April 8, 2003

Mr. William R. Kanda
Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P.O. Box 97, A200
10 Center Road
Perry, OH 44081

SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT (TAC NO. MB4695)

Dear Mr. Kanda:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 126 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit 1. This amendment revises the Technical Specifications (TSs) in response to your application dated March 14, 2002 (PY-CEI/NRR-2607L) as supplemented by letter dated January 20, 2003 (PY-CEI/NRR-2679L).

This amendment revises TS 5.5.12, “Primary Containment Leakage Rate Testing Program,” to allow a one-time exception to Nuclear Energy Institute 94-01, “Industry Guidance for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J,” that extends the test interval of the containment integrated leak rate test from 10 to 15 years.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Stephen P. Sands, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures: 1. Amendment No. 126 to License No. NPF-58
2. Safety Evaluation

cc w/encls: See next page

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NAME	SSands	THarris	DTerao*	SWeerakkody**	SCole	AMendiola
DATE	3/26/03	3/26/03	02/05/03	02/21/03	3/24/03	4/13/03

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*See DTerao Memorandum to AMendiola dated 2/5/03
**See SWeerakkody Memorandum to AMendiola dated 2/21/03
***No legal objection with changes

Perry Nuclear Power Plant, Unit 1

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FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 126
License No. NPF-58

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

   A. The application for amendment by the FirstEnergy Nuclear Operating Company (the licensee) dated March 14, 2002, as supplemented by letter dated January 20, 2003, 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. NPF-58 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 126 are hereby incorporated into this license. The FirstEnergy Nuclear Operating Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 8, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 126

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

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 a marginal line indicating the area of change.

Remove	Insert
5.0-15a	5.0-15a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 126 TO FACILITY OPERATING LICENSE NO. NPF-58

FIRSTENERGY NUCLEAR OPERATING COMPANY

PERRY NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By letter dated March 14, 2002, as supplemented by letter dated January 20, 2003, FirstEnergy Nuclear Operating Company (FENOC), the licensee for the Perry Nuclear Power Plant (PNPP), proposed amending Technical Specification (TS) (Ref. 7.1). The proposed TS change would revise TS 5.5.12, "Primary Containment Leakage Rate Testing Program," thereby allowing a one-time, 5-year extension of the Appendix J, Type A, Containment Integrated Leak Rate Test (ILRT) frequency from 10 years to 15 years. The licensee requested approval of the proposed TS amendment to support its ninth refueling outage.

This safety evaluation discusses (1) the acceptability of the licensee's procedures and activities that would ensure containment structural integrity by managing the aging degradation of the containment pressure boundary components, and (2) the risk assessment associated with extending the Type A test interval to 15 years. These activities are directly related to the proposed one-time TS amendment to extend the test interval for performing the containment ILRT from the currently required 10 years to 15 years. The last ILRT was performed in July 1994 and the next ILRT is proposed to be no later than June 29, 2009.

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

2.0 REGULATORY EVALUATION

When implementing Title 10 of the Code of Federal Regulations, Part 50 (10 CFR 50), Appendix J, Option B (Ref. 7.2), licensees follow the guidelines of Regulatory Guide (RG) 1.163 (Ref. 7.3). RG 1.163, Section C, "Regulatory Position" states, "licensees intending to comply with the Option B in the amendment to Appendix J should establish test intervals based upon the criteria in Section 11.0 of Nuclear Energy Institute (NEI) 94-01 (Ref. 7.4), rather than using test intervals specified in American National Standards Institute/American Nuclear Society 56.8-1994 (Ref. 7.5)." The industry guidelines in Section 11 of NEI 94-01 state that Type A testing shall be performed at a frequency of at least once every 10 years. The licensee's proposed TS amendment would change the 10-year ILRT interval to a 15-year interval on a one-time basis. The licensee's request for this one-time extension of the ILRT interval is based on the staff guidelines in RG 1.174 (Ref. 7.6). There are no changes to any Code or regulatory requirement.

3.0 TECHNICAL EVALUATION - CONTAINMENT STRUCTURAL INTEGRITY

The staff has reviewed the licensee's regulatory and technical analyses in support of its license amendment, which are described in the licensee's submittal dated March 14, 2002, and a request for additional (RAI) response dated January 20, 2003 (Ref. 7.7). The following evaluation addresses the acceptability of issuing an amendment pursuant to 10 CFR 50.92 (Ref. 7.8).

3.1 Applicable Technical Specification Requirements

The licensee proposes to revise TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program," by inserting the following paragraph: The provisions of NEI 94-01, Section 9.2.3 are revised to include the following exception: The first Type A test performed after the Type A test completed on July 1, 1994, shall be completed no later than June 29, 2009.

3.2 Inservice Inspection (ISI) for Primary Containment Integrity

PNPP utilizes a General Electric boiling water reactor with a Mark III type primary containment. The containment vessel consists of a continuous and essentially leak-tight steel membrane which includes the cylindrical portion, the torispherical portion, and the floor liner plate on the top of the basemat. The cylindrical portion is backed by hoop stiffeners and structural concrete in the lower 23 ft 6 in. above the top of the basemat. The containment design incorporates a cylindrical drywell and a cylindrical weir wall concentric with the containment cylindrical wall; forming a suppression pool. The containment vessel is penetrated by access penetrations, process piping and electrical penetrations. The integrity of the penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) as required by 10 CFR Part 50, Appendix J. The overall leak-tight integrity of the primary containment is verified through ILRTs. These tests are performed to verify the essentially leak-tight characteristics of the containment at the design-basis accident pressure. The last ILRT at PNPP was performed in July 1994. With the extension of the ILRT interval, the licensee commits to perform the next ILRT no later than June 29, 2009.

The licensee provided information related to the ISI of the containment and discussed potential areas of degradation in the containment that might not be apparent in the risk assessment. In addition, in its letter dated January 20, 2003 (Ref. 7.7), the licensee provided responses to the staff's RAI to explicitly address five issues related to containment inspection and degradation. The staff's evaluation of the licensee's responses to the ISI-related issues is discussed in the following paragraphs.

The licensee is using the 1992 Edition and the 1992 Addenda of the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsections IWE (Ref. 7.9) for ISI of the steel containment. In response to the staff's question on implementation of the ISI requirements, the licensee provided the following information: Visual Examination procedure, NQ1-1042, "Visual Examination." For the IWE General Visual and VT-3 examinations, recordable indications are structural deformation or degradation, missing or detached items, cracked or broken welds, erosion, excessive corrosion, wear, pitting, arc strikes, gouges, surface discontinuities, dents, and degraded coatings. The acceptance criteria are no structural deformation or degradation such that the component's function is impaired, no missing or detached items, no cracks, and no corrosion or erosion of structural metal which exceeds 10 percent of the nominal wall thickness. All other recordable indications are evaluated by the Registered Professional Engineer (RPE), or knowledgeable individual under the RPE's direction, to determine whether

the recorded indications affect either the containment structural integrity or leak tightness.

The only identifiable in-service degradations have been minor flaking and peeling of coatings on the interior surfaces of the containment and drywell, which are addressed by PNPP's nuclear coatings program, and numerous areas of general surface corrosion on the exterior surfaces of the containment (which received a primer coat, but never a top coat). The corrosion areas were checked for material loss and no significant material loss was found. Based on the summary of the examination procedures provided in Reference 7.7, the staff finds the licensee's program for examining the accessible portions of the containment steel surfaces adequate for detecting flaws and degradation.

In response to the staff's RAI on examination and testing of seals, gaskets, and pressure retaining bolts, the licensee provided the following information:

> Type B testing at PNPP affects a total of 63 components encompassing electrical penetrations, the containment airlocks, containment equipment hatch "O" rings, the inclined fuel transfer tube bellows, containment vacuum breaker "O" rings, and containment expansion bellow assemblies.

3.2.1 Electrical Penetrations

There are 36 electrical penetrations. Of this population, approximately 20 percent are tested during each refueling outage. The PNPP administrative leakage limit for the electrical penetrations is 25 standard cubic centimeters per minute (sccm). If an electrical penetration fails to meet the acceptance criteria, all electrical penetrations will be tested during the refueling outage to establish if a common mode failure mechanism exists. Presently, 9 of the 36 electrical penetrations are scheduled for Type B testing in RFO9. Historical leakage data for the 36 electrical penetrations has shown that none have ever exhibited a leakage greater than the lowest sensitivity of the test equipment used.

3.2.2 Containment Airlocks

The containment airlocks are comprised of several components that are periodically leak tested albeit at different frequencies. A large and a small seal are tested in parallel on both the outer and inner door of both airlocks resulting in eight (8) components being tested. The door seals for both the lower and upper containment airlocks are tested once per 30 days. The acceptance criteria for the door seals is that leakage be less than 1180 sccm. Considering the large number of cycles the doors are subjected to, the door seal leakage tests have historically exhibited minimal leakage (< 100 sccm). The containment airlock barrel tests are performed every 30 months. The acceptance criteria for the airlock barrel overall leakage is also less than 1180 sccm. The highest recorded leakage for either barrel was 945 sccm for the lower airlock barrel in 1994.

3.2.3 Containment Equipment Hatch

The containment equipment hatch is removed during each refueling outage to support outage work activities. Each “O” ring of the containment equipment hatch’s double "O" ring seal is Type B tested during each refueling outage. The hatch seal has an assigned leakage limit of 250 sccm. As-found leakage has never exceeded 20 sccm. The equipment hatch bolting is scheduled for VT-1 examination in RFO10 in accordance with Table IWE-2500-1, Category E-G, Item E8.10.

3.2.4 Inclined Fuel Transfer Tube Bellows

A two-ply bellows assembly surrounds the inclined fuel transfer tube and thus provides a flexible seal between the Inclined Fuel Transfer System (IFTS) containment penetration flange and the IFTS piping. The bellows and associated components were never subject to extended interval testing at PNPP given that the IFTS containment penetration flange is removed each refueling outage to support outage activities. Type B testing of each of the inclined fuel transfer tube bellows remains on the original 2-year frequency. The leakage limit for the penetration is 100 sccm. Historically, the recorded as-found leakage rates for the IFTS bellows assembly have been no greater than 40 sccm.

3.2.5 Containment Vacuum Breaker "O" Rings

The inboard isolation valves (check valves) in the four containment vacuum breaker penetrations are tested each refueling outage. This requires the valve body-to-pipe flange "O" rings to be tested. An administrative leakage limit of 100 sccm is assigned to each of the containment vacuum breaker "O" rings. The highest leakage ever recorded was 20 sccm in 1994.

3.2.6 Containment Expansion Bellows Assemblies

Eleven containment expansion bellows assemblies are tested in parallel and are on a 5-year extended test interval. The group of assemblies has a leakage limit of 100 sccm. This group has displayed minimal leakage rates since 1986. These assemblies were successfully tested during RFO8 (March 2001). The bellows assemblies are currently scheduled for testing in 2005 during RFO10. The staff finds that the schedule for testing containment penetration seals, gaskets, pressure retaining bolting, vacuum breaker isolation valves, and expansion bellows is consistent with the regulations and licensee’s operating experience. Therefore, the staff finds the test schedules acceptable for the extended ILRT interval.

In response to the staff’s RAI related to the potential degradation of bellows and the ability to test the bellows by Type B testing, the licensee provided the following response: PNPP reviewed and assessed the IFTS containment bellows regarding Type B testing in March 1994 and communicated that assessment to the Nuclear Regulatory Commission (NRC) by letter dated March 3, 1994, (PY-CEI/NRR-1676L). The IFTS containment bellows assembly is a two-ply design similar in construction and subject to the same Type B testing limitations as described in NRC Information Notice (IN) 92-20, "Inadequate Local Leak Rate Testing." IN 92-20 indicated that the LLRT test methodology utilized could not be relied upon above a threshold value that was determined to be 6 standard cubic feet per hour (scfh). This value corresponds to 2832 standard cubic centimeters per minute (sccm). The IFTS bellows testing consists of a post-maintenance test between the ply leak rate test and an as-found/as-left external containment boundary local leak rate test (Type B test) each refueling outage. Historically,

PNPP leakage rates determined for the IFTS containment bellows assembly have been no greater than 40.0 sccm. As stated in the licensee's response to the RAI, the IFTS bellows is Type B tested during each refueling outage. The Type B leakage rate for the penetration during RFO8 was determined to be less than 20.0 sccm. If the leakage was found to be greater than 100 sccm during testing, additional confirmatory testing and/or corrective actions would be necessary to reduce the leakage to less than the 100 sccm limit. Additionally, the latest ASME Code Section XI (VT-3) visual examination of the IFTS bellows assembly external and internal surfaces was performed in RFO8 (February 2001). No indications on the IFTS bellows assembly were noted.

Based on the above discussion, the staff finds that the implementation of the licensee's containment ISI program, including the areas subjected to subsequent inspections and testing, provides reasonable assurance that the identified degradation occurring in the accessible areas of the Perry containment will be adequately monitored during the ILRT interval extension.

In response to the question on incorporating the potential degradation in uninspectable areas of the containment in the risk assessment, in Reference 7.7, the licensee considered the following steps in its risk assessment:

- Determining the containment basemat and the containment cylinder and dome areas to identify accessible and inaccessible areas;
- Determining historical steel shell flaw likelihood due to concealed corrosion;
- Consideration of the impact of aging;
- Determining corrosion leakage dependency on containment pressure; and
- Establishing the likelihood that visual inspections will be effective at detecting a flaw.

The acceptance of the licensee's risk-assessment is discussed in Section 3.3 in this safety evaluation. Based on the information provided by the licensee, the staff finds that (1) the structural degradation of the accessible areas of the PNPP containment will be adequately monitored through the periodic ISI conducted as required by Subsection IWE of Section XI of the ASME Code, and (2) the integrity of the penetrations and containment isolation valves will be periodically verified through Type B and Type C tests as required by 10 CFR Part 50, Appendix J. In addition, the system pressure tests for containment pressure boundary (i.e., Appendix J tests, as applicable) are required to be performed following repair and replacement activities in accordance with Subarticle IWE-5000 of Section XI of the ASME Code. Significant degradation of the primary containment pressure boundary is required to be reported under 10 CFR 50.72 or 10 CFR 50.73.

3.3 TECHNICAL EVALUATION - RISK EVALUATION

The licensee performed a risk impact assessment of extending the Type A test interval to 15 years. The assessment was provided to the staff in the March 14, 2002, application for license amendment. Additional analysis and information were provided by the licensee in a letter dated January 20, 2003. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) 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 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 for the pressurized-water reactor and boiling-water reactor representative plants confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 in 10 years to 1 in 20 years leads to an “imperceptible” increase in risk on the order of 0.2 percent and a fraction of one person-rem per year.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The licensee quantified the risk 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 staff has issued RG 1.174 on the use of probabilistic risk assessment 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}/year and increases in large early release frequency (LERF) less than 10^{-7}/year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original 3 in 10 year interval. 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 for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided an analysis which estimated all of these risk metrics and its methodology is consistent with previously approved submittals. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

1. A slight increase in risk is predicted when compared to that estimated from current requirements. Given the change from a 3 in 10 year test interval to a 1 in 15 year test interval, the increase in the total integrated plant risk is estimated to be 0.04 person-rem/year. This increase is comparable to that estimated in NUREG-1493, in which it was concluded that a reduction in the frequency of tests from 3 in 10 years to 1 in

20 years leads to an “imperceptible” increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. 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 6.4×10^{-8}/year. However, there is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the IWE visual examination of the containment surfaces (as identified in American Society of Mechanical Engineers [ASME] Boiler and Pressure Vessel Code, Section XI, Subsection IWE). The most recent visual examination of the Perry containment was performed in November 2000. The next scheduled IWE containment inspection is scheduled for spring 2005, during refueling outage 10. Visual inspections are expected to be effective in detecting large flaws in the visible regions of the containment, and would reduce the impact of the extended test interval on LERF. The licensee performed additional risk analysis to consider the potential impact of corrosion in inaccessible areas of the containment shell on the proposed change. The risk analysis considered the likelihood of an age-adjusted flaw that would lead to a breach of the containment. The risk analysis also considered the likelihood that the flaw was not visually detected but could be detected by a Type A ILRT. When possible corrosion of the containment surfaces is considered, 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 6.9×10^{-8}/year. The staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

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 licensee estimates the change in the conditional containment failure probability to be an increase of about 1 percentage point for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. The staff finds that the defense-in-depth philosophy is maintained based on the change in the conditional containment failure probability for the proposed amendment.

Based on the above technical evaluation, the 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 Perry containment. Thus, granting a one-time 5-year extension to the current 10-year test interval for the containment integrated leak-rate testing, as proposed by the licensee in TS Section 5.5.12 is acceptable. Based on the risk evaluation, the staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

This 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, or which change an inspection or a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding (68 FR 5676). Accordingly, this 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 this amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

7.1 Letter from FENOC to NRC, "License Amendment Request Pursuant to 10 CFR 50.90: One-time Five Year Deferral of the Containment Integrated Leak Rate Test (ILRT)," March 14, 2002.

7.2 Title10, Part 50, Appendix J of Code of Federal Regulations (CFR), "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

7.3 USNRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

7.4 Nuclear Energy Institute Document, NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."

7.5 ANSI/ANS-56.8, "Containment System Leakage Testing Requirements," 1994.

7.6 USNRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

7.7 Letter from FENOC to NRC, "Supplement to License Amendment Request Pursuant to 10 CFR 50.90: One-time Five Year Deferral of the Containment Integrated Leak Rate Test (ILRT)," January 20, 2003.

7.8 10 CFR 50.92, "Issuance of Amendment."

7.9 ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants," 1992 Edition including 1992 Addenda.

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Date: April 8, 2003