October 4, 2001

Mr. Oliver D. Kingsley, President
Exelon Nuclear
Exelon Generation Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: EXTENSION OF THE CONTAINMENT INTEGRATED LEAK RATE TEST (TAC NO. MB2094)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment No. 244 to Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station, Unit 3. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 30, 2001 (two letters), as supplemented by two letters dated July 24, 2001, and a letter dated August 13, 2001.

This amendment revises TS 5.5.12 to allow a one-time change in the containment integrated leak rate test (ILRT) interval from the current 10 years to a test interval of 15 years. The ILRT is required by Title 10 of the *Code of Federal Regulations*, Part 50, Appendix J, Type A test.

A copy of the safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly *Federal Register* Notice.

Sincerely,

/RA/

John P. Boska, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-278

Enclosures: 1. Amendment No. 244 to DPR-56
2. Safety Evaluation

cc w/encls: See next page

Peach Bottom Atomic Power Station
Units 2 and 3

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John P. Boska, Project Manager, Section 2
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cc w/encls: See next page

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MO'Brien	GHubbard	MRubin	DTerao
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EAdensam	HAshar		

ADAMS ACCESSION NUMBER: ML012210108
* SE provided, no major changes made.

OFFICE	PM/PD1-2	LA/PD1-2	SC/SPLB	SC/SPSB	SC/EMEB
NAME	JBoska	MO'Brien	See SE dated*	See SE dated*	See SE dated*
DATE	9/13/01	9/14/01	9/10/01	9/10/01	8/29/01

OFFICE	SC/RTSB	OGC	SC/PD1-2		
NAME	RDennig	SUttal	JClifford		
DATE	9/14/01	9/24/01	10/3/01		

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EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 244
License No. DPR-56

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), PSEG Nuclear LLC, and Atlantic City Electric Company (the licensees) dated May 30, 2001 (two letters), as supplemented by two letters dated July 24, 2001, and a letter dated August 13, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I.

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-56 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 244, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 4, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 244

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
5.0-17	5.0-17

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 244 TO FACILITY OPERATING

LICENSE NO. DPR-56

EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

ATLANTIC CITY ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

DOCKET NO. 50-278

1.0 INTRODUCTION

By letter dated May 30, 2001 (two letters), as supplemented by two letters dated July 24, 2001, and a letter dated August 13, 2001, Exelon Generation Company, LLC (the licensee) submitted a request for changes to the Peach Bottom Atomic Power Station (PBAPS) Unit 3, Technical Specifications (TSs). The requested change would revise TS 5.5.12 to allow a one-time change in the containment integrated leak rate test (ILRT) interval from the current 10 years to a test interval of 15 years. The ILRT is required by Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix J, Type A test. For the initial submittal on May 30, 2001, one letter contained the safety assessment and the information supporting a determination of no significant hazards, while the second letter contained the performance based, risk-informed analysis. The July 24, 2001 (two letters), and August 13, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original *Federal Register* notice.

2.0 BACKGROUND

PBAPS Unit 3 utilizes a General Electric boiling water reactor Mark I primary containment consisting of a drywell, a wetwell, six vents connecting the drywell and wetwell, primary containment access penetrations, and other process piping and electrical penetrations. The integrity of the penetrations and isolation valves are verified through Type B and Type C local leak rate tests (LLRTs) as required by 10 CFR Part 50, Appendix J, and the overall leak-tight integrity of the primary containment is verified through the ILRT. These tests are performed to verify the essentially leak-tight characteristics of the containment at the design-basis accident (DBA) pressure. The last ILRT for PBAPS Unit 3 was performed in December 1991. The ILRT, the LLRTs, and inservice inspection (ISI) of the containment collectively ensure the leak-tight and structural integrity of the containment. The licensee is using the 1992 Edition and the 1992 Addenda of Subsections IWE of Section XI of the ASME Boiler and Pressure Vessel Code (the ASME Code) for conducting the ISI of the PBAPS Unit 3 containment, with certain approved relief from some ASME Code requirements. The current containment ISI 10-year interval began in 1998, and will end in 2008. The licensee's responses indicate that the accessible areas of the containment pressure boundary will be periodically monitored for signs of degradation.

There is a 10 CFR Part 50, Appendix J, Option B requirement that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. PBAPS Unit 3 TS 5.5.12 requires that 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 Part 50, Appendix J, Option B, as modified by approved exemptions. PBAPS Unit 3 TS 5.5.12 further requires that this program shall be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995, as modified by exceptions set forth in the TSs. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) Report, NEI 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated July 26, 1995.

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at PBAPS Unit 3 have been successful, so their current interval requirement is 10 years.

The licensee is requesting an addition to TS 5.5.12, “Primary Containment Leakage Rate Testing Program,” which would indicate that they are allowed to take an exception to the guidelines of RG 1.163 regarding the Type A test interval. Specifically, the proposed TS says that the first Type A test performed after the December 1991 Type A test shall be performed no later than December 2006. This would make the interval 15 years between tests.

The licensee states that the requested extension would provide substantial cost savings and reduce personnel radiation exposure by approximately 2 rem.

3.0 EVALUATION

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. The licensee provided the assessment to the staff in a May 30, 2001, letter (ADAMS Accession No. ML011620078). The licensee provided a sensitivity analysis in a July 24, 2001, letter (ADAMS Accession No. ML012120271). In performing the risk assessment, they considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) Technical Report, TR-104285, “Risk Impact Assessment of Revised Containment Leak Rate Testing,” and RG 1.174,“ An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.”

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, provided 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 U.S. Nuclear Regulatory Commission's (NRC's) rulemaking basis, NEI undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test frequency from 3 in 10 years to 1 in 10 years, will increase the average time that a leak detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10-percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage, in percent of person-rem/year, for the pressurized water reactor representative plant was estimated to increase from .032 percent to .035 percent. This confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 per 10 years to 1 per 10 years leads to an "imperceptible" increase in risk.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The licensee quantified the leakage from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking in 1995, the NRC staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking. RG 1.174 defines very small 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 which the licensee estimated. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability as an element demonstrating that the defense-in-depth philosophy is met.

The licensee examined plant-specific accident sequences from their Individual Plant Examination. The assessment considered the following sequences:

- Core damage sequences in which the containment remains intact initially and in the long term.

- Core damage sequences in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress. These sequences involve Type A tests and potential failures not detectable by local leak-rate tests (e.g., a hole in the containment liner). The impact on risk from changes in Type A test frequency are evaluated by investigating these sequences.

- Core damage sequences in which containment integrity is impaired due to 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.

- Accident sequences in which the containment is bypassed or involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents. Therefore, these sequences were not evaluated further after their initial quantification.

The licensee took the following steps to perform the risk assessment:

- Quantified the base-lined risk in terms of frequency per reactor year for each of the eight accident classes evaluated in EPRI TR-104285.

- Developed plant-specific person-rem dose per year of reactor operation for the surrounding population for each of the eight accident classes.

- Evaluated the risk impact of extending the Type A test interval from 10 to 15 years and the cumulative impact of extending the interval from the original 3 in 10 years to 1 in 15 years.

- Determined the change in risk in terms of LERF in accordance with RG 1.174.

Determining the change in risk in terms of LERF involves the potential that a core damage event that normally would result in only a small radioactive release from containment could in fact result in a large release due to failure to detect a pre-existing leak during the extension period. The licensee designated these sequences as Class 3B sequences and estimated a frequency of 9.52×10^{-8}/year, based on the original 3-year test interval. The licensee then used the EPRI methodology to estimate the impact of the Type A test interval on the leakage probability. Extending the Type A test interval from the original test interval to 10 years increases the average time that a leak detectable only by a Type A test goes undetected from 18 to 60 months. For a 10-year test interval there is a 10-percent increase in the overall probability of leakage (3 * 60/18) versus 15 percent for a 15-year test interval. The licensee estimated a Class 3B sequence frequency of 1.09×10^{-7}/year for the 15-year interval. Therefore, the increase in LERF can be estimated by the change in the frequency of Class 3B sequences. Extending the Type A test interval from the current 10-year interval to 15 years results in a 4.8×10^{-9}/year increase. If the risk increase is measured from the original 3 in 10-year interval, the increase in LERF is 1.4×10^{-8}/year.

The NRC staff made the following conclusions from the licensee's risk assessment associated with extending the Type A test frequency:

1. The risk assessment predicted a slight increase in risk when compared to that estimated from current requirements. Given the change from a 10-year test interval to a 15-year test interval, the increase in the total integrated plant risk (person-rem/year within 50 miles) is estimated to be 0.04 percent. The increase in the total integrated plant risk, given the change from a 3 in 10-year test interval to a 1 in 15-year test interval, was found to be 0.12 percent. This is reasonable when compared to the range of risk increase, 0.02 to 0.14 percent, estimated in NUREG-1493 when going from a 3 in 10-year test interval to a 1 in 10-year interval. NUREG-1493 concluded that a reduction in the frequency of tests from 3 per 10 years to 1 per 10 years leads to an "imperceptible" increase in risk. The increase in the total integrated plant risk for the

proposed change is considered small and, therefore, the NRC staff finds that this supports the proposed change.

2. RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in CDF less than 10^{-6} per reactor year and increases in LERF less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A test interval from 1 in 10 years to 1 in 15 years is estimated to be 4.8×10^{-9}/year. The increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be 1.4×10^{-8}/year. Increasing the Type A interval to 15 years is considered to be a very small change in LERF.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in the conditional containment failure probability was estimated to be an increase of 0.001 for the proposed change and 0.0031 for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the very small change in the conditional containment failure probability for the proposed change.

   The NRC staff recognizes the limitations of a conditional containment failure probability approach. For plants, such as Peach Bottom, with core damage frequency estimates well below 10^{-4}, the ability of the containment to withstand events of even lower probability becomes less clear. Therefore, it is important to consider other risk metrics in conjunction with the conditional containment failure probability, such as total LERF. The licensee's submittal has sufficiently demonstrated that the total LERF is less than 10^{-5}.

Based on these conclusions, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance criteria while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

The licensee also provided information related to the ISI Program for containment integrity. The NRC staff, in a letter dated July 6, 2001, asked the licensee about the testability of the bellows on the vent lines from the drywell to the wetwell. The licensee responded that these two-ply bellows are testable by Type B testing. A review of the licensee's records since 1977 revealed no leakage failures of these bellows. Under Option B of Appendix J leak-rate testing, these bellows will be pressure-tested every 6 years.

The NRC staff also asked the licensee to describe the examination of the primary containment pressure boundary seals and gaskets. The licensee stated that under Option B of Appendix J, their leak-tight integrity will be pressure-tested every 6 years during Type B testing. Moreover, the licensee states that the examination of seals and gaskets will continue to be performed each time a specific joint is disassembled.

The NRC staff asked the licensee about the likelihood of degradation from the uninspectable side of the drywell steel shell and steel liner of the primary containment. The licensee stated that PBAPS Unit 3 primary containment has a number of design features and inspection requirements that guard against any corrosion taking place in the sand-cushion area and in the air gap region of the drywell. Also, the licensee performs a visual examination on the drywell air gap drain lines once each inspection period when the refueling cavity is flooded to look for signs of water leakage. The licensee notes that with no water intrusion in these areas, potential degradation on the outside uninspectable surface of the drywell is prevented. Additionally, the licensee points out that penetrations, hatches, bolting surfaces, and structural members (including the steel shell) are considered in the PRA structural analysis to assess the failure pressure and failure location of containment. If evidence of component degradation occurs, then that information is fed back into the PRA to reassess the containment failure pressure, temperature and failure location. The conservative increase in the failure probability calculated in the ILRT submittal to account for the extension of the ILRT interval (i.e. a factor of 100 increase in containment leakage probability) reflects the fact that such failure modes may go undetected for an additional period of time.

The licensee stated in their letter dated July 24, 2001, that during power operation, a positive pressure is continuously maintained in the drywell, and that gross leakage would be detected by the increase in nitrogen makeup. The licensee stated that: (1) the primary containment is typically maintained at an average positive pressure of 0.5 pounds per square inch gage (psig) to ensure that no external sources of oxygen are introduced into the nitrogen inerted primary containment; (2) at a positive pressure lower than 0.25 psig, and higher than 0.75 psig in the drywell, an annunciation is made in the main control room; (3) the primary containment gross leakage rate detection test is performed every 72-hour period to identify excessive primary containment trends; (4) if a primary containment leak is identified, then the TS action for an inoperable primary containment would be entered. The NRC staff finds that this continuous monitoring of slight positive pressure will ensure that areas of containment degradation will be detected before they could result in large leakage rates.

Based on the licensee's procedures discussed above to preclude excessive degradation of the primary containment components, the NRC staff concludes that granting the requested ILRT extension will not adversely affect the leak-tight integrity of the primary containment. It should be noted that Subarticle IWE-5000 of the ASME Code, Section XI, requires leak-rate testing following repair, modification, or replacement of containment components. An ILRT might be required to confirm that these activities are adequate and that further degradation does not exist in other areas of the containment. The licensee is required to report serious degradation of the containment pressure boundary pursuant to 10 CFR 50.72 or 10 CFR 50.73. Based on the licensee’s submittals, the NRC staff finds that the licensee has adequate procedures to examine and monitor potential age-related and environmental degradations of the pressure retaining components of the PBAPS Unit 3 primary containment.

Based on the foregoing evaluation, the NRC staff finds that the interval until the next Type A test at PBAPS Unit 3 may be extended to 15 years, and that the proposed changes to TS 5.5.12 are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (66 FR 36341). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: H. Ashar, J. Pulsipher, M. Snodderly

Date: October 4, 2001