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10 CFR 50.90

RS-04-004

January 15, 2004

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

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Subject: Request for Amendment to Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program"

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. DPR-19 and DPR-25 for the Dresden Nuclear Power Station (DNPS) Units 2 and 3. The proposed change revises Technical Specification (TS) Section 5.5.12, "Primary Containment Leakage Rate Testing Program," to reflect a one-time deferral of the primary containment Type A test to no later than February 27, 2011, for Unit 2, and no later than July 13, 2009, for Unit 3.

TS Section 5.5.12 provides the requirements for the Primary Containment Leakage Rate Testing Program. TS Section 5.5.12.a requires that this program establish the leakage testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. Additionally, the testing is required to conform to the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

A plant-specific, risk-based evaluation has been performed in support of the one-time deferral of the Type A test frequency from once in 10 years to once in 15 years. The evaluation demonstrates that a change in the Type A test interval from 10 years to 15 years represents a very small impact on risk, as defined by NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, dated November 2002.

This request is subdivided as follows.

- Attachment 1 provides an evaluation supporting the proposed change.

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- Attachment 2 contains the marked-up TS page with the proposed change indicated.
- Attachment 3 provides retyped TS pages with the proposed change incorporated.
- Attachment 4 provides the risk assessment supporting the proposed change.

The proposed changes have been reviewed by the DNPS Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

Approval of the proposed changes will result in significant cost savings. The Type A test imposes significant expense on EGC while the safety benefit of performing the Type A test within 10 years, versus 15 years, is minimal. The proposed changes are similar to changes approved by the NRC for Susquehanna Steam Electric Station on March 8, 2002, Seabrook Station on April 11, 2002, and the Brunswick Steam Electric Plant Units 1 and 2 on March 6, 2002, and November 21, 2002, respectively.

The next Type A test for DNPS is currently scheduled to be performed during the Unit 3 outage in November 2004. To support incorporation of the Type A testing changes into the schedule for the upcoming DNPS Unit 3 refuel outage, EGC requests approval of the proposed amendments by October 15, 2004. Once approved, the amendments shall be implemented within 60 days. This implementation period will provide adequate time for station documents to be revised using the appropriate change control mechanisms.

In accordance with 10 CFR 50.91(b), EGC is notifying the State of Illinois of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions related to this letter, please contact
Mr. Kenneth M. Nicely at (630) 657-2803.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 15th day of January 2004.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing

Attachments:

Attachment 1: Evaluation of Proposed Change
Attachment 2: Markup of Proposed Technical Specifications Page Change
Attachment 3: Retyped Technical Specifications Pages for Proposed Change
Attachment 4: ERIN Report No. C467030603-5572, "Dresden Risk Assessment to Support ILRT (Type A) Interval Extension Request," September 2003

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Dresden Nuclear Power Station
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 1
Evaluation of Proposed Change

ATTACHMENT 1
Evaluation of Proposed Change

1.0 INTRODUCTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. DPR-19 and DPR-25 for the Dresden Nuclear Power Station (DNPS) Units 2 and 3. The proposed change revises Technical Specifications (TS) Section 5.5.12, "Primary Containment Leakage Rate Testing Program," to reflect a one-time deferral of the primary containment Type A test to no later than February 27, 2011, for Unit 2, and no later than July 13, 2009, for Unit 3.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed change involves a one-time exception to the 10 year frequency of the performance-based leakage rate testing program for Type A tests as required by Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0 (Reference 1). Specifically, the proposed change revises TS Section 5.5.12 of the DNPS Units 2 and 3 TS to add the following statement:

", as modified by the following exceptions:

1. NEI 94-01 – 1995, Section 9.2.3: The first Unit 2 Type A test performed after the February 28, 1996, Type A test shall be performed no later than February 27, 2011.

2. NEI 94-01 – 1995, Section 9.2.3: The first Unit 3 Type A test performed after the July 14, 1994, Type A test shall be performed no later than July 13, 2009."

Attachment 2 provides a TS page markup indicating the proposed change. Attachment 3 provides the retyped TS pages incorporating the proposed change.

3.0 BACKGROUND

DNPS Units 2 and 3 are General Electric BWR/3 plants with Mark I primary containments. The Mark I primary containment consists of a drywell, which encloses the reactor vessel, reactor coolant recirculation system, and branch lines of the reactor coolant system (RCS); a toroidal-shaped pressure suppression chamber containing a large volume of water; and a vent system connecting the drywell to the water space of the suppression chamber. The primary containment is penetrated by access, piping, and electrical penetrations.

The integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak-tight integrity of the primary containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify the essentially leak-tight characteristics of the primary containment at the design basis accident pressure.

Revisions to 10 CFR 50, Appendix J (i.e., Option B) allow individual plants to extend Type A ILRT surveillance testing frequency from three-in-ten years to at least once per 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 is less than the maximum allowable primary containment leakage rate. In Reference 2, NRC approval to implement Option B was requested. The NRC subsequently approved implementation of Option B in Reference 3.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01. This document was established in 1995 during development of the performance-based Option B. Section 11.0 states that NUREG-1493, "Performance-Based Containment Leak Test Program," dated September 1995 (Reference 4), provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B of 10 CFR 50, 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 Electrical Power Research Institute (EPRI) Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals" (Reference 5).

Option B of 10 CFR 50, Appendix J, requires that a Type A test be conducted at a periodic interval based on historical performance of the overall primary containment system. DNPS TS Section 5.5.12 provides the requirements for the Primary Containment Leakage Rate Testing Program. TS Section 5.5.12.a requires that this program establish the leakage testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. Additionally, the testing is required to conform to the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995 (Reference 6). Reference 6 endorses, with certain exceptions, NEI 94-01.

NEI 94-01 specifies for Type A tests, an initial test interval of 48 months and allows an extension of the interval to 10 years, based on two consecutive successful tests. DNPS Units 2 and 3 are currently on 10-year testing intervals.

The proposed change adds two exceptions to TS Section 5.5.12 to allow a one-time deferral from the guidelines contained in Regulatory Guide 1.163 and NEI 94-01 regarding the Type A test interval. The proposed change will extend the next Type A test for Units 2 and 3 to a 15-year interval.

The last Type A test for Unit 2 was performed on February 28, 1996, and the last Type A test for Unit 3 was performed on July 14, 1994. The proposed change will require the next Type A tests be performed by February 27, 2011, for Unit 2, and by July 13, 2009, for Unit 3.

4.0 REGULATORY REQUIREMENTS & GUIDANCE

10 CFR 50.36, "Technical specifications," provides the regulatory requirements for the content of a licensee's TS.

10 CFR 50, Appendix J, Section V.B, "Implementation," specifies that the regulatory guide or other implementing documents used to develop a performance-based leakage testing program must be included, by general reference, in the plant TS. Additionally, deviations from guidelines endorsed in a regulatory guide are to be submitted as a revision to the plant TS.

5.0 TECHNICAL ANALYSIS

5.1 10 CFR 50, Appendix J, Option B

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the primary containment, including systems and components that penetrate the primary containment, does not exceed allowable leakage rate values specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumptions in the safety analyses are not exceeded. The limitation of primary containment leakage provides assurance that the primary containment would perform its design function following an accident, up to and including the design basis accident.

10 CFR 50, Appendix J, was revised effective October 26, 1995. The purpose of this revision was to allow licensees to choose primary containment leakage testing under Option A "Prescriptive Requirements" or Option B. Amendment Nos. 148 and 142 for Units 2 and 3 (Reference 3) were issued to permit implementation of 10 CFR 50, Appendix J, Option B. TS Section 5.5.12 currently requires the establishment of a Primary Containment Leakage Rate Testing Program in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program implements the guidelines contained in Regulatory Guide 1.163, which specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01, subject to several regulatory positions stated in the regulatory guide.

Deviations from Regulatory Guide 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 primary containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed or its acceptance criteria; 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 and as-left leakage 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. NUREG-1493 made the following observations with regard to decreasing the test frequency.

- Reducing the Type A testing frequency to once per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is small because Type A tests 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 only

been marginally above the existing requirements. Given the insensitivity of risk to primary containment leakage rate, and the small fraction of leakage detected solely by Type A testing, increasing the interval between Type A testing has minimal impact on public risk.

- While Type B and C tests identify the vast majority (i.e., 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 lease once per 10 years based upon an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart or refueling cycles where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3.

5.2 DNPS Integrated Leak Rate Testing 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 detects a large percentage of containment leakages and that the percentage of containment leakages detected only by integrated containment leakage testing is very small. Results of the previous Type A tests for each unit, presented below, demonstrate the DNPS Units 2 and 3 containment structures remain essentially leak-tight barriers and represent minimal risk to increased leakage. These plant specific results support the conclusions of NUREG-1493.

Unit	Test Date	Total Leakage*	Fraction of L_a
2	March 1985	0.4732	0.2957
2	December 1986	0.6366	0.3979
2	December 1990	0.7428	0.4642
2	May 1993	0.8184	0.5115
2	February 1996	0.3380	0.2112
3	March 1988	0.4800	0.300
3	February 1990	0.7695	0.4809
3	March 1992	0.5546	0.3466
3	July 1994	0.65405	0.4088

* Leakage rates are expressed in units of percent containment air weight per day. The maximum allowable primary containment leakage rate, L_a, 1.6% of primary containment air weight per day. TS leakage rate acceptance criteria for a Type A test for unit startup is $0.75L_a$ (i.e., 1.2% containment air weight per day), as

discussed in TS Section 5.5.12. Calculated results are expressed at a 95% confidence level.

5.3 Type B and C Testing

Type B and C testing assures containment penetrations such as flanges, sealing mechanisms, and containment isolation valves are essentially leak-tight. Type B and C tests identify the vast majority of all potential leakage paths. The Type B and C testing requirements will not be changed as a result of the extended ILRT interval.

5.4 Containment Inspections

a. Appendix J Visual Inspections

The Appendix J program requires visual inspections to be performed of accessible interior and exterior surfaces of the containment system for structural problems that may affect either the containment structural leakage integrity or performance of the Type A test. These examinations are conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test, based on a 10-year Type A test frequency (Reference 6).

These requirements will not be changed as a result of the extended ILRT interval.

b. Containment Inservice Inspection Program

A comprehensive primary containment inspection is performed to the requirements of American Society of Mechanical Engineers (ASME) Section XI, "Inservice Inspection," Subsections IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements of Class CC Concrete Components of Light-Water Cooled Power Plants." The DNPS Containment Inservice Inspection (CISI) plan was developed in accordance with the requirements of the 1998 Edition of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsection IWE, as modified by NRC final rulemaking to 10 CFR 50.55a published in the Federal Register on August 8, 1996; and NRC-approved relief requests (i.e., DNPS relief request MCR-02). In Reference 7, the NRC authorized use of the 1998 Edition of the ASME Code for the DNPS CISI program as an alternative to the 1992 Edition and 1992 Addenda of the ASME Code, which is incorporated by reference in 10 CFR 50.55a.

The components subject to Subsection IWE and IWL requirements are those which make up the containment structure, its leak-tight barrier (including integral attachments), and those that contribute to its structural integrity. Specifically included are Class MC pressure retaining components, including metallic shell and penetration liners of Class CC pressure retaining components, and their integral attachments. DNPS has no Class CC components which meet the criteria of IWL-1100; therefore, DNPS is not required to have acceptance criteria for concrete and reinforcing bar degradation, and no requirements to perform

examinations in accordance with Subsection IWL are incorporated into the DNPS CISI plan.

The DNPS CISI Program consists of a CISI Basis Document, Program Plan, and CISI drawings.

- The Basis Document defines the scope of accessible and inaccessible areas and components.
- The Program Plan contains information such as the inspection schedules and program relief requests.
- The CISI drawings provide information such as the type of components, their as-built details, and the associated penetrations.

Metal components included in the scope of the CISI Program were identified by evaluating identified metal components to the criteria of IWE-1100, and making a determination to include (or exclude) the component. A component specific basis for accessibility or inaccessibility of metal components for inspection was established in accordance with ASME Code, Section XI, Subarticle IWE-1100, Subsubarticle IWE-1230.

The first interval of the DNPS CISI Program is effective from September 10, 1996, through September 9, 2008, for both DNPS units. DNPS is currently in the second period of the first CISI interval, and the Unit 2 and Unit 3 CISI inspections for the first period have been completed. The second period of the first CISI interval is scheduled to end on September 9, 2005.

For CISI inspections performed, various indications were observed, documented, evaluated, and determined to be acceptable. No areas of the containment liner surfaces require augmented examination, and no loss of structural integrity of primary containment was observed. No significant degradation of containment for either DNPS unit has been identified since implementation of the CISI Program visual inspections in 1996.

Since implementing the CISI Program in 1996, the only indications noted during inspections that did not meet acceptance criteria were related to the Units 2 and 3 drywell moisture barriers. Specifically, the drywell moisture barriers experienced age-related degradation and, as required by 10 CFR 50.55a(b)(2)(ix)(D)(1) and IWA 6000, these indications were reported to the NRC in the Inservice Inspection (ISI) Summary Reports for outages D2R17and D3R17 (i.e., References 8 and 9, respectively). The most recent Unit 2 outage (i.e., D2R18) was just concluded in November 2003, and its ISI Summary Report is being prepared at this time.

The corrective actions for the degraded drywell moisture barriers included:

- Removal of the defective drywell moisture barriers;
- Visual inspection of metal containment surfaces covered by and adjacent to the moisture barriers (no unacceptable degradation was noted); and

- Installation and visual inspection of new drywell moisture barriers (new drywell moisture barriers were found acceptable).

Beyond the moisture barriers noted above, no indications failed to meet acceptance criteria during the last containment visual examinations for each unit. ISI Summary Reports for the most recent outage for Unit 3 (i.e., D3R17) was provided to the NRC in Reference 9.

There will be no change to the schedule for these inspections as a result of the extended ILRT interval.

c. Containment Coatings Inspections

A program to maintain containment coatings was developed to meet the guidelines of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," Revision 0 (Reference 10). Each refueling outage, a preventive maintenance activity to inspect the protective coatings in the containment is performed. The most recent inspections for Units 2 and 3 were performed in October 2003, and October 2002, respectively. There have been minor issues noted (e.g., coating peeling); however, overall the containment coatings are in an acceptable condition.

The inspection requirements of the containment coatings program will not be changed as a result of the extended ILRT interval.

5.5 Containment Pressure Monitoring

The drywell and suppression chamber comprise the primary containment volume at DNPS (i.e., Mark I containment design). During power operation, the primary containment atmosphere is inerted with nitrogen to mitigate concerns with hydrogen production following certain design basis accidents. TS Section 3.6.2.5, "Drywell-to-Suppression Chamber Differential Pressure," requires that containment drywell pressure be maintained at a positive pressure relative to containment suppression chamber pressure (i.e., ≥1.0 psid differential pressure). The TS also require that drywell pressure be maintained ≤1.5 psig. The Containment Atmospheric Control System provides a supply of makeup nitrogen to automatically maintain primary containment pressure within the TS limits. Drywell and suppression chamber pressure are continuously recorded in the control room. The control room operators monitor drywell pressure and drywell to suppression chamber differential pressure via shiftly surveillances. Additionally, drywell high or low pressure annunciators in the control room alert the operators to off normal conditions. The containment pressure monitoring described above provide indications of changes to containment leakage, and no changes to these monitoring requirements will be made as a result of the extended ILRT interval.

5.6 Major Modifications to Containment

DNPS is committed to the requirements of IWE-5200 from the 1998 Edition of ASME Section XI and, as such, is required to perform a leakage test following a major

modification or replacement of component(s). There are no major modifications or replacement of component(s) planned during the requested ILRT interval extension period. However, if a major modification or replacement of component(s) were to occur during the extension period, performance of a leakage test would be required to satisfy the requirements of IWE-5200.

5.7 Information Notice 92-20

Information Notice 92-20, "Inadequate Local Leak Rate Testing," (Reference 11) was issued to alert licensees of problems with local leak rate testing of two-ply stainless steel bellows. The information notice discusses an event at Quad Cities Nuclear Power Station (QCNPS) Unit 1 where a Type B test on the containment penetration bellows could identify leakage, but could not accurately quantify the extent of the leakage.

In Reference 12, the NRC granted an exemption from certain Type B testing requirements of 10 CFR 50, Appendix J, for the two-ply containment penetration expansion bellows at DNPS and QCNPS. The exemption was needed because the bellows design is such that they cannot be properly tested to satisfy Type B testing requirements, barring replacement with bellows of a different design.

The exemption specifies an alternative program of bellows testing and replacement that involves testing with air at a reduced leakage limit, testing any leaking bellows with helium (i.e., sniffer testing), replacing bellows that are unacceptable, and performing a Type A test each refueling outage until all of the bellows have been replaced with testable bellows. This testing program is intended to assure that at least one ply of a two-ply bellows is intact and that overall containment leakage is within its allowable limit as shown by Type A testing. Reference 12 stated that the Type A test is essential to this program because it is the only test available that can properly quantify the bellows' leakages, albeit not individually. This is especially important for those bellows which are known to leak but will not be replaced until after another cycle.

As stated in Reference 12, the NRC found that the proposed testing program will detect bellows assemblies with significant flaws and result in replacement of flawed assemblies within one operating cycle, during which period there is reasonable assurance that the bellows assemblies will not suffer excessive degradation.

In Reference 13, Commonwealth Edison Company requested a revision to the exemption granted in Reference 12. The revised exemption would delete the requirement to perform a Type A test each refueling outage based on alternative Type B tests developed, since the original exemption was issued, to determine the leakage from the two-ply containment penetration expansion bellows. These alternative tests can be applied to ensure the intent of requiring a Type A test, as part of the original exemption, is met. As stated in Reference 13, the requirement to perform a Type A test every outage is not necessary to ensure that the bellows assemblies are adequately tested and leakage from any leaking bellows assembly is adequately quantified. This position was developed based upon the following insights gained during testing of two-ply bellows:

- there is minimal probability for the occurrence of a large leak in a two-ply bellows;

- the special testing program is effective for identifying small leaks in two-ply bellows;
- the Type A test is ineffective for identifying small leaks in two-ply bellows; and
- more cost effective alternative methods have been developed for quantifying leakage.

For a two-ply bellows that leaks through both plies, the revised exemption allows: (1) a valid Type B test using one of the alternative tests to ensure compliance to license limits, or (2) a Type A test as required in the original exemption and, before the return to power in a subsequent refuel outage, replacement of the bellows with a testable bellows assembly or a valid Type B test to ensure license limits are met.

In Reference 14, the NRC granted the revised exemption. As stated in Reference 14, the NRC found that the underlying purpose of the regulation will be met in that the proposed testing program will detect bellows assemblies with significant flaws and result in replacement of flawed assemblies within one operating cycle, or be tested with a Type B test to ensure license limits are met during which period there is reasonable assurance that the bellows assemblies will not suffer excessive degradation.

The proposed change to extend the Type A test frequency from once in 10 years to once in 15 years does not affect the conclusions documented in References 12 and 14. The NRC-approved testing program will continue to detect bellows assemblies with significant flaws and result in replacement of flawed assemblies within one operating cycle, or be tested with a Type B test to ensure license limits are met during which period there is reasonable assurance that the bellows assemblies will not suffer excessive degradation.

5.8 Risk Information

A plant-specific, risk-based evaluation has been performed in support of the one-time deferral of the Type A test frequency from once in 10 years to once in 15 years. The risk analysis is contained in Attachment 4. The 2002A DNPS Unit 2 full power internal events Level 1 and 2 probabilistic safety assessment (PSA) model was used as input to this analysis and is characteristic of the as-built, as-operated plant. There are no substantive differences between Unit 2 and Unit 3 that are judged to affect the conclusions of the PSA. As such, no separate PSA quantification was conducted for Unit 3. Since the PSA is judged applicable to both Units 2 and 3, the ILRT interval extension risk evaluation is applicable to both units.

The risk analysis determined that the proposed change results in the following.

- Increasing the current 10 year ILRT interval to 15 years results in a negligible increase in total population dose rate of 0.002 person-rem/year.
- The increase in the large early release frequency (LERF) risk measure is also insignificant, an approximately 8E-9/year increase. This LERF increase is categorized as a "very small" increase per NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on

Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002 (Reference 15).

- Likewise, the conditional containment failure probability (CCFP) increases insignificantly by 0.4%.

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency below 10^{-6}/year and increases in LERF below 10^{-7}/year. Since the ILRT does not impact core damage frequency, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from once per 10 years to once per 15 years, using the change in the EPRI Category 3b frequency per the NEI Interim Guidance (Reference 16), is approximately 8E-9/year. Regulatory Guide 1.174 defines very small changes in LERF as below 10^{-7}/year. Therefore, increasing the DNPS ILRT interval from 10 years to 15 years results in a very small change in risk, and is an acceptable plant change from a risk perspective.

The change in CCFP is also calculated as an additional risk measure to demonstrate the impact on defense-in-depth. The change in CCFP is found to be very small (i.e., 0.4% increase) and represents a negligible change in the DNPS defense-in-depth.

5.9 Conclusion

Based on the above, the proposed change to TS Section 5.5.12 will continue to provide assurance that leakage through the DNPS primary containment will not exceed allowable leakage rate values specified in the TS and Bases, and that the containment features will continue to perform their design function following an accident, up to and including the design basis accident.

6.0 REGULATORY ANALYSIS

10 CFR 50.36 provides the regulatory requirements for the content required in a licensee's TS. 10 CFR 50.36(c)(5), "Administrative controls," requires provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner will be included in a licensee's TS.

Additionally, 10 CFR 50, Appendix J, Section V.B, specifies that the regulatory guide or other implementing documents used to develop a performance-based leakage testing program must be included, by general reference, in the plant's TS. Additionally, deviations from guidelines endorsed in a regulatory guide are to be submitted as a revision to the plant's TS.

The proposed change will revise TS Section 5.5.12 to reflect a one-time deferral from the program requirements for the Type A test for DNPS Units 2 and 3. The deferral represents an exception to the guidelines contained in Regulatory Guide 1.163 and NEI 94-01. Thus, the proposed change is consistent with the requirements of 10 CFR 50.36(c)(5) and 10 CFR 50, Appendix J, Section V.B.

Additionally, in accordance with 10 CFR 50, Appendix J, Section V.B, the proposed changes to DNPS TS do not require a supporting request for an exemption to Option B of Appendix J, in accordance with 10 CFR 50.12, "Specific exemptions."

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license 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.

EGC has evaluated the proposed change to the TS for DNPS Units 2 and 3 using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

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

Response: No

The proposed change will revise Dresden Nuclear Power Station (DNPS) Units 2 and 3 Technical Specifications (TS) Section 5.5.12, "Primary Containment Leakage Rate Testing Program," to reflect a one-time deferral of the primary containment Type A test to no later than February 27, 2011, for Unit 2, and no later than July 13, 2009, for Unit 3. The current Type A test interval of 10 years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test.

The function of the primary containment is to isolate and contain fission products released from the reactor coolant system (RCS) following a design basis loss-of-coolant accident (LOCA) and to confine the postulated release of radioactive material to within limits. The test interval associated with Type A testing is not a precursor of any accident previously evaluated. Therefore, extending this test interval on a one-time basis from 10 years to 15 years does not result in an increase in the probability of occurrence of an accident. The successful performance history of Type A testing provides assurance that the DNPS primary containments will not exceed allowable leakage rate values specified in the TS and will continue to perform their design function following an accident. The risk assessment of the proposed change has concluded that there is an insignificant increase in total population dose rate and an insignificant increase in the conditional containment failure probability.

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

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

   Response: No

   The proposed change for a one-time extension of the Type A tests for DNPS Units 2 and 3 will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed change does not introduce any new equipment or modes of system operation. No installed equipment will be operated in a new or different manner. As such, no new failure mechanisms are introduced.

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

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

   Response: No

   DNPS Units 2 and 3 are General Electric BWR/3 plants with Mark I primary containments. The Mark I primary containment consists of a drywell, which encloses the reactor vessel, reactor coolant recirculation system, and branch lines of the RCS; a toroidal-shaped pressure suppression chamber containing a large volume of water; and a vent system connecting the drywell to the water space of the suppression chamber. The primary containment is penetrated by access, piping, and electrical penetrations.

   The integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak-tight integrity of the primary containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify the essentially leak-tight characteristics of the primary containment at the design basis accident pressure. The proposed change for a one-time extension of the Type A tests do not affect the method for Type A, B, or C testing, or the test acceptance criteria. In addition, based on previous Type A testing results, EGC does not expect additional degradation, during the extended period between Type A tests, which would result in a significant reduction in a margin of safety.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

8.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or would change an inspection or surveillance requirement. However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," Paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, Paragraph (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

9.0 PRECEDENT

The proposed amendment incorporates into the DNPS TS changes that are similar to changes approved by the NRC for Susquehanna Steam Electric Station on March 8, 2002, Seabrook Station on April 11, 2002, and the Brunswick Steam Electric Plant Units 1 and 2 on March 6, 2002, and November 21, 2002, respectively.

10.0 IMPACT ON PREVIOUS SUBMITTALS

EGC has reviewed the proposed change for impact on previous submittals awaiting NRC approval for DNPS, and has determined that there is an impact to a submittal dated October 10, 2002 (Reference 17). The change proposed in Reference 17 revises TS page 5.5-12 to increase the maximum allowable primary containment leakage rate specified in TS 5.5.12.c from 1.6% of primary containment air weight per day to 3% of primary containment air weight per day. The retyped TS pages provided in Attachment 3 of this submittal **do not** reflect the change proposed in Reference 17.

A sensitivity study was performed to evaluate the impact of the proposed increase in maximum allowable primary containment leakage rate to 3% of primary containment air weight per day. This change in TS containment leakage rate would affect the dose rates for EPRI Categories 1, 3a, and 3b. The results of the sensitivity study are documented in Section 3.5.3 of Attachment 4. The results conclude that if the leakage rate were 3%, then increasing the DNPS ILRT interval from 10 years to 15 years would result in a total population dose rate increase of approximately 0.01 person-rem/year. This increase is negligible compared to the total population dose rate of 10.3 person-rem/year.

11.0 REFERENCES

1. Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, July 26, 1995

2. Letter from P. L. Piet (Commonwealth Edison Company) to U. S. NRC, "Request for Amendment to Facility Operating Licenses NPF-11, NPF-18, DPR-19, DPR-25, DPR-29 and DPR-30, Appendix A, Technical Specifications, Incorporation of Option B to 10CFR50, Appendix J," November 14, 1995

3. Letter from J. F. Stang (U. S. NRC) to D. L. Farrar (Commonwealth Edison Company), "Issuance of Amendments Related to 10 CFR Part 50, Appendix J, Option B (TAC Nos. M94061, M94062, M94065, and M94066)," January 11, 1996

4. NUREG 1493, "Performance-Based Containment Leak-Test Program," September 1995

5. Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," August 1994

6. NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995

7. Letter from A. J. Mendiola (U. S. NRC) to O. D. Kingsley (Commonwealth Edison Company), "Byron, Dresden and LaSalle – Evaluation of Relief Requests: Use of 1998 Edition of Subsections IWE and IWL of the ASME Code for Containment Inspection (TAC Nos. MA8933, MA8934, MA8935, MA8936, MA8937 AND MA8938)," dated September 18, 2000

8. Letter from P. Swafford (Exelon Generating Company) to U. S. NRC, "Inservice Inspection (ISI) Summary Report Fall 2001 Inservice Inspection Period," dated February 5, 2002

9. Letter from R. J. Hovey (Exelon Generation Company) to U. S. NRC, "Inservice Inspection (ISI) Summary Report Fall 2002 Inservice Inspection Period," dated January 24, 2003

10. NRC Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," June 1973

11. NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing," March 3, 1992

12. Letter from B. A. Boger (U. S. NRC) to T. J. Kovach (Commonwealth Edison Company), "Exemption from the Testing Requirements of Appendix J to 10 CFR Part 50 for Dresden and Quad Cities Nuclear Power Stations (TAC Nos. M81299, M81300, M81301, and M81302)," February 6, 1992

13. Letter from J. L. Schrage (Commonwealth Edison Company) to W. T. Russell (U. S. NRC), "Request to Revise Exemption from 10CFR50 Appendix J Type B Testing Requirement for Two-Ply Containment Penetration Bellows," October 4, 1994

14. Letter from R. M. Pulsifer (U. S. NRC) to D. L. Farrar (Commonwealth Edison Company), "Revision to Exemption from Appendix J to 10 CFR Part 50 for Quad Cities,

Units 1 and 2, and Dresden, Units 2 and 3 (TAC Nos. M90628, M90629, M90630 and M90631)," February 9, 1995

15. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002

16. Letter from A. Petrangelo (NEI) to NEI Administrative Points of Contact, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Surveillance Intervals," November 13, 2001

17. Letter from K. R. Jury (Exelon Generation Company) to U. S. NRC, "Request for License Amendments Related to Application of Alternative Source Term," October 10, 2002

ATTACHMENT 2
Markup of Proposed Technical Specifications Page Change

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

REVISED TECHNICAL SPECIFICATIONS PAGE

5.5-11

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

b. A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems described in b.1 and b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

a. This program shall establish the leakage testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995. INSERT

b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 43.9 psig.

(continued)

Insert for Technical Specification 5.5.12

, as modified by the following exceptions:

1. NEI 94-01 – 1995, Section 9.2.3: The first Unit 2 Type A test performed after the February 28, 1996, Type A test shall be performed no later than February 27, 2011.

2. NEI 94-01 – 1995, Section 9.2.3: The first Unit 3 Type A test performed after the July 14, 1994, Type A test shall be performed no later than July 13, 2009.

ATTACHMENT 3
Retyped Technical Specifications Pages for Proposed Change

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

REVISED TECHNICAL SPECIFICATIONS PAGES

5.5-11
5.5-12

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

b. A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems described in b.1 and b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

a. This program shall establish the leakage testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. 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. NEI 94-01 - 1995, Section 9.2.3: The first Unit 2 Type A test performed after the February 28, 1996, Type A test shall be performed no later than February 27, 2011.

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

2. NEI 94-01 - 1995, Section 9.2.3: The first Unit 3 Type A test performed after the July 14, 1994, Type A test shall be performed no later than July 13, 2009.

b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 43.9 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, is 1.6% of primary containment air weight per day.

d. Leakage rate acceptance criteria are:

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

2. Air lock testing acceptance criteria is the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

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

ATTACHMENT 4

ERIN Report No. C467030603-5572, "Dresden Risk Assessment to Support ILRT (Type A) Interval Extension Request," September 2003

RM Documentation Approval

RM DOCUMENTATION NO. SA-1214 **REV: 0** **PAGE NO. 1**

STATION: Dresden

UNIT(S) AFFECTED: Units 2 and 3

TITLE: Dresden ILRT Risk Assessment

SUMMARY (Include UREs incorporated):

This Support Application (SA) documents the Dresden ILRT Risk Assessment for a one time extension of the Dresden Integrated Leak Rate Test (ILRT) interval from the currently approved once in 10 years to once in 15 years.

The results demonstrate that the proposed change in the ILRT interval represents a "very small" impact on risk as defined by Reg. Guide 1.174.

Internal RM Documentation

Electronic Calculation Data Files: See body of the documentation.

Prepared by: Grant A. Teagarden / [signature] / 9-19-03
Print Sign Date

Reviewed by: John E. Steinmetz / [signature] / 9-23-03
Print Sign Date

Method of Review: [X] Detailed [ ] Alternate

This RM documentation supersedes: _____ **in its entirety.**

Approved by: E. T. Burns / [signature] / 9-23-03.
Print Sign Date

External RM Documentation

Reviewed by: _____ / _____ / _____
Print Sign Date

Approved by: _____ / _____ / _____
Print Sign Date

Do any ASSUMPTIONS / ENGINEERING JUDGEMENTS require later verification? [ ] Yes [X] No

Tracked By: AT#, URE# etc.) _____