (ILRT) testing frequency to one per twenty (20) years was found to lead to imperceptible increase in risk. The estimated increase in risk is small because ILRTs identify only a few potential 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 the existing requirements. Given the insensitivity of risk to containment leakage rate, and the small fraction of leakage detected solely by Type A testing, increasing the interval between ILRT testing has minimal impact on public risk.

- While Type B and C tests identify the vast majority (greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

NEI 94-01 requires that Type A testing be performed at least once per ten (10) years based upon an acceptable performance history. Acceptable performance history is defined as two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than 1.0 L_a. Based upon the acceptable ILRTs for Unit 1 (March 1993 and March 2002) and for Unit 2 (April 1992 and March 1995), the current test interval for VEGP is once every ten (10) years, with the next test due to be performed by March 2012 on Unit 1 and March 2005 on Unit 2.

b. VEGP Integrated Leak Rate Test History

Type A testing is performed to verify the integrity of the containment structure in its Loss of Coolant Accident (LOCA) configuration. Industry test experience has demonstrated that Type B and C testing detect a large percentage of containment leakage and that the percentage of containment leakage that is detected only by integrated containment leakage testing is very small.

VEGP Unit 1 has undergone three operational Type A tests and Unit 2 has undergone two operational Type A tests, in addition to the pre-operational Structural Integrity Test performed on each unit. The results of these tests demonstrate that the VEGP containment structures for Unit 1 and Unit 2 remain essentially leak-tight barriers and represent minimal risk to increased leakage. These plant-specific results support the conclusions of NUREG-1493. As specified in VEGP Technical Specifications Section 5.5.17, the maximum allowable containment leakage rate L_a, at P_a, is 0.2% of primary containment air weight per day. The VEGP ILRT results are provided below.

Unit 1

Refueling Outage	(95% Upper Confidence Limit % Cnmt Air Mass/Day)
1R2 (March 1990)	0.1048
1R4 (March 1993)	0.1344
1R10 (March 2002)	0.0336

Unit 2

Refueling Outage	(95% Upper Confidence Limit % Cnmt Air Mass/Day)
2R2 (April 1992)	0.1373
2R4 (March 1995)	0.0938

c. Containment Inspections

Containment leak tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the 1992 Edition through the 1992 Addenda of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI. More specifically, subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure retaining components and their integral attachments in light water cooled plants. Furthermore, NRC regulations, 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas on the interior of the containment three times every 10 years. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak tight integrity of containment penetration bellows, airlocks, seals, and gaskets are not affected by the change to the Type A test frequency. Likewise the Appendix J, Type C local leak tests, which are performed to verify the leak tight integrity of containment isolation valves, are not affected by the change to the Type A test frequency.

d. Typical Questions

The NRC has sent Requests for Additional Information (RAI) to several licensees concerning their request for a technical specification revision allowing a one-time ILRT interval extension. These RAIs contain five typical questions. Listed below are the five questions with the VEGP responses:

1. NRC Question

Since there is no description (or summarization) regarding the containment ISI program being implemented at VEGP, please provide a description of the ISI methods that provide assurance that in the absence of an ILRT for 15 years, the containment structural and leak tight integrity will be maintained.

VEGP Response:

As described in Section c above, containment leak tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the 1992 Edition through the 1992 Addenda of ASME Code Section XI. More specifically, subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure retaining components and their integral attachments in light water cooled plants. Furthermore, NRC regulations, 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas in the interior of the containment three times every 10 years. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak tight integrity of containment penetration bellows, airlocks, seals, and gaskets are not affected by the change to the Type A test frequency. Likewise the Appendix J, Type C local leak tests, which are performed to verify the leak tight integrity of containment isolation valves, are not affected by the change to the Type A test frequency.

The ASME Code Section XI IWE and IWL containment inspections provide a high degree of assurance that any degradation of the containment structure is identified and corrected before a containment leakage path is introduced.

2. **NRC Question**

IWE-1240 requires licensees to identify the containment surface areas requiring augmented examinations. Please provide the locations of the containment liner surfaces that have been identified as requiring augmented examination and a summary of the findings of the examinations performed.

VEGP Response:

There are no areas of the VEGP Unit 1 or Unit 2 containment liners that require augmented examinations per IWE-1240.

3. **NRC Question**

For the examination of seals and gaskets, and examination and testing of bolted connections associated with the primary containment pressure boundary (Examination Categories E-D and E-G), relief from the requirements of the Code had been requested. As an alternative, it was proposed to examine them during the leak rate testing of the primary containment. However, Option B of Appendix J for Type B and Type C testing (as per Nuclear Energy Institute 94-01 and Regulatory Guide 1.163) and the ILRT extension requested in this amendment for Type A testing provide flexibility in the scheduling of these inspections. Please provide your schedule for examination and testing of seals, gaskets, and bolts that provide assurance regarding the integrity of the containment pressure boundary.

VEGP Response:

The one-time extension requested by SNC applies only to the 10 CFR 50, Appendix J, Type A integrated leak rate test that is currently on a 10-year interval pursuant to Appendix J, Option B, Performance Based Requirements. Appendix J, Type B and Type C tests are performed at the intervals required by Appendix J, Option B and will be tested at least once in the 10-year interval. This frequency of testing of seals, gaskets, and containment pressure retaining bolting provides reasonable assurance that the integrity of the containment pressure boundary is maintained during the period of the extension.

4. **NRC Question**

The stainless steel bellows have been found to be susceptible to trans-granular stress corrosion cracking and the leakage through them is not readily detectable by Type B testing (see Information Notice 92-20). If applicable, please provide information regarding inspection and testing of the bellows, and how such behavior has been factored into the risk assessment.

VEGP Response:

NRC Information Notice 92-20, Inadequate Local Leak Rate Testing, discussed the inadequate local leak rate testing of two-ply stainless steel bellows. VEGP does not have such bellows as a part of the containment pressure boundary.

5. **NRC Question**

Inspections of some reinforced concrete and steel containment structures have found degradation on the uninspectable (embedded) side of the drywell steel shell and steel liner of the primary containment. These degradations cannot be found by visual (i.e., VT-1 or VT-3) examinations unless they are through the thickness of the shell or liner, or 100% of the uninspectable surfaces are periodically examined by ultrasonic testing. Please provide information (additional analyses) addressing how potential leakage under high pressure during core damage accidents is factored into the risk assessment related to the extension of the ILRT.

VEGP Response:

The attached "Risk Assessment for Vogtle Electric Generating Plant Regarding the ILRT (Type A) Extension Request" provides a sensitivity evaluation considering potential corrosion impacts within the framework of the ILRT interval extension risk assessment. The analysis confirms that the ILRT interval extension has a minimal impact on plant risk. Additionally, a series of parametric sensitivity studies regarding the potential age-related corrosion effects on the steel liner also indicate that even with very conservative assumptions, the conclusions from the original analysis would not change. That is, the ILRT interval extension is judged to have a minimal impact on plant risk and is therefore acceptable.

The attached analysis also clarifies the delta LERF for the original License Bases "three tests in 10 years" and the proposed "one test in 15 years." The analysis also provides a discussion on the effects ILRT interval extension would have on the total LERF (internal and external events) for VEGP. The conclusion shows that the total LERF for both VEGP Units is well below the RG 1.174 acceptance criteria for total LERF of 1.0E-05.

e. Risk Assessment

Attached is a detailed performance based, risk informed assessment, "Risk Assessment for Vogtle Electric Generating Regarding the ILRT (Type A) Extension Request," to support this request.

f. Similar Requests

This request is similar to the requests for change of the Indian Point 3 ILRT frequency that was approved by the NRC on April 17, 2001, Crystal River 3 approved in an NRC letter dated August 30, 2001, and Peach Bottom 3 approved in an NRC letter dated October 4, 2001. The PRA has been enhanced with the knowledge gained from the NRC's evaluation of the recent Crystal River Unit 3 submittal.

g. Conclusion

Based on the attached risk assessment results, the containment leak rate test results, and containment inspection results, the requested change is concluded to be acceptable.

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Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension

Enclosure 2

10 CFR 50.92 Evaluation

Vogtle Electric Generating Plant
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Enclosure 2

10 CFR 50.92 Evaluation

In 10 CFR 50.92 (c), the NRC provides the following standards to be used in determining the existence of a significant hazards consideration:

> ...a proposed amendment to an operating license for a facility licensed under §50.21(b) or §50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

Southern Nuclear Operating Company has reviewed the proposed license amendment request and determined its adoption does not involve a significant hazards consideration based on the following discussion.

Basis for No Significant Hazards Consideration Determination

1. *The proposed Technical Specifications change does not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The proposed revision to Technical Specifications 5.5.17, "Containment Leakage Rate Testing Program," involves a one-time extension to the current interval for Type A containment leak testing. The current test interval of ten (10) years would be extended on a one-time basis to no longer than fifteen (15) years from the last Type A test. The proposed Technical Specifications 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 reactor 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, the proposed Technical Specification change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG-1493, that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. VEGP test history supports this conclusion. NUREG-1493 concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk. The integrity of the reactor containment is subject to two types of failure mechanism 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 design change control 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 reactor containment itself combined with the containment inspections performed in accordance with ASME Section XI, the Maintenance Rule, and the containment coatings program 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. Therefore, the proposed Technical Specifications change does not involve a significant increase in the consequences of an accident previously evaluated.

2. *The proposed Technical Specifications change does not create the possibility of a new or different kind of accident from any accident previously evaluated.*

The proposed revision to Technical Specifications involves a one-time extension to the current interval for Type A containment leak 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 Technical Specifications change does not involve a physical change to the plant or the manner in which the plant is operated or controlled. Therefore, the proposed Technical Specifications change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *The proposed Technical Specifications change does not involve a significant reduction in a margin of safety.*

The proposed revision to Technical Specifications involves a one-time extension to the current interval for Type A containment leak testing. The proposed Technical Specifications 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 Containment Leakage Rate Testing Program, as defined in Technical Specifications, 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 leakage rate limit specified by Technical Specifications is maintained. The

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Enclosure 3

VEGP Technical Specification Marked-up Page

VEGP Technical Specification Typed Pages proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications.

VEGP and industry experience strongly support 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, the Maintenance Rule, and the containment coatings program 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. Therefore, the proposed Technical Specifications change does not involve a significant reduction in a margin of safety.

Environmental Impact

The proposed Technical Specifications changes were reviewed against the criteria of 10 CFR 51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, a significant increase in the amounts of effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposures. Based on the foregoing, Southern Nuclear Operating Company concludes that the proposed Technical Specifications change meets the criteria given in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

Conclusion

SNC has concluded that the proposed change to the VEGP Technical Specifications does not involve a Significant Hazards Consideration.

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Changed Page List

Changed Page	Revision Instruction
5.5-20	Replace
5.5-21	Replace

5.5 Programs and Manuals (continued)

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment 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-Testing Program," dated September 1995, as modified by the following exceptions:

1. Leakage rate testing for containment purge valves with resilient seals is performed once per 18 months in accordance with LCO 3.6.3, SR 3.6.3.6 and SR 3.0.2.

2. Containment personnel air lock door seals will be tested prior to reestablishing containment integrity when the air lock has been used for containment entry. When containment integrity is required and the air lock has been used for containment entry, door seals will be tested at least once per 30 days during the period that containment entry(ies) is (are) being made.

3. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC. At the discretion of the licensee, the containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage.

4. **A one time exception to NEI 94-01, Rev. 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J":**

   Section 9.2.3: **The next Type A test, after the March 2002 test for Unit 1 and the March 1995 test for Unit 2, shall be performed within 15 years.**

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 37 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.2% of primary containment air weight per day.

5.5 Programs and Manuals (continued)

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment 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-Testing Program," dated September 1995, as modified by the following exceptions:

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3. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC. At the discretion of the licensee, the containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage.

4. A one time exception to NEI 94-01, Rev. 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

   Section 9.2.3: The next Type A test, after the March 2002 test for Unit 1 and the March 1995 test for Unit 2, shall be performed within 15 years.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 37 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.2% of primary containment air weight per day.

(continued)

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program (continued)

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criteria are $\leq 1.0\ L_a$. 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;

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,

2) For each door, the leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.18 Configuration Risk Management Program

The Configuration Risk Management Program (CRMP) provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for which a risk-informed allowed outage time has been granted. The program shall include the following elements:

a. Provisions for the control and implementation of a Level 1 at power internal events PRA-informed methodology. The assessment shall be capable of evaluating the applicable plant configuration.

b. Provisions for performing an assessment prior to entering the LCO Condition for preplanned activities.

c. Provisions for performing an assessment after entering the LCO Condition for unplanned entry into the LCO Condition.

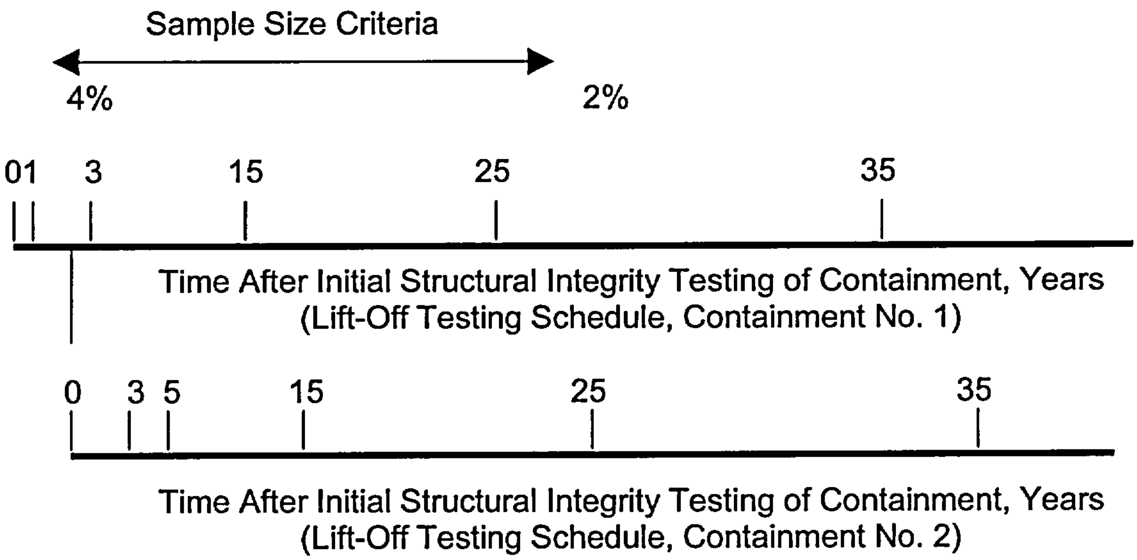
(continued)

5.5 Programs and Manuals

5.5.18 Configuration Risk Management Program (continued)

d. Provisions for assessing the need for additional actions after the discovery of additional equipment out of service conditions while in the LCO Condition.

e. Provisions for considering other applicable risk significant contributors such as Level 2 issues and external events, qualitatively or quantitatively.



<u>Schedule to be used provided:</u>

a. The containments are identical in all aspects such as size, tendon system, design, materials of construction, and method of construction. The tendon system for Unit 2 does not provide for detensioning. Detensioning can be performed only on the Unit 1 tendon system.

b. The 1-year inspection for Unit 2 will consist of a visual inspection only. No lift-off testing will be performed on Unit 2 until the 3-year inspection.

c. There is no unique situation that may subject either containment to a different potential for structural or tendon deterioration.

d. The Unit 1 and Unit 2 surveillances may be performed back-to-back to facilitate detensioning of Unit 1 tendons during the Unit 2 surveillance.

Figure 5.5.6-1 Schedule of Lift-Off Testing for Two Containments at a Site

Vogtle Electric Generating Plant
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Attachment

Risk Assessment for Vogtle Electric Generating Plant
Regarding the ILRT (Type A) Extension Request

RISK ASSESSMENT FOR

VOGTLE ELECTRIC GENERATING PLANT

REGARDING THE

ILRT (TYPE A) EXTENSION REQUEST

Prepared for:

Southern Nuclear Operating Co.

Prepared by:

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February 2003

Vogtle Electric Generating Plant

RISK ASSESSMENT FOR VOGTLE ELECTRIC GENERATING PLANT REGARDING THE ILRT (TYPE A) EXTENSION REQUEST

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Revisions:

Rev.	Description	Preparer/Date	Reviewer/Date	Approver/Date

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SECTION 1
PURPOSE OF ANALYSIS

1.0 PURPOSE

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A integrated leak rate test (ILRT) to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for the Vogtle Electric Generating Plant. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], the NEI "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 [3], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [4], and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval [5].

1.1 BACKGROUND

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three in ten years to at least once in ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least

24 months apart in which the calculated performance leakage was less than normal containment leakage of 1La.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995 [6], provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals."

The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative PWR plant (i.e., Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for Vogtle.

The NEI Interim Guidance for performing risk impact assessments in support of ILRT extensions builds on the EPRI Risk Assessment methodology, EPRI TR-104285. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI. More specifically, Subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E) require licensees to conduct visual inspections of the accessible areas of the interior of the containment three times every ten years. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

1.2 CRITERIA

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this one-time extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines <u>very small</u> changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. RG 1.174 also defines <u>small</u> changes in LERF as below 10^{-6} per reactor year. RG 1.174 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. Therefore, the increase in the conditional containment failure probability (CCFP) that helps to ensure that the defense-in-depth philosophy is maintained is also calculated.

In addition, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter. (No criteria have been established for this parameter change.)

SECTION 2
METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years. The approach is consistent with that presented in NEI Interim Guidance [3], EPRI TR-104285 [2], NUREG-1493 [6] and the Calvert Cliffs liner corrosion analysis [5]. The analysis uses results from a Level 2 analysis of core damage scenarios from the current Vogtle PSA model and subsequent containment response resulting in various fission product release categories (including no or negligible release). This risk assessment is applicable to Vogtle Units 1 and 2.

The six general steps of this assessment are as follows:

1. Quantify the baseline risk in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report.
2. Develop plant-specific person-rem (population dose) per reactor year for each of the eight containment release scenario types from plant specific consequence analyses.
3. Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval to fifteen years.
4. Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 [4] and compare with the acceptance guidelines of RG 1.174.
5. Determine the impact on the Conditional Containment Failure Probability (CCFP)

6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis and to the fractional contribution of increased large isolation failures (due to liner breach) to LERF.

This approach is based on the information and approaches contained in the previously mentioned studies. Furthermore,

- Consistent with the other industry containment leak risk assessments, the Vogtle assessment uses population dose as one of the risk measures. The other risk measures used in the Vogtle assessment are LERF and the conditional containment failure probability (CCFP) to demonstrate that the acceptance guidelines from RG 1.174 are met.
- This evaluation for Vogtle uses ground rules and methods to calculate changes in risk metrics that are similar to those used in the NEI Interim Guidance.

SECTION 3
GROUND RULES

The following ground rules are used in the analysis:

- The Vogtle Level 1 and Level 2 internal events PSA models provide representative results.
- It is appropriate to use the Vogtle internal events PSA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire and seismic events were to be included in the calculations.
- Dose results for the containment failures modeled in the PSA can be characterized by information provided in NUREG/CR-4551 [7]. They are estimated by scaling the NUREG/CR-4551 results by population differences for Vogtle compared to the NUREG/CR-4551 reference plant.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [2] and are summarized in Section 4.2.
- The representative containment leakage for Class 1 sequences is $1L_a$. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is $10L_a$. based on the previously approved methodology performed for Indian Point Unit 3 [8, 9].
- The representative containment leakage for Class 3b sequences is $35L_a$. based on the previously approved methodology [8, 9].
- The Class 3b can be very conservatively categorized as LERF based on the previously approved methodology [8, 9]. The Class 3b category increase is used as a surrogate for LERF in this application even though the releases associated with a $35L_a$ release would not necessarily be consistent with a "Large" release for Vogtle. (See, for example, the calculated population dose results for EPRI Class 3b in Table 5-4 of 7.77E3 person-rem compared to the 4.25E5 person-rem associated with EPRI Class 8 for containment bypass scenarios.)

- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.

- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

SECTION 4
INPUTS

This section summarizes the general resources available as input (Section 4.1) and the plant specific resources required (Section 4.2).

4.1 GENERAL RESOURCES AVAILABLE

Various industry studies on containment leakage risk assessment are briefly summarized here:

1. NUREG/CR-3539 [10]
2. NUREG/CR-4220 [11]
3. NUREG-1273 [12]
4. NUREG/CR-4330 [13]
5. EPRI TR-105189 [14]
6. NUREG-1493 [6]
7. EPRI TR-104285 [2]
8. NUREG-1150 [15] and NUREG/CR-4551 [7]
9. NEI Interim Guidance [3]
10. Calvert Cliffs liner corrosion analysis [5]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PSA for the size of containment leakage that is considered significant and is to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. The eighth study provides an ex-plant consequence analysis for a 50-mile radius surrounding a plant that is used as the bases for the consequence analysis of the ILRT interval extension for VEGP. The ninth study includes the NEI recommended methodology for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. Finally, the tenth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations.

<u>NUREG/CR-3539 [11]</u>

Oak Ridge National Laboratory documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [16] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

NUREG/CR-4220 [12]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage.

NUREG-1273 [12]

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

NUREG/CR-4330 [13]

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

> "...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

EPRI TR-105189 [14]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because it provides insight regarding the impact of containment testing on shutdown risk. This study contains a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk. The conclusion from the study is that a small but measurable safety benefit is realized from extending the test intervals.

NUREG-1493 [6]

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk
- Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending ILRT and LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 uses a simplified Containment Event Tree to subdivide representative core damage frequencies into eight classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failures due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded:

> *"... the proposed CLRT* [containment leak rate tests] *frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.02 person-rem per year ..."*

NUREG-1150 [15] and NUREG/CR 4551 [7]

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Surry. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the VEGP Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent VEGP. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

<u>NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [3]</u>

The guidance provided in this document builds on the EPRI risk impact assessment methodology [2] and the NRC performance-based containment leakage test program [6], and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

The approach included in this guidance document is used in the VEGP assessment to determine the estimated increase in risk associated with the ILRT extension. This document includes the bases for the values assigned in determining the probability of leakage for the EPRI Class 3a and 3b scenarios in this analysis as described in Section 5.

<u>Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension [5]</u>

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms were factored into the risk assessment for the ILRT one-time extension. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. Vogtle has a similar type of containment.

4.2 PLANT-SPECIFIC INPUTS

The plant-specific information used to perform the VEGP ILRT Extension Risk Assessment includes the following:

- Level 1 Model results [17]
- Level 2 Model results [17]
- Release category definitions used in the Level 2 Model [18]
- Population within a 50-mile radius [19]
- ILRT results to demonstrate adequacy of the administrative and hardware issues [19] (1)
- Containment failure probability data [18]

Level 1 Model

The Level 1 PSA model that is used for VEGP is characteristic of the as-built plant. The current Level 1 model is a linked fault tree model, and was quantified with the total Core Damage Frequency (CDF) = 1.59E-5/yr. This applies to both Unit 1 and Unit 2.

(1) The two most recent Type A tests at VEGP Unit 1 and Unit 2 have been successful, so the current Type A test interval requirement is 10 years.

Level 2 Model

The Level 2 Model that is used for VEGP was developed to calculate the LERF contribution as well as the other elease categories evaluated in the model. The total LERF, which corresponds to VEGP release categories D, G, and T in Table 4.2-1, is 5.89E-8/yr. Table 4.2-1 summarizes the pertinent VEGP results in terms of release category. (Note that after adjustments are made to include the uncategorized releases into the results as described in Section 5.1, the baseline total LERF value is 6.57E-8.)

Table 4.2-1

VEGP Level 2 PSA Model Release Categories and Frequencies [17, 18]

Release Category	Definition	Frequency/yr
A	No containment failure within 48-hour mission time, but failure could eventually occur without further mitigating action; noble gases and less than 0.1% volatiles released	1.42E-5
D	Containment bypassed with noble gases and up to 10% of the volatiles released	4.26E-9
G	Containment failure prior to vessel failure with noble gases and up to 10% of the volatiles released (containment not isolated)	5.98E-10
K	Late containment failure with noble gases and less than 0.1% volatiles released (containment failure greater than 6 hours after vessel failure; containment not bypassed; isolation successful prior to core damage)	2.23E-8
S	Success (leakage only, successful maintenance of containment integrity; containment not bypassed; isolation successful prior to core damage)	4.32E-9
T	Containment bypassed with noble gases and more than 10% of the volatiles released.	5.40E-8
	Total Release Category Frequency	1.42E-5
	Core Damage Frequency (including uncategorized releases)	1.59E-5

Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 and adjusting the results for VEGP. Each of the release categories from Table 4.2-1 was associated with an applicable Collapsed Accident Progression Bin (APB) from NUREG/CR-4551 (see below). The collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 7 bins that are relevant to the analysis. The definitions of the 7 collapsed APBs are provided in NUREG/CR-4551 and are reproduced in Table 4.2-2 for references purposes. Table 4.2-3 summarizes the calculated population dose for Surry associated with each APB from NUREG/CR-4551.

Table 4.2-2

Summary Accident Progression Bin (APB) Descriptions [7]

Summary APB Number	Description
1	CD, VB, Early CF, Alpha Mode Core damage occurs followed by a very energetic molten fuel-coolant interaction in the vessel; the vessel fails and generates a missile that fails the containment as well. Includes accidents that have an Alpha mode failure of the vessel and the containment except those follow Event V or an SGTR. It includes Alpha mode failures that follow isolation failures because the Alpha mode containment failure is of rupture size.
2	CD, VB, Early CF, RCS Pressure > 200psia Core Damage occurs followed by vessel breach. Implies Early CF with the RCS above 200 psia when the vessel fails. Early CF means at or before VB, so it includes isolation failures and seismic containment failures at the start of the accident as well as containment failure at VB. It does not include bins in which containment failure at VB follows Event V or an SGTR, or Alpha mode failures.

Table 4.2-2

Summary Accident Progression Bin (APB) Descriptions [7]

Summary APB Number	Description
3	CD, VB, Early CF, RCS Pressure < 200 psia Core damage occurs followed by vessel breach. Implies Early CF with the RCS below psia when the containment fails. It does not include bins in which the containment failure at VB or an SGTR, or Alpha mode failures.
4	CD, VB, Late CF Core Damage occurs followed by vessel breach. Includes accidents in which the containment was not failed or bypassed before the onset of core-concrete interaction (CCI) and in which the vessel failed. The failure mechanisms are hydrogen combustion during CCI, Basemat Melt-Through (BMT) in several days, or eventual overpressure due to the failure to provide containment heat removal in the days following the accident.
5	CD, Bypass Core Damage occurs followed by vessel breach. Includes Event V and SGTRs no matter what happens to the containment after the start of the accident. It also includes SGTRs that do not result in VB.
6	CD, VB, No CF Core Damage occurs followed by vessel breach. Includes accidents not evaluated in one of the previous bins. The vessel's lower head is penetrated by the core, but the containment does not fail and is not bypassed.
7	CD, No VB Core Damage occurs but is arrested in time to prevent vessel breach. Includes accident progressions that avoid vessel failures except those that bypass the containment. Most of the bins placed in this reduce bin have no containment failure as well as no VB. It also includes bins in which the containment is not isolated at the start of the accident and the core is brought to a safe stable state before the vessel fails.

Table 4.2-3

Calculation of Surry Population Dose Risk at 50 Miles [7]

Collapsed Bin #	Fractional APB Contributions to Risk (MFCR) (1)	NUREG/CR-4551 Population Dose Risk at 50 miles (person-rem/yr, mean) (2)	NUREG/CR-4551 Collapsed Bin Frequencies (per year) (3)	NUREG/CR-4551 Population Dose at 50 miles (person-rem) (4)
1	0.029	0.158	1.23E-07	1.28E+06
2	0.019	0.106	1.64E-07	6.46E+05
3	0.002	0.013	2.012E-08	6.46E+05 (5)
4	0.216	1.199	2.42E-06	4.95E+05
5	0.732	4.060	5.00E-06	8.12E+05
6	0.001	0.006	1.42E-05	4.23E+02
7	0.002	0.011	1.91E-05	5.76E+02
Totals	1.000	5.55	4.1E-05	

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(1) Mean Fractional Contribution to Risk calculated from the average of two samples delineated in Table 5.1-3 of NUREG/CR-4551.

(2) The total population dose risk at 50 miles from internal events in person-rem is provided as the average of two samples in Table 5.1-1 of NUREG/CR-4551. The contribution for a given APB is the product of the total PDR50 and the fractional APB contribution.

(3) NUREG/CR-4551 provides the conditional probabilities of the collapsed APBs in Figure 2.5-3. These conditional probabilities are multiplied by the total internal CDF to calculate the collapsed APB frequency.

(4) Obtained from dividing the population dose risk shown in the third column of this table by the collapsed bin frequency shown in the fourth column of this table.

(5) Assumed population dose at 50 miles for Collapsed Bin #3 equal to that of Collapsed Bin #2. Collapsed Bin Frequency #3 was then back calculated using that value. This does not influence the results of this evaluation since Bin #3 does not appear as part of the results for VEGP.

Population Estimate Methodology

The person-rem results in Table 4.2-3 can be used as an approximation of the dose for the Vogtle Electric Generating Plant if it is corrected for the population surrounding VEGP. The total population within a 50-mile radius of VEGP is 645,000 [19].

This population value is compared to the population value that is provided in NUREG/CR-4551 in order to get a "Population Dose Factor" that can be applied to the APBs to get dose estimates for VEGP.

Total VEGP Population$_{50miles}$ = 6.45E+05
Surry Population from NUREG/CR-4551 = 1.23E+06
Population Dose Factor = 6.45E+05 / 1.23E+06 = 0.524

The difference in the doses at 50 miles is assumed to be in direct proportion to the difference in the population within 50 miles of each site. This does not take into account differences in meteorology data, detailed environmental factors or detailed differences in containment designs, but does provide a first-order approximation for VEGP of the population doses associated with each of the release categories from NUREG/CR-4551. This is considered adequate since the conclusions from this analysis will not be substantially affected by the actual dose values that are used.

Table 4.2-4 shows the results of applying the population dose factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for VEGP.

Table 4.2-4

Calculation of VEGP Population Dose Risk at 50 Miles

Accident Progression Bin (APB)	NUREG/CR-4551 Population Dose at 50 miles (person-rem)	Bin Multiplier used to obtain VEGP Population Dose	VEGP Adjusted Population Dose at 50 miles (person-rem)
1	1.28E+06	0.524	6.71E+05
2	6.46E+05	0.524	3.39E+05
3	6.46E+05	0.524	3.39E+05
4	4.95E+05	0.524	2.59E+05
5	8.12E+05	0.524	4.25E+05
6	4.23E+02	0.524	2.22E+02
7	5.76E+02	0.524	3.02E+02

<u>Application of VEGP PSA Model Results to NUREG/CR-4551 Level 3 Output</u>

A major factor related to the use of NUREG/CR-4551 in this evaluation is that the results of the VEGP PSA Level 2 model are not defined in the same terms as reported in NUREG/CR-4551. In order to use the Level 3 model presented in that document, it was necessary to match the VEGP PSA Level 2 release categories to the collapsed APBs. The assignments are shown in Table 4.2-5, along with the corresponding EPRI/NEI classes (see below).

Table 4.2-5

VEGP Level 2 Model Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins and EPRI / NEI Accident Classes

VEGP Level 2 Release Category	Definition	NUREG/ CR-4551 APB	EPRI/NEI Class
A	No containment failure within 48-hour mission time, but failure could eventually occur without further mitigating action; noble gases and less than 0.1% volatiles released	6	1
D	Containment bypassed with noble gases and up to 10% of the volatiles released	5	8
G	Containment failure prior to vessel failure with noble gases and up to 10% of the volatiles released(containment not isolated)	2	2
K	Late containment failure with noble gases and less than 0.1% volatiles released(containment failure greater than 6 hours after vessel failure; containment not bypassed; isolation successful prior to core damage)	4	7
S	Success (leakage only, successful maintenance of containment integrity; containment not bypassed; isolation successful prior to core damage)	7	1
T	Containment bypassed with noble gases and more than 10% of the volatiles released.	5	8

Release Category Definitions

Table 4.2-6 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI/NEI methodology [2]. These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 5 of this report.

Table 4.2-6

EPRI/NEI CONTAINMENT FAILURE CLASSIFICATIONS [2]

Class	Description
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a, under Appendix J for that plant
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is a failure to isolate the containment.
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

4.3 IMPACT OF EXTENSION ON DETECTION OF COMPONENT FAILURES THAT LEAD TO LEAKAGE (SMALL AND LARGE)

The ILRT can detect a number of component failures such as liner breach, failure of certain bellows arrangements and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class, as defined in Table 4.2-6, is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures, respectively.

The probability of the EPRI Class 3a and 3b failures is determined consistent with the NEI Guidance [3]. For Class 3a, the probability is based on the mean failure from the available data (i.e., 5 "small" failures in 182 tests leads to a 5/182=0.027 mean value). For Class 3b, a non-informative prior distribution is assumed for no "large" failures in 182 tests (i.e., 0.5/(182+1) = 0.0027).

In a follow on letter [20] to their ILRT guidance document [3], NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC regulatory guide 1.174. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for delta LERF. NEI describes ways to demonstrate that, using plant-specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The supplemental information states:

> The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain

conservatism. However, some plant-specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage.

The application of this additional guidance to the analysis for VEGP, as detailed in Section 5, involves the following:

1. The Class 2 and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 and Class 8 events refer to sequences with either large pre-existing containment isolation failures or containment bypass events. These sequences are already considered to contribute to LERF in the Vogtle Level 2 PSA analysis.

2. A review of Class 1 accident sequences shows that several of these cases involve successful operation of containment sprays. It is assumed that, for calculation of the Class 3b and 3a frequencies, the fraction of the Class 1 CDF associated with successful operation of containment sprays can also be subtracted. A review of the VEGP accident bins that contribute to Class 1 (VEGP release categories A and S) reveals that sprays are available in 2.35% of the cases. Table 4.3-1 provides a detailed breakdown of the sequences in categories A and S. Sprays are not credited for any of the other release categories.

Table 4.3-1

VEGP Level 2 Sequences Contributing to EPRI/NEI Class 1 [17]

Bin #	VEGP Release Category	Frequency	Sprays Available?
1	A	1.30E-05	No
2	A	3.40E-07	No
3	A	2.39E-07	No
4	A	1.92E-07	No
5	A	1.00E-07	Yes
6	A	9.56E-08	Yes
7	A	6.25E-08	Yes
10	A	3.03E-08	Yes
11	A	3.19E-08	No
12	A	2.49E-08	Yes
15	A	1.82E-08	No
17	A	7.96E-09	Yes
18	A	7.87E-09	Yes
20	A	5.26E-09	No
21	S	4.32E-09	Yes
Total		1.42E-05	

Consistent with the NEI Guidance [3], the change in the leak detection probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three-year test interval is 1.5 years (3 yr / 2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 yr / 2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by ILRT testing. Correspondingly, an extension of the ILRT interval to fifteen years can be estimated to lead to about a factor of 5.0 (7.5/1.5) increase in the non-detection probability of a leak.

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the IP3 request for a one-time ILRT extension that was approved by the NRC [9]) because it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur.) Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

4.4 IMPACT OF EXTENSION ON DETECTION OF STEEL LINER CORROSION THAT LEADS TO LEAKAGE

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis [5]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. Vogtle has a similar type of containment.

The following approach is used to determine the change in likelihood, due to extending the ILRT, of detecting corrosion of the containment steel liner. This likelihood is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment basemat and the containment cylinder and dome
- The historical steel liner flaw likelihood due to concealed corrosion

- The impact of aging
- The corrosion leakage dependency on containment pressure
- The likelihood that visual inspections will be effective at detecting a flaw

Assumptions

- Consistent with the Calvert Cliffs analysis, a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures. (See Table 4.4-1, Step 1.)
- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to this Vogtle containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner.
- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is also limited to 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date (and have been performed since the time frame of the Calvert Cliffs analysis), and there is no evidence that additional corrosion issues were identified. (See Table 4.4-1, Step 1.)
- Consistent with the Calvert Cliffs analysis, the steel liner flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel liner ages. (See Table 4.4-1, Steps 2 and 3.) Sensitivity studies are included that address doubling this rate every ten years and every two years.
- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure, and the selected values are consistent with a pressure that corresponds to the ILRT target pressure of 37 psig. For Vogtle, the containment failure probabilities are less than these values at 37 psig [18]. Conservative probabilites of 1% for the cylinder and dome and 0.1% for the basemat are used in this analysis, and sensitivity studies are included that increase and decrease the probabilities by an order of magnitude. (See Table 4.4-1, Step 4.)

- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than the containment cylinder and dome region. (See Table 4.4-1, Step 4.)
- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. To date, all liner corrosion events have been detected through visual inspection. (See Table 4.4-1 , Step 5.) Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.
- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

Analysis

Table 4.4-1

Steel Liner Corrosion Base Case

Step	DESCRIPTION	Containment Cylinder and Dome		Containment Basemat	
1	**Historical Steel Liner Flaw Likelihood**	Events: 2		Events: 0 (assume half a failure)	
	Failure Data: Containment location specific (consistent with Calvert Cliffs analysis).	2/(70 * 5.5) = **5.2E-3**		0.5/(70 * 5.5) = **1.3E-3**	
2	Age Adjusted Steel Liner Flaw Likelihood	Year	Failure Rate	Year	Failure Rate
	During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).	1 avg 5-10 15	2.1E-3 5.2E-3 1.4E-2	1 avg 5-10 15	5.0E-4 1.3E-3 3.5E-3
		15 year average = 6.27E-3		**15 year average = 1.57E-3**	

Table 4.4-1

Steel Liner Corrosion Base Case

Step	DESCRIPTION	Containment Cylinder and Dome	Containment Basemat
3	Flaw Likelihood at 3, 10, and 15 years Uses age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis – See Table 6 of Reference [5]).	**0.71% (1 to 3 years)** **4.06% (1 to 10 years)** **9.40% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the desired presentation of the results.	**0.18% (1 to 3 years)** **1.02% (1 to 10 years)** **2.35% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with desired presentation of the results.
4	Likelihood of Breach in Containment Given Steel Liner Flaw The failure probability of the cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1%, (compared to 0.11% in the Calvert Cliffs analysis).	**1%**	**0.1%**

Table 4.4-1

Steel Liner Corrosion Base Case

Step	DESCRIPTION	Containment Cylinder and Dome	Containment Basemat
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4* 5)	**0.00071% (at 3 years)** 0.71% * 1% * 10% **0.0041% (at 10 years)** 4.1% * 1% * 10% **0.0094% (at 15 years)** 9.4% * 1% * 10%	**0.00018% (at 3 years)** 0.18% * 0.1% * 100% **0.0010% (at 10 years)** 1.0% * 0.1% * 100% **0.0024% (at 15 years)** 2.4% * 0.1% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the containment cylinder and dome and the containment basemat as summarized below.

Total Likelihood Of Non-Detected Containment Leakage Due To Corrosion:

- At 3 years: 0.00071% + 0.00018% = 0.00089%
- At 10 years: 0.0041% + 0.0010% = 0.0051%
- At 15 years: 0.0094% + 0.0024% = 0.0118%

SECTION 5
RESULTS

The application of the approach based on NEI Interim Guidance [3], EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [5, 8, 21, 22, 23] have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 5-1 lists these accident classes.

The analysis performed examined VEGP-specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the break down of the severe accidents contributing to risk were considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellows leakage. (EPRI TR-104285 Class 3 sequences).
- Core damage sequences in which containment integrity is impaired due to post-containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test. (EPRI TR-104285 Class 6 sequences). Consistent with the NEI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.
- Accident sequences involving containment bypassed (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.

- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 5-1

ACCIDENT CLASSES

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (liner breach)
3b	Large Isolation Failures (liner breach)
4	Small Isolation Failures (Failure to seal –Type B)
5	Small Isolation Failures (Failure to seal—Type C)
6	Other Isolation Failures (e.g., dependent failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End states (including very low and no release)

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 - Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 5-1.

Step 2 - Develop plant-specific person-rem dose (population dose) per reactor year for each of the eight accident classes.

Step 3 - Evaluate risk impact of extending Type A test interval from 3 to 15 and 10 to 15 years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174.

Step 5 - Determine the impact on the Conditional Containment Failure Probability (CCFP)

5.1 STEP 1 - QUANTIFY THE BASE-LINE RISK IN TERMS OF FREQUENCY PER REACTOR YEAR

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-104285). The question on containment integrity was modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes were considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies for the severe accident classes defined in Table 5-1 were developed for VEGP by first determining the frequencies for Classes 1, 2, 7 and 8 using the categorized sequences and the identified correlations shown in Table 4.2-5, scaling these frequencies to account for the uncategorized sequences, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. Furthermore, adjustments were made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion of the steel liner per the methodology described in Section 4.4.

The total frequency of the categorized sequences is 1.42E-5/yr, and the total CDF is 1.59E-5, so the scale factor is 1.1158. Table 5-2 contains the frequencies from the categorized sequences, and the resulting frequencies due to the scale factor. The results are summarized below and in Table 5-3.

Table 5-2

Vogtle Categorized Accident Classes and Frequencies

EPRI/NEI Class	VEGP Release Category	Frequency Based on Categorized Results (per yr)	Adjusted Frequency Using Scale Factor of 1.1158 (per yr)
1	A + S	1.42E-5 + 4.32E-9	1.58E-5
2	G(1)	5.98E-10	6.67E-10
7	K	2.23E-8	2.49E-8
8	D + T(1)	4.26E-9 + 5.40E-8	6.50E-8
	Total Frequency	1.42E-5	1.59E-5

(1) The estimated Total LERF value corresponding to release categories D, G, & T is now 6.57E-8/yr.

<u>Class 1 Sequences</u>. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year is initially determined from the Level 2 Release Categories A and S, listed in Table 5-2, minus the EPRI/NEI Class 3a and 3b frequency, calculated below.

Class 2 Sequences. This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. The frequency per year for these sequences is obtained from the Release Category G, listed in Table 5-2.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small ($2L_a$ to $35L_a$) or large ($>35L_a$).

The respective frequencies per year are determined as follows:

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.027 [see Section 4.3]
$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.0027 [see Section 4.3]

As described in section 4.3, additional consideration is made to not apply these failure probabilities on those cases that are already LERF scenarios (i.e., the Class 2 and Class 8 contributions), or that would include containment spray operation such that a Large Release would be unlikely (i.e., 2.35% of the VEGP Release Categories A and S).

CLASS_3A_FREQUENCY = 0.027 ∗ (CDF-Class 2-Class 8–0.0235*Class 1)
= 0.027 * (1.59E-05 – 6.67E-10 – 6.50E-08 – 0.0235 * 1.58E-05) = 4.17E-7/yr
CLASS_3B_FREQUENCY = 0.0027 ∗ (CDF-Class 2-Class 8–0.0235*Class 1)
=0.0027 * (1.59E-05 – 6.67E-10 – 6.50E-08 – 0.0235 * 1.58E-05) = 4.17E-8/yr

For this analysis, the associated containment leakage for Class 3A is $10L_a$ and for Class 3B is $35L_a$ These assignments are consistent with the NEI Interim Guidance.

Class 4 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 5 Sequences. This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type C test components. Because the failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 6 Sequences. This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. Consistent with the NEI Interim Guidance, however, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (e.g., overpressure). For this analysis, the frequency is determined from Release Category K from the VEGP Level 2 results.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. For this analysis, the frequency is determined from Release Categories D and T from the VEGP Level 2 results.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definitions of accident classes defined in EPRI-TR-104285 and the NEI Interim Guidance. Table 5-3 summarizes these accident frequencies by accident class for VEGP.

Table 5-3

RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (VEGP BASE CASE)

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	
		NEI Methodology	NEI Methodology Plus Corrosion
1	No Containment Failure	1.53E-05	1.53E-05
2	Large Isolation Failures (Failure to Close)	6.67E-10	6.67E-10
3a	Small Isolation Failures (liner breach)	4.17E-07	4.17E-07
3b	Large Isolation Failures (liner breach)	4.17E-08	4.19E-08
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA
7	Failures Induced by Phenomena (Early and Late)	2.49E-08	2.49E-08
8	Bypass (Interfacing System LOCA)	6.50E-08	6.50E-08
CDF	All CET end states	1.59E-05	1.59E-05

5.2 STEP 2 - DEVELOP PLANT-SPECIFIC PERSON-REM DOSE (POPULATION DOSE) PER REACTOR YEAR

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on information provided by NUREG/CR-4551 with adjustments made for the site demographic differences compared to the reference plant as described in Section 4.2, and summarized in Table 4.2-4. The results of applying these releases to the EPRI/NEI containment failure classification are as follows:

Class 1	=	222 person-rem (at $1.0L_a$)	=	222 person-rem [(1)]
Class 2	=	3.39E+05 [(2)]		
Class 3a	=	222 person-rem x $10L_a$	=	2.22E+03 person-rem [(3)]
Class 3b	=	222 person-rem x $35L_a$	=	7.77E+03 person-rem [(3)]
Class 4	=	Not analyzed		
Class 5	=	Not analyzed		
Class 6	=	Not analyzed		
Class 7	=	2.59E+05 person-rem [(4)]		
Class 8	=	4.25E+05 person-rem [(5)]		

(1) The derivation is described in Section 4.2 for VEGP. Class 1 is assigned the dose from the "no containment failure" APBs from NUREG/CR-4551 (i.e., APB #6 and APB #7). The dose is calculated as a weighted average of the dose for these bins using the CDFs for categories A and S.

(2) The Class 2, containment isolation failures, dose is assigned from APB #2 (Early CF).

(3) The Class 3a and 3b dose are related to the leakage rate as shown. This is consistent with the NEI Interim Guidance.

(4) The Class 7 dose is assigned from APB #4 (Late CF).

(5) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are assigned from APB #5 (Bypass).

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology [2] containment failure classifications, and consistent with the NEI guidance [3] are provided in Table 5-4.

Table 5-4

VEGP POPULATION DOSE ESTIMATES FOR POPULATION WITHIN 50 MILES

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)
1	No Containment Failure	2.22E+02
2	Large Isolation Failures (Failure to Close)	3.39E+05
3a	Small Isolation Failures (liner breach)	2.22E+03
3b	Large Isolation Failures (liner breach)	7.77E+03
4	Small Isolation Failures (Failure to seal –Type B)	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA
6	Other Isolation Failures (e.g., dependent failures)	NA
7	Failures Induced by Phenomena (Early and Late)	2.59E+05
8	Bypass (Interfacing System LOCA)	4.25E+05

The above dose estimates, when combined with the results presented in Table 5-3, yield the VEGP baseline mean consequence measures for each accident class. These results are presented in Table 5-5.

Table 5-5

VEGP ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure (2)	2.22E+02	1.53E-05	3.41E-03	1.53E-05	3.41E-03	-3.04E-08
2	Large Isolation Failures (Failure to Close)	3.39E+05	6.67E-10	2.26E-04	6.67E-10	2.26E-04	–
3a	Small Isolation Failures (liner breach)	2.22E+03	4.17E-07	9.26E-04	4.17E-07	9.26E-04	–
3b	Large Isolation Failures (liner breach)	7.77E+03	4.17E-08	3.24E-04	4.19E-08	3.25E-04	1.07E-06
4	Small Isolation Failures (Failure to seal – Type B)	NA	NA	NA	NA	NA	NA
5	Small Isolation Failures (Failure to seal— Type C)	NA	NA	NA	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA	NA	NA	NA
7	Failures Induced by Phenomena (Early and Late)	2.59E+05	2.49E-08	6.44E-03	2.49E-08	6.44E-03	–
8	Bypass (Interfacing System LOCA)	4.25E+05	6.50E-08	2.76E-02	6.50E-08	2.76E-02	–
CDF	All CET end states		1.59E-05	0.0390	1.59E-05	0.0390	1.04E-6

1) Only release Classes 1 and 3b are affected by the corrosion analysis.

2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

The VEGP dose compares favorably with other locations given the relative population densities surrounding each location:

Plant	Annual Dose (Person-Rem/Yr)	Reference
Indian Point 3	14,515	[9]
Peach Bottom	6.2	[21]
Farley Unit 2	2.4	[22]
Farley Unit 1	1.5	[22]
Crystal River	1.4	[23]
Vogtle	0.0390	[Table 5-5]

5.3 STEP 3 - EVALUATE RISK IMPACT OF EXTENDING TYPE A TEST INTERVAL FROM 10-TO-15 YEARS

The next step is to evaluate the risk impact of extending the test interval from its current ten-year value to fifteen-years. To do this, an evaluation must first be made of the risk associated with the ten-year interval since the base case applies to a 3-year interval (i.e., a simplified representation of a 3-in-10 interval).

Risk Impact Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is impacted. The risk contribution is changed based on the NEI guidance as described in Section 4.3 by a factor of 3.33 compared to the base case values. The results of the calculation for a 10-year interval are presented in Table 5-6.

Risk Impact Due to 15-Year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 compared to the 3-year interval value, as described in Section 4.3. The results for this calculation are presented in Table 5-7.

Table 5-6

VEGP ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	2.22E+02	1.43E-05	3.17E-03	1.43E-05	3.17E-03	-1.74E-07
2	Large Isolation Failures (Failure to Close)	3.39E+05	6.67E-10	2.26E-04	6.67E-10	2.26E-04	–
3a	Small Isolation Failures (liner breach)	2.22E+03	1.39E-06	3.08E-03	1.39E-06	3.08E-03	–
3b	Large Isolation Failures (liner breach)	7.77E+03	1.39E-07	1.08E-03	1.40E-07	1.09E-03	6.09E-06
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA	NA	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA	NA	NA	NA
7	Failures Induced by Phenomena (Early and Late)	2.59E+05	2.49E-08	6.44E-03	2.49E-08	6.44E-03	–
8	Bypass (Interfacing System LOCA)	4.25E+05	6.50E-08	2.76E-02	6.50E-08	2.76E-02	–
CDF	All CET end states		1.59E-05	0.0416	1.59E-05	0.0416	5.92E-6

1) Only release classes 1 and 3b are affected by the corrosion analysis.

2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5-7

VEGP ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/15 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Person-Rem/yr[1]
1	No Containment Failure [2]	2.22E+02	1.35E-05	3.00E-03	1.35E-05	3.00E-03	-4.03E-07
2	Large Isolation Failures (Failure to Close)	3.39E+05	6.67E-10	2.26E-04	6.67E-10	2.26E-04	–
3a	Small Isolation Failures (liner breach)	2.22E+03	2.09E-06	4.63E-03	2.09E-06	4.63E-03	–
3b	Large Isolation Failures (liner breach)	7.77E+03	2.09E-07	1.62E-03	2.10E-07	1.63E-03	1.41E-05
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA	NA	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA	NA	NA	NA
7	Failures Induced by Phenomena (Early and Late)	2.59E+05	2.49E-08	6.44E-03	2.49E-08	6.44E-03	–
8	Bypass (Interfacing System LOCA)	4 25E+05	6.50E-08	2.76E-02	6.50E-08	2.76E-02	–
CDF	All CET end states		1.59E-05	0.0436	1.59E-05	0.0436	1.37E-5

1) Only release classes 1 and 3b are affected by the corrosion analysis.

2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.4 STEP 4 - DETERMINE THE CHANGE IN RISK IN TERMS OF LARGE EARLY RELEASE FREQUENCY (LERF)

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the NEI guidance, 100% of the Class 3b contribution would be considered LERF. For Vogtle, however, the Class 3b radionuclide release person-rem is significantly less than a typical LERF contributor as can be seen by comparing the relative population dose for Class 3b to that of Class 2 (7.77E3 person-rem / 3.39E5 person-rem or 2.3%). Additionally, the Vogtle calculated dose for Accident Class 3b is also much lower than that calculated in previous submittals for an ILRT extension (e.g., 4.94E7 person-rem for IP3 [8] and 3.45E4 person-rem for Crystal River [23], both of which were approved by the NRC).

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines <u>very small</u> changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr, and <u>small</u> changes in LERF as below 10^{-6}/yr. Because the ILRT does not impact CDF, the relevant metric is LERF.

For Vogtle, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the NEI guidance methodology). Based on a ten-year test interval from Table 5-6, the Class 3b frequency is 1.40E-7/yr; and, based on a fifteen-year test interval from Table 5-7, it is 2.10E-7. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is 1.69E-7/yr. Similarly, the increase due to increasing the interval from 10 to 15 years is 7.07E-8/yr. As can be seen, even with the conservatisms included in the evaluation (per the NEI methodology), the estimated change in LERF for is below the threshold criteria for a very small change when comparing the 15 year results to the current 10-year requirement, and just above that criteria when compared to the original 3-year requirement.

5.5 STEP 5 – DETERMINE THE IMPACT ON THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY (CCFP)

Another parameter that the NRC guidance in RG 1.174 states can provide input into the decision-making process is the change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The CCFP can be calculated from the results of this analysis. One of the difficult aspects of this calculation is providing a definition of the "failed containment." In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state. The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP can be calculated by using the method specified in the NEI Interim Guidance. The NRC has previously accepted similar calculations [9] as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy.

$$CCFP = [1 - (\text{Class 1 frequency} + \text{Class 3a frequency}) / CDF] * 100\%$$

$$CCFP_3 = 0.83\%$$
$$CCFP_{10} = 1.45\%$$
$$CCFP_{15} = 1.89\%$$

$$\Delta CCFP = CCFP_{15} - CCFP_{3} = 1.06\%$$

$$\Delta CCFP = CCFP_{15} - CCFP_{10} = 0.44\%$$

The change in CCFP of slightly more than 1% by extending the test interval to 15 years from the original 3-in-10 year requirement is judged to be insignificant.

5.6 SUMMARY OF RESULTS

The results from this ILRT extension risk assessment for Vogtle are summarized in Table 5-8.

Table 5-8

VEGP ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	2.22E+02	1.53E-05	3.41E-03	1.43E-05	3.17E-03	1.35E-05	3.00E-03
2	3.39E+05	6.67E-10	2.26E-04	6.67E-10	2.26E-04	6.67E-10	2.26E-04
3a	2.22E+03	4.17E-07	9.26E-04	1.39E-06	3.08E-03	2.09E-06	4.63E-03
3b	7.77E+03	4.19E-08	3.25E-04	1.40E-07	1.09E-03	2.10E-07	1.63E-03
7	2.59E+05	2.49E-08	6.44E-03	2.49E-08	6.44E-03	2.49E-08	6.44E-03
8	4.25E+05	6.50E-08	2.76E-02	6.50E-08	2.76E-02	6.50E-08	2.76E-02
Total		1.59E-05	0.0390	1.59E-05	0.0416	1.59E-05	0.0436
ILRT Dose Rate from 3a and 3b		1.25E-03		4.17E-03		6.27E-03	
Delta Total Dose Rate	From 3 yr	—		2.92E-03		4.61E-03	
	From 10 yr	—		—		1.93E-03	
3b Frequency (LERF)		4.19E-08		1.40E-07		2.10E-07	
Delta LERF	From 3 yr	—		9.65E-08		1.69E-07	
	From 10 yr	—		—		7.07E-08	
CCFP %		0.83%		1.45%		1.89%	
Delta CCFP %	From 3 yr	—		0.62%		1.06%	
	From 10 yr	—		—		0.44%	

SECTION 6
SENSITIVITIES

6.1 SENSITIVITY TO CORROSION IMPACT ASSUMPTIONS

The results in Tables 5-5, 5-6 and 5-7 show that including corrosion effects calculated using the assumptions described in Section 4.4 does not significantly affect the results of the ILRT extension risk assessment.

Sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the cylinder and dome and the basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 6-1. In every case the impact from including the corrosion effects is very minimal. Even the upper bound estimates with very conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 5.37E-8 /yr. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

Table 6-1

Steel Liner Corrosion Sensitivity Cases

Age (Step 3 in the corrosion analysis)	Containment Breach (Step 4 in the corrosion analysis)	Visual Inspection & Non-Visual Flaws (Step 5 in the corrosion analysis)	Increase in Class 3b Frequency (LERF) for ILRT Extension 3 to 15 years (per Rx-yr)	
			Total Increase	Increase Due to Corrosion
Base Case Doubles every 5 yrs	Base Case (1% Cylinder, 0.1% Basemat)	Base Case 10%	1.69E-07	1.68E-09
Doubles every 2 yrs	Base	Base	1.70E-07	3.84E-09
Doubles every 10 yrs	Base	Base	1.68E-07	1.42E-09
Base	Base	*15%*	1.69E-07	2.35E-09
Base	Base	*5%*	1.68E-07	1.01E-09
Base	10% Cylinder, 1% Basemat	Base	1.84E-07	1.68E-08
Base	0.1% Cylinder, 0.01% Basemat	Base	1.67E-07	1.68E-10
Lower Bound				
Doubles every 10 yrs	0.1% Cylinder, 0.01% Basemat	5% 1%	1.67E-07	8.49E-11
Upper Bound				
Doubles every 2 yrs	10% Cylinder, 1% Basemat	15% 100%	2.21E-07	5.37E-08

6.2 SENSITIVITY TO CLASS 3B CONTRIBUTION TO LERF

The Class 3b frequency for the base case of a three in ten-year ILRT interval is 4.19E-8/yr [Table 5-5]. Extending the interval to one in ten years results in a frequency of 1.40E-7/yr [Table 5-6]. Extending it to one in fifteen years results in a frequency of 2.10E-7/yr [Table 5-7], which is an increase of 1.69E-7/yr. If 100% of the Class 3b sequences are assumed to have potential releases large enough for LERF, then the increase in LERF due to extending the interval from three in ten to one in fifteen is above the RG 1.174 threshold for very small changes in LERF of 1E-7/yr.

Realistically, only a fraction of the Class 3b frequency would have the potential to contribute to LERF at Vogtle. Figure 6-1 shows shows a representative population dose frequency distribution for Class 3b with the calculated mean value of 7.77E3 person-rem (based on a characteristic 35La release magnitude) using a lognormal distribution and with an assumed range factor of 10. As can be seen, only a small probability would be associated with a release magnitude that would be categorized as sufficiently large for a LERF release. The reverse cumulative distribution plot shown in Figure 6-2 bears out the fact that the probability of a large release (i.e., assumed to be represented by a magnitude approaching 5E4 person-rem or higher) is quite small in this example. Consequently, even with a larger assumed error factor, it can be conservatively stated that no more than 10% of the Class 3b releases could have the potential to be LERF for Vogtle. Therefore, a factor of 10 reduction can be made to the calculated EPRI Class 3b scenarios in reporting a more realistic assessment of the LERF contribution from the ILRT extension. This correlates to an estimated LERF increase of 1.69E-8/yr assuming as described above, that 10% of the frequency of Class 3B sequences can be used as a first-order estimate to approximate a more realistic potential increase in LERF from the ILRT interval extension. This estimated change in LERF then below the threshold criteria for a very small change. These results are summarized in Table 6-2.

Figure 6-1
POPULATION DOSE (PERSON-REM) FREQUENCY DISTRIBUTION FOR ACCIDENT CLASS 3B

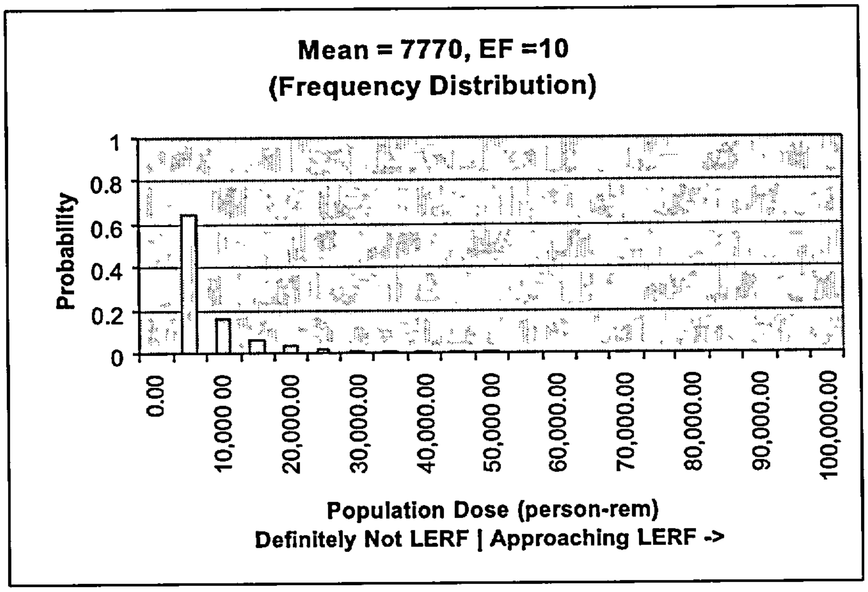

Figure 6-2
POPULATION DOSE (PERSON-REM) REVERSE CUMULATIVE DISTRIBUTION FOR ACCIDENT CLASS 3B

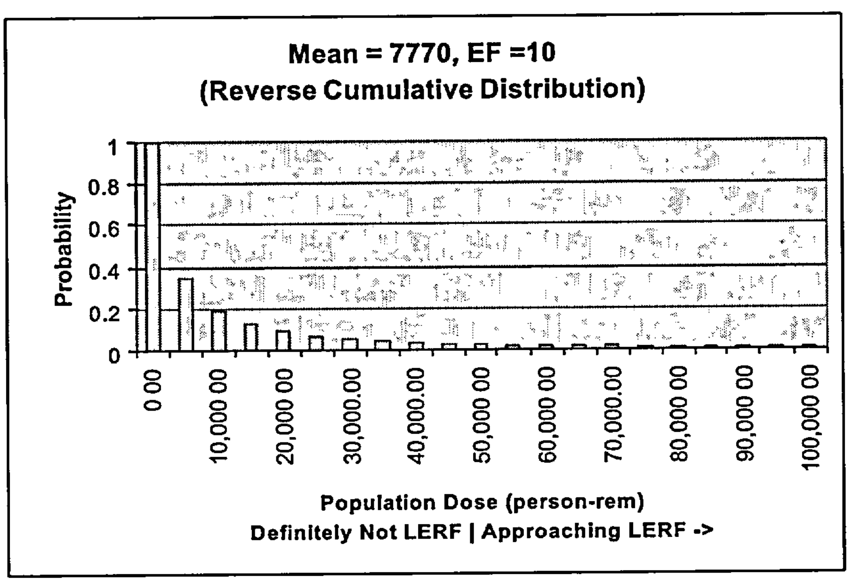

Table 6-2

Increase in LERF Due to Class 3b

	LERF due to 100% 3b Frequency			**LERF due to 10% 3b Frequency**		
	3 in 10 years	**Extend to 1 in 10 years**	**Extend to 1 in 15 years**	**3 in 10 years**	**Extend to 1 in 10 years**	**Extend to 1 in 15 years**
	4.19E-08	1.40E-07	2.10E-07	4.19E-09	1.40E-08	2.10E-08
Delta from 3 yr	—	9.79E-08	1.69E-07	—	9.79E-09	1.69E-08
Delta from 10 yr	—	—	7.07E-08	—	—	7.07E-09

6.3 POTENTIAL IMPACT FROM EXTERNAL EVENTS CONTRIBUTION

In the Vogtle IPEEE, the dominant risk contributor from external events was found to be from fire events. Other potential contributors such as seismic and high winds were found to be within acceptable limits.

At the time of the IPEEE, the Vogtle internal events CDF was 4.45E-05/reactor-year (single model for both units) and the calculated fire CDF was 1.01E-05/reactor-year. A fire LERF was not calculated for the IPEEE [24].

At the time of the fire analysis, LOSP was the dominant contributor to core damage in the Vogtle PRA. The fire high risk areas involved the main control room, switchgear rooms, and other areas affecting electrical power supply and control (electrical raceways, cable spreading, and electrical penetration rooms) in which a fire could lead to an SBO causing a loss of RCP seal cooling resulting in core uncovery due to a seal LOCA.

Since the IPEEE, the Vogtle PRA has been converted from a large event tree model to a linked fault tree model using CAFTA software. Due to the PRA conversion process and four subsequent updates, LOSP is no longer the dominant contributor to internal events CDF, which has been reduced to 1.59E-05/reactor year. The internal events CDF is now dominated by a complete loss of nuclear service cooling water (NSCW) special initiating event. A complete loss of NSCW causes a loss of all RCP seal cooling resulting in a RCP seal LOCA, leading to core uncovery. Therefore, it is reasonable to assume that the External Events CDF can be approximated as no greater than the current Internal Events CDF for calculating the potential impact of the ILRT extension.

For Vogtle, the reported total Internal Events LERF as determined from a simplified LERF model is 7.80E-08/reactor-year [24]. Table 5-2 from this analysis provides an estimated total Internal Events LERF value of 6.57E-8/reactor-year. There are some known conservatisms in the simplified LERF model and truncation value impacts that account for this difference, but the higher value will be used in the discussion below for illustration purposes.

Additionally, the External Events baseline LERF would be expected to be less than the Internal Events baseline LERF because some of the Internal Events baseline LERF comes from events that are not events that are initiated by fires (i.e., ISLOCA and SGTR). However, as shown below, even if it is conservatively assumed that the External Events baseline LERF is equivalent to the Internal Events baseline LERF, the total LERF would still be far below the Regulatory Guide 1.174 criteria of 1.0E-05 following the ILRT extension.

Two cases are examined. The first case utilizes the NEI methodology directly in estimating the LERF increase from the ILRT extension (i.e., no reduction in the 3b LERF contribution is made). The second case utilizes the 10% reduction factor in applying what could be considered a more reasonable LERF contribution from the ILRT extension for Vogtle (see Section 6.2 discussion). The results from each of these calculations are shown in Table 6-3.

Table 6-3

Vogtle Estimated Total LERF Including External Events Impact

Contributor	**NEI Directly (With 100% of Class 3b to LERF from ILRT)**	**NEI Enhanced (With 10% of Class 3b to LERF from ILRT)**
Internal Events LERF	7.80E-08	7.80E-08
External Events LERF	7.80E-08	7.80E-08
Internal Events LERF due to ILRT (at 15 years)	2.10E-07	2.10E-08
External Events LERF due to ILRT (at 15 years)	2.10E-07	2.10E-08
Total:	**5.76E-07**	**1.98E-07**

SECTION 7
CONCLUSIONS

Based on the results from Section 5 and the sensitivity calculations presented in Section 6, the following conclusions regarding the assessment of the plant risk are associated with extending the Type A ILRT test frequency to fifteen years:

- Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/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 very conservatively estimated as 1.69E-7/yr using the NEI guidance as written, and more realistically estimated at 1.69E-8/yr with a slight variation to the NEI guidance. As such, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.

- Regulatory Guide 1.174 [4] also states that when the calculated increase in LERF is in the range of 1.0E-06 per reactor year to 1.0E-07 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-05 per reactor year. If the 10% reduction factor is not applied to the Class 3b frequencies for determining LERF from the ILRT interval extension as described in Section 6.2, then the overall results could fall into this range. As such, an additional assessment of the impact from external events was also made. In that case, the total LERF was conservatively estimated as 5.76E-07 for Vogtle Units 1 and 2. This is well below the RG 1.174 acceptance criteria for total LERF of 1.0E-05.

- 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 0.0046 person-rem/yr. Therefore, the risk impact when compared to other severe accident risks is negligible.

- The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen year interval is 1.06%. Although no official acceptance criteria exist for this risk metric, it is judged to be very small.

Therefore, increasing the ILRT interval to 15 years is considered to be insignificant since it represents a very small change to the Vogtle Electric Generating Plant risk profile.

Previous Assessments

The NRC in NUREG-1493 [4] has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from three per 10 years to one per 20 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. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment structure.

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

SECTION 8
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