From: Cayetano Santos
Sent: Monday, April 10, 2023 12:57 PM
To: Vogtle PEmails
Subject: Draft LAR 23-007 for pre-submittal meeting
Attachments: .LAR-23-007_ILRT Exception draft for PSM.pdf

Hearing Identifier: Vogtle_COL_Docs_Public
Email Number: 658

Mail Envelope Properties (MW4PR09MB90100D7FA4B06010EA805EABE5959)

Subject: Draft LAR 23-007 for pre-submittal meeting
Sent Date: 4/10/2023 12:57:28 PM
Received Date: 4/10/2023 12:57:34 PM
From: Cayetano Santos

Created By: Cayetano.Santos@nrc.gov

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"Vogtle PEmails" <Vogtle.PEmails@usnrc.onmicrosoft.com>
Tracking Status: None

Post Office: MW4PR09MB9010.namprd09.prod.outlook.com

Files	**Size**	**Date & Time**
MESSAGE	3	4/10/2023 12:57:34 PM
.LAR-23-007_ILRT Exception draft for PSM.pdf	766366	

Options
Priority: Normal
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:

ENCLOSURE

License Amendment Request for Exception to Regulatory Guide 1.163 (LAR-23-007)

1. SUMMARY DESCRIPTION

Southern Nuclear Operating Company (SNC) requests an amendment to the combined license (COL) for Vogtle Electric Generating Plant (VEGP) Unit 3 (License Number NPF-91). The proposed change would revise Appendix A of the COL by adding an exception to Regulatory Guide (RG) 1.163, Performance-Based Containment Leak-Test Program, in Technical Specification (TS) 5.5.8, Containment Leakage Rate Testing Program.

2. DETAILED DESCRIPTION

2.1 System Design and Operation

The containment is a free-standing steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low-leakage steel vessel designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA) such that offsite radiation exposures are maintained within limits. The containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The inner steel containment and its penetrations establish the leakage limiting boundary of the containment. The TS leakage rate Surveillance Requirements comply with 10 CFR Part 50, Appendix J, Option B, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Performance-Based Requirements,” as modified by approved exemptions.

2.2 Current Requirements

TS 5.5.8 requires that a program 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 RG 1.163, “Performance-Based Containment Leak-Test Program, dated September 1995,” as modified by approved exceptions.

RG 1.163 (September 1995), Section C provides the following Regulatory Position:

> NEI 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance Based Option of 10 CFR 50, Appendix J," prepared by the Nuclear Energy Institute, provides methods acceptable to the NRC staff for complying with the provisions of Option B in Appendix J to 10 CFR Part 50, subject to four identified considerations.

The UFSAR Appendix 1A identifies that VEGP Unit 3 conforms to the above RG 1.163 (September 1995) position and thus, conforms to NEI 94-01, Revision 0.

NEI 94-01, Rev. 0, Section 9.2.2, Initial Test Intervals identifies the following:

> A preoperational Type A test shall be conducted prior to initial reactor operation. If initial reactor operation is delayed longer than 36 months after completion of the

> preoperational Type A test, a second preoperational Type A test shall be performed prior to initial reactor operations.
>
> The first periodic Type A test shall be performed within 48 months after the successful completion of the last preoperational Type A test. Periodic Type A tests shall be performed at a frequency of at least once per 48 months, until acceptable performance is established in accordance with Section 9.2.3. The interval for testing should begin at initial reactor operation. Each test interval begins upon completion of a Type A test and ends at the start of the next test.
>
> If the test interval ends while primary containment integrity is either not required or it is required solely for shutdown activities, the test interval may be extended indefinitely. However, a successful Type A test shall be completed prior to entering the operating mode requiring primary containment integrity.

While Revision 2 of NEI 94-01 has revised the second paragraph of Section 9.2.2 to provide a clarification, the clarification does not impact this request.

> The first periodic Type A test shall be performed after commencing reactor operation and within 48 months after the successful completion of the last preoperational Type A test. Periodic Type A tests shall be performed at a frequency of at least once per 48 months, until acceptable performance is established in accordance with Section 9.2.3. Each test interval begins upon completion of a Type A test and ends at the start of the next test.

2.3 Reason for Proposed Change

The pre-operational integrated leak rate test (ILRT), or Type A test, was completed for VEGP Unit 3 on July 12, 2020. Initial reactor operation, as defined by imposing containment operability as required by Technical Specification 3.6.1, Containment, was achieved December 8, 2022, i.e., first entry into Mode 4. The result was that initial operation was 29 months after the initial preoperational Type A test. As such, the 36-month criteria in the first paragraph of NEI 94-01 Section 9.2.2 does not require a second pre-operational performance of the Type A test prior to July 2023 pursuant to this criterion.

However, the second paragraph of NEI 94-01 Section 9.2.2 requires the first periodic Type A test be performed within 48 months of the pre-operational Type A test. To meet the 48-month schedule, the first periodic Type A test by July 2024 would require an unscheduled shutdown of Unit 3. Although the pre-operational Type A test was completed July 12, 2020, consistent with guidance provided in NRC Regulatory Issue Summary (RIS) 2008-027, the scheduling and planning of the next Type A test can utilize the month and year. “This means that a licensee who has determined the due date for a Type A test using the stated interval definition gains limited flexibility by being able to commence the test no later than the last day of the month in which it becomes due, without seeking NRC approval.”

With the additional flexibility from the RIS provisions for conducting Type A tests, the 48-month criterion from July of 2020 would require that the first periodic Type A test be performed by the end of July 2024.

Further, Section 9.2.2 of NEI 94-01 states, "If the test interval ends while primary containment integrity is either not required or it is required solely for shutdown activities, the test interval may be extended indefinitely. However, a successful Type A test shall be completed prior to entering the operating mode requiring primary containment integrity." As such, if the first periodic Type A test is not performed by the end of July 2024, the unit must be placed in, and maintained in, an operating mode where containment is not required until the first periodic Typed A test is successfully completed.

At the time of performance of the pre-operational ILRT in July of 2020, the Unit 3 was scheduled for commercial operation in May of 2021, and a refueling outage would have then been scheduled in 2022 during which the first operational ILRT would be performed to meet this 48-month requirement. Due to delays in the startup of Unit 3, commercial operation was delayed until May 2023 with the first refueling outage now scheduled for October 2024, i.e., beyond July of 2024.

With the Unit 3 commercial operation delayed, and the need to achieve the necessary fuel burnup for entering the first refueling outage, it is expected that the Unit 3 first refueling outage will commence after July of 2024. In order to avoid the potential for an unscheduled shutdown of Unit 3, SNC is requesting an extension of the 48-month criterion for the first periodic Type A test schedule until no later than prior to startup following the first refueling outage. This extension is anticipated to be no more than a few months, with completion of the testing or the Unit in Modes 4 or 5, no later than May 31, 2025. This represents a maximum extension of ten months, with flexibility to schedule the refueling outage in the fall of 2024 or the spring of 2025.

2.4 Description of Proposed Change

The proposed change would modify the VEGP Unit 3 COL Appendix A Technical Specification (TS) 5.5.8 to provide a one-time exception to RG 1.163. This would allow (1) for Unit 3 only, the next Type A test performed after the pre-operational Type A test shall be performed a) prior to startup following the first refueling outage, and b) prior to startup following an outage initiated no later than May 31, 2025.

A markup showing this change is provided in the Attachment.

3. TECHNICAL EVALUATION

In July 2020, the initial pre-operational Type A test of the VEGP Unit 3 containment was completed using the ANSI/ANS-56.8-1994, Containment System Leakage Testing Requirements, mass point data analysis technique.

The as-left leakage rate result was found to be acceptable at 0.0285 %wt/day when compared to the acceptance criterion of 0.075 %wt/day (i.e., 75% of the maximum allowable leakage rate, La of 0.1 %wt/day). The design pressure for VEGP Unit 3 is 59.0 psig and the calculated peak accident pressure, Pa, is 58.1 psig. The Type A test pressure (at the end of the hold test) was > 58.46 psig.

Margin: With pre-operational Type A test results of less than 40% of the allowable as-left leakage and less than 71.5% of the maximum allowed leakage, the containment leakage has substantial margin that would not be expected to be exceeded during the requested extension. Acceptable as-left test results of 75% La would provide a 25% La margin for a typical test interval. For the Vogtle Unit 3 first cycle, the margin of 71.5% La provided by the results of the pre-operational test is more than twice the necessary margin at startup of the typical Type A test interval.

In addition, the local leak rate (Type B & C) testing of penetrations has a combined 0.6 La acceptance criteria value of 120,400 sccm. The pre-operational local leak rate tests had a total value of 29,480 sccm, which was < 15% of their overall 1.0 La value (200,667 sccm) or < 25% of 0.6 La. This further contributes to the leak tightness of the containment penetrations and available margin for the Unit 3 containment.

Duration: The requested extension from July 2024 to the startup following the first refueling outage is expected to be, at most, a few months with a requested maximum of ten months, which would most likely be the result of longer than anticipated startup testing duration or capacity factor less than anticipated during the first cycle operation. A Unit 3 extension of no more than a few months will allow the first periodic Type A test to be reasonably scheduled and performed at the first refueling outage, rather than requiring an extended shutdown prior to the refueling outage.

Fuel Burnup: Cycle 1 has a maximum cycle length of 465 effective full power days (EFPD), which is driven by the fuel burnup analysis of the first core design. A cycle length less than 415 EFPD results in poor fuel utilization for cycle 1 fuel or additional unscheduled startups and shutdowns. Due to the large amount of carry-over energy, the core design for cycle 2 requires utilizing off-normal core design strategies which can create additional issues downstream depending on cycle 2 and additional follow-on cycle lengths. If the combined energy of cycle 1 and cycle 2 is below approximately 910 EFPD, some of the cycle 1 assemblies do not receive the necessary burnup to meet the TS 3.7.12 curve for Region 2 spent fuel pool storage. A shortened first cycle has the potential for fuel assemblies to become stranded in Region 1 spent fuel pool storage.

Another fuel related restriction that could be important for a shortened fuel cycle and fuel utilization is the maximum number of discharged fuel assemblies. This in turn limits the maximum number of feed assemblies loaded per cycle. This number is tied to spent fuel pool cooling restrictions, as well as an analysis assumption of 69 fuel assemblies discharged per refueling cycle.

General Risk: NUREG-1493, "Performance-Based Containment Leak-Test Program," was used by the Nuclear Regulatory Commission (NRC) in implementing an initiative to eliminate requirements that are marginal to safety and yet impose a significant regulatory burden on licensees. The containment leakage-testing requirements for power reactors were identified as one area where performance-based requirements could replace the prescriptive requirements with only a marginal impact on safety. NUREG-1493 provided the technical support document (TSD) and included the technical bases for the NRC's rulemaking to revise leakage-testing requirements for nuclear power reactors in 10 CFR Part 50, Appendix J to include Option B.

The NUREG-1493 report identified alternatives to the prescriptive containment testing requirements not found in 10 CFR Appendix J Option A that would meet the NRC's Safety Goals and achieve greater efficiency in the use of resources. The resulting changes (i.e., Option B) in the containment integrated leak rate testing frequencies reduce the integrated leakage-rate tests frequency from three times in 10 years to once in ten years.

As noted in NUREG-1493, Section 1.1, "Technical studies have consistently shown that design basis containment leakage is a relatively minor contributor to reactor accident risk. Reactor accident risk is dominated by accidents in which the containment fails or is bypassed. Therefore, modifying the containment leakage rate and/or test frequency is not expected to have a significant impact on reactor accident risk."

As identified in Section 1.2 of NUREG-1493, an additional Type A test prior to the first refueling outage also has economic and occupational exposure costs. Type A tests by their nature preclude any other reactor maintenance activities and thus are on the critical path for return to service from the outage during which the test is performed. In addition to the cost of the test itself, the additional unscheduled startup and shutdown and the test period would impose a burden of replacement power costs.

More recently, EPRI Report 1009325 demonstrates that, in a generally applicable sense, there is a small risk increase associated with the extension of ILRT intervals from 10 years to 15 years. While this report and NUREG-1493 require evaluation prior to direct application, both show low risk for extensions of Type A testing intervals. As noted in the EPRI report, "Defense-in-depth as well as safety margins are maintained through the continued inspection of containment as required by ASME Section XI, Subsections IWE and IWL, and other required inspections, such as those performed to satisfy the Maintenance Rule."

In summary, the requested extension for Unit 3 to the first refueling outage (with a maximum extension of ten months) for the first periodic Type A test following the pre-operational integrated leakage rate test is not a significant impact on reactor accident probability since containment is not an accident initiator. Additionally, the impact to consequences of an event during the extended operation period would also not be significant due to the margin identified by the pre-operational testing.

4 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.54(o) - Primary reactor containments for water cooled power reactors, other than facilities for which the certifications required under §§ 50.82(a)(1) or 52.110(a)(1) of this chapter have been submitted, shall be subject to the requirements set forth in appendix J to this part.

10 CFR Part 50, Appendix A, General Design Criterion (GDC) 16 — Containment design. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

10 CFR Part 50, Appendix A, GDC 50 — Containment design basis. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

10 CFR Part 50, Appendix J — Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors – One of the conditions required of all operating licenses and combined licenses for light–water–cooled power reactors as specified in § 50.54(o) is that primary reactor containments meet the leakage-rate test requirements in either Option A or B of this appendix. These test requirements ensure that (a) leakage through these containments or systems and components penetrating these containments does not exceed allowable leakage rates specified in the technical specifications; and (b) integrity of the containment structure is maintained during its service life. Option B of this appendix identifies the performance-based requirements and criteria for preoperational and subsequent periodic leakage-rate testing. The footnote to this discussion indicates that specific guidance concerning a performance-based leakage-test program, acceptable leakage-rate test methods, procedures, and analyses that may be used to implement these requirements and criteria are provided in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program."

TS 5.5.8 is based on these regulatory requirements and as noted above, requires that a program be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995," as modified by approved exceptions.

4.2 Precedent

Two pertinent precedents were identified. The first of these is a similar pre-operational test to first post operational test at the first refueling outage but for local leak rate testing.

By letter dated February 16, 2017 (ADAMS Accession No. ML 17048A514) Tennessee Valley Authority (TVA, the licensee) submitted a license amendment request (LAR) for Watts Bar Nuclear Plant (Watts Bar) Unit 2 to allow a one-time interval extension for the local leak rate tests (LLRTs) of a select group of containment isolation valves (CIVs). The proposed amendment revises Technical Specification (TS) 5.7.2.19, "Containment Leakage Rate Testing Program," to add an exception to, or deviation from, the provisions of Nuclear Energy Institute (NEI) 94-01, Revision 0, July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix J," as endorsed by Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. RG 1.163 is identified in TS 5.7.2.19 as the implementation document used to develop the licensee's performance-based leakage-testing program. NEI 94-01 Section 10.2.3.1 states that Type C (CIV) tests shall be performed prior to initial reactor operation and that subsequent periodic Type C tests be performed at a frequency of at least once per 30 months until adequate performance has been established to justify an extended test interval greater than 30 months. A CIV demonstrates adequate performance by not exceeding the administrative leakage limit established by the licensee during two consecutive periodic LLRTs.

After the Watts Bar Unit 2 Operating License was issued on October 22, 2015, TVA began the process of starting up the unit, which included completion of required TS equipment surveillances prior to entry into their associated operating mode of applicability. In addition to completing the surveillance requirements (SRs) required to enter each reactor operating mode, TVA performed power accession testing to confirm that the unit operated as designed. Initial plans for Watts Bar Unit 2 were to declare Unit 2 ready for commercial operation by the end of 2015. With commercial operation originally projected in 2015, TVA planned the first refueling outage to occur in September 2016. Due to delays in the startup of Unit 2, commercial operation was delayed until October 19, 2016.

TVA initially planned to perform Type C LLRTs for the CIVs that are the subject of this amendment during the September 2016 Refueling Outage. When commercial operation was delayed to October 19, 2016, TVA revised the planned date for the first refueling outage to October 2017. To support continuous plant operation until the October 2017 Refueling Outage date, the LAR proposed a one-time change to extend the LLRT intervals for the CIVs that are the subject of this amendment. The licensee indicated in the proposed amendment that these LLRTs cannot be conducted during power operations. The proposed extended date for LLRTs will allow these tests to be performed during the Watts Bar Unit 2 Cycle 1 Refueling Outage, which is scheduled to commence in October 2017. Specifically, the LLRTs for these CIVs will be completed prior to Watts Bar Unit 2 entering Mode 4 following the Cycle 1 Refueling Outage, but no later than December 31, 2017.

With the proposed one-time LLRT interval extensions, a plant shutdown solely to perform these tests would be avoided. If such a mid-cycle outage were to occur, these tests would likely need to be repeated during the first refueling outage to synchronize their performance with the schedule for subsequent refueling outages. The extra cold shutdown (Mode 5) mid-cycle outage would involve an unnecessary transient on the plant and associated risks and additional personnel radiological dose.

The amendment granting this request was issued to TVA on May 18, 2017 (ADAMS Accession No. ML 17123A228).

The second pertinent precedent a similar extension of an integrated leak rate test to the next refueling outage.

By letter dated February 24, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21055A822), Exelon

Generation Company, LLC (Exelon or the licensee) submitted a license amendment request (LAR) for Clinton Power Station, Unit 1 (Clinton or CPS) to allow a one-time extension of the Type A integrated leakage rate testing (ILRT) frequency. The proposed amendment revises Technical Specification (TS) 5.5.13, “Primary Containment Leakage Rate Testing Program,” to add two exceptions:

- An exception to allow the next Type A test to be performed no later than October 31, 2023, which represents an extension of 8 months.
- An exception to allow the Type A test interval to be further extended if the Type A test has not been performed by October 31, 2023, and the unit is in Mode 4 or 5, in which case the Type A test is to be performed prior to entering Mode 2.

The amendment granting this request was issued to Exelon on August 11, 2021 (ADAMS Accession No. ML 21188A020).

4.3 Significant Hazards Consideration

Southern Nuclear Operating Company (SNC) is requesting an amendment to Combined Licenses (COL) No. NPF-91 for Vogtle Electric Generating Plant (VEGP) Unit 3. The license amendment request (LAR) proposes to revise the content of the COL Appendix A, Technical Specification 5.5.8, Containment Leak Rate Testing Program, to include an exception to Regulatory Guide (RG) 1.163, Performance-Based Containment Leak-Test Program. This exception would provide an extension of the first testing interval to allow the first Type A integrated leak rate test (ILRT) following the pre-operational ILRT to be performed prior to startup following the first refueling outage with a maximum extension of ten months. An evaluation to determine whether or not a significant hazards consideration is involved with the proposed amendment was completed by focusing on the three standards set forth in 10 CFR 50.92(c), “Issuance of amendment,” as discussed below.

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

   Response: No

   The proposed changes do not affect accident probability evaluations since there are no changes to the plant, no changes to analysis of the plant, and no changes to testing of the plant, other than scheduling. The proposed changes do not affect the operation of any structures, systems, or components (SSCs) associated with an accident initiator or initiating sequence of events. The proposed changes continue to maintain the initial conditions and operating limits assumed during normal operation, assumed by the accident analysis, and assumed in anticipated operational occurrences. Therefore, the proposed changes do not result in any increase in probability of an analyzed accident occurring.

   The proposed changes do not involve a change to any mitigation sequence or the predicted radiological releases due to postulated accident conditions. With the margin identified by the pre-operational test, the extended interval between Type A tests would also not result in a significant increase to containment leakage that would involve exceeding the consequences. Thus, the consequences of the accidents previously evaluated are not adversely affected.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed changes have been found to continue to provide the required functional capability of the safety systems for previously evaluated accidents and anticipated operational occurrences. The proposed revisions do not change the function of the related systems, and thus, the changes do not introduce a new failure mode, malfunction or sequence of events that could adversely affect safety or safety-related equipment.

   Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   The proposed changes continue to provide the required functional capability of the safety systems for previously evaluated accidents and anticipated operational occurrences. The proposed changes do not change the function of the related systems nor significantly affect the margins provided by the systems. No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the requested changes. Reactor accident risk is dominated by accidents in which the containment fails or is bypassed. Therefore, modifying the containment leakage rate test frequency is not expected to have a significant impact on reactor accident risk.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, it is concluded that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

4.4 Conclusions

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. Therefore, it is concluded that the requested amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

5 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed changes require an amendment to the COL. A review of the anticipated construction and operational effects of the requested amendment has determined that the requested amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), in that:

(i) There is no significant hazards consideration.

As documented in Section 4.3, Significant Hazards Consideration, of this license amendment request, an evaluation was completed to determine whether or not a significant hazards consideration is involved by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment.” The Significant Hazards Consideration evaluation determined that (1) the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) the proposed amendment does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of “no significant hazards consideration” is justified.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed changes are unrelated to any aspect of plant operation that would introduce any change to effluent types (e.g., effluents containing chemicals or biocides, sanitary system effluents, and other effluents) or affect any plant radiological or non-radiological effluent release quantities. Furthermore, the proposed changes do not affect any effluent release path or diminish the functionality of any design or operational features that are credited with controlling the release of effluents during plant operation. Therefore, it is concluded that the proposed amendment does not involve a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change in the requested amendment does not affect the shielding capability of, or alter any walls, floors, or other structures that provide shielding. Plant radiation zones and controls under 10 CFR 20 preclude a significant increase in occupational radiation exposure. Therefore, the proposed amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above review of the proposed amendment, it has been determined that anticipated operational effects of the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

Licensing Basis Document Mark-ups for License Amendment Request

5.5.8 Containment Leakage Rate Testing Program

a. 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-Test Program, dated September 1995," as modified by ~~approved~~ the following exception~~s~~: (1) for Unit 3 only, the next Type A test performed after the pre-operational Type A test shall be performed a) prior to startup following the first refueling outage and b) prior to startup following an outage initiated no later than May 31, 2025.