

December 8, 1995

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Mr. Donald C. Shelton  
Vice President Nuclear - Perry  
Centerior Service Company  
P. O. Box 97, A200  
Perry, OH 44081

SUBJECT: AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-58 - PERRY  
NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M90684)

Dear Mr. Shelton:

The Commission has issued the enclosed Amendment No. 76 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1. This amendment revises the Technical Specifications in response to your application dated October 21, 1994 (PY-CEI/NRR-1650L).

This amendment revises Specifications 3/4.6.1.2, "Primary Containment Leakage," and its associated Bases to reflect the partial exemptions to the requirements of 10 CFR Part 50, Appendix J, Option A, Sections III.A.5(b)(2), III.B.3, III.C.3, III.A.1(d), III.D.1(a), and III.D.3 that were granted by the NRC on December 4, 1995.

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

Sincerely,

Original signed by:  
Jon B. Hopkins, Sr. Project Manager  
Project Directorate III-3  
Division of Reactor Projects III/IV  
Office of Nuclear Reactor Regulation

Docket No. 50-440

- Enclosures: 1. Amendment No. 76 to License No. NPF-58  
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555-0001

December 8, 1995

Mr. Donald C. Shelton  
Acting Vice President Nuclear - Perry  
Centerior Service Company  
P. O. Box 97, A200  
Perry, OH 44081

SUBJECT: AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-58 - PERRY  
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A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

A handwritten signature in cursive script that reads "Jon B. Hopkins, Sr.".

Jon B. Hopkins, Sr. Project Manager  
Project Directorate III-3  
Division of Reactor Projects III/IV  
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures: 1. Amendment No. 76 to  
License No. NPF-58  
2. Safety Evaluation

cc w/encls: See next page

Mr. Donald C. Shelton  
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Perry Nuclear Power Plant  
Unit Nos. 1 and 2

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UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555-0001

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76  
License No. NPF-58

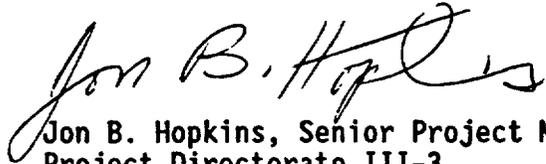
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, and Toledo Edison Company (the licensees) dated October 21, 1994, 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. 76 are hereby incorporated into this license. The Cleveland Electric Illuminating Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented not later than 90 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Jon B. Hopkins, Senior Project Manager  
Project Directorate III-3  
Division of Reactor Projects III/IV  
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: December 8, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 76

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove

3/4 6-3

3/4 6-4

B 3/4 6-2

B 3/4 6-2a

B 3/4 6-2b

Insert

3/4 6-3

3/4 6-4

B 3/4 6-2

B 3/4 6-2a

B 3/4 6-2b

## CONTAINMENT SYSTEMS

### PRIMARY CONTAINMENT LEAKAGE

#### LIMITING CONDITION FOR OPERATION

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3.6.1.2 Primary containment leakage rates shall be limited to:

- a. An overall integrated leakage rate, except for the main steam line isolation valves#, of less than or equal to  $0.75 L_a$ , where  $L_a$  is 0.20 percent by weight of the primary containment air per 24 hours at  $P_a$ .
- b. A combined leakage rate of less than or equal to  $0.60 L_a$  for all penetrations and all valves, except for main steam line isolation valves# and valves which are hydrostatically leak tested, subject to Type B and C tests when pressurized to  $P_a$ .
- c. Less than or equal to 25 scf per hour for any one main steam line through the isolation valves when tested at  $P_a$ .
- d. A combined leakage rate of less than or equal to  $0.0504 L_a$  for all penetrations that are secondary containment bypass leakage paths when pressurized to the required test pressure.
- e. A combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at greater than or equal to  $1.10 P_a$ .

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, AND 3, with the reactor coolant system temperature greater than 200 °F.

#### ACTION:

- With:
- a. The measured overall integrated primary containment leakage rate, except for the main steam line isolation valves#, exceeding  $0.75 L_a$ , or
  - b. The measured combined leakage rate for all penetrations and all valves except for main steam line isolation valves# and valves which are hydrostatically leak tested, subject to Type B and C tests exceeding  $0.60 L_a$ , or
  - c. The measured leakage rate exceeding 25 scf per hour for any one main steam line through the isolation valves, or
  - d. The combined leakage rate for all penetrations that are secondary containment bypass leakage paths exceeding  $0.0504 L_a$ , or

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# Exemption to Appendix J of 10 CFR 50.

## CONTAINMENT SYSTEMS

### PRIMARY CONTAINMENT LEAKAGE

#### LIMITING CONDITION FOR OPERATION (Continued)

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- e. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves:

Restore the leakage rate to less than or equal to the above limit(s) within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

#### SURVEILLANCE REQUIREMENTS

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4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50, except that the provisions of Bechtel Topical Report BN-TOP-1 may be used for Type A tests having a duration less than 24 hours:

- a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at  $40 \pm 10$  month intervals during shutdown at  $P_a$  during each 10-year service period.
- b. If any periodic Type A test fails to meet  $0.75 L_a$ , the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet  $0.75 L_a$ , a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet  $0.75 L_a$ , at which time the above test schedule may be resumed.
- c. The accuracy of each Type A test shall be verified by a supplemental test which:

## CONTAINMENT SYSTEMS

### BASES

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#### 3/4.6.1 CONTAINMENT (Continued)

##### 3/4.6.1.2 CONTAINMENT LEAKAGE (Continued)

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J to 10 CFR 50 with the exception of the exemptions listed below. Additionally, Bechtel Topical Report BN-TOP-1 may be utilized for ILRTs with a duration of less than 24 hours in accordance with Section 7.6 of ANSI N45.4-1972 (Reference 1).

- a) Section III.D.2(b)(ii) - The air lock seal leakage test of Section III.D.2(b)(iii) of Appendix J may be substituted (following normal air lock door opening) for the full-pressure test provided that no maintenance has been performed that would affect the air locks sealing capability (Reference 2).
- b) Sections III.A.1(d), III.A.5(b)(2), and III.B.3 and III.C.3 - The main steam lines between the inboard and outboard MSIVs (including the volume up to the outboard MSIV before seat drain line valves) are not required to be vented and drained for Type A testing, and the main steam line isolation valve leak rates are exempted from inclusion in the overall integrated primary containment leak rate and the combined local leak rate (Reference 3).
- c) Section III.D.1(a) - The third Type A test for each 10-year service period is not required to be conducted when the plant is shutdown for the 10-year plant inservice inspection (Reference 3).
- d) Section III.D.3 - Type C local leak rate testing may be performed at other convenient intervals in addition to shutdown during refueling, but at intervals no greater than 2 years (Reference 3).

#### References

- (1) Letter from NRC (B. J. Youngblood) to CEI (M. R. Edelman), "Performance of the Preoperational Containment Integrated Leak Rate Test - Perry Nuclear Power Plant, Unit 1," dated June 10, 1985.
- (2) PNPP Safety Evaluation Report Supplement 7, Section 6.2.6 "Containment Leakage Testing," November 1985.
- (3) Letter from NRC (J. B. Hopkins) to CEI (D. C. Shelton), "Issuance of Exemption from the Requirements of 10 CFR Part 50, Appendix J - Perry Nuclear Power Plant, Unit 1," dated December 4, 1995.

##### 3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

## CONTAINMENT SYSTEMS

### BASES

#### 3/4.6.1 CONTAINMENT (Continued)

##### 3/4.6.1.3 CONTAINMENT AIR LOCKS (Continued)

An allowance has been provided within Action a.1 for access into or through the containment air locks when an interlock mechanism in one or both air locks is inoperable. Action a.1 requires that at least one of the two OPERABLE doors for each affected air lock be maintained closed, and if the interlock mechanism has not been restored to OPERABLE status within 24 hours, one door must be locked closed. The provisions of footnote \* may be utilized for entries and exits. The administrative controls of footnote \* allow the unlocking and use of the air lock provided that an individual is stationed at the air lock, dedicated to assuring that at least one OPERABLE air lock door remains closed at all times. This allowance is provided to address those situations when the use of an air lock with only an inoperable interlock mechanism may be preferred over the use of the other air lock, such as when the other air lock has an inoperable door.

An allowance has also been provided in Action a.2 for access into or through the containment air locks when one air lock door in one or both air locks is inoperable. The first sentence of footnote \*\* provides that entry and exit through the OPERABLE door on one or both air locks is permissible under administrative controls for the performance of repairs of the affected air lock components. The second sentence of footnote \*\* provides for entry into and exit from the containment for activities other than just the repairs of affected air lock components under administrative controls, but only permits these entries when both air locks have an inoperable door, and limits such use to a 7 day period. The administrative controls for the second sentence shall define limits on entry and exit, in order to minimize openings of the OPERABLE door.

The administrative controls for both sentences of footnote \*\* include provisions that after each entry and exit, the OPERABLE door must be promptly closed. The allowances of footnote \*\* are acceptable because of the low probability of an event that could pressurize the containment during the short time that the OPERABLE door will be open for entry into and exit from the containment.

The air supply to the containment air lock and seal system is the service and instrument air system. The system consists of two 100% capacity air compressors per unit and can be cross-connected. This system is redundant and extremely reliable and provides system pressure indication in the control room.

##### 3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steam line isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage

## CONTAINMENT SYSTEMS

### BASES

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#### 3/4.6.1 CONTAINMENT (Continued)

##### 3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM (Continued)

control system will reduce the untreated leakage from the MSIV's when isolation of the primary system and containment is required. An LCO 3.0.4 exception is provided to permit changes in Operational Conditions when the Inboard MSIV-LCS subsystem becomes inoperable due to condensate buildup between the MSIVs when the plant is operated below 50% rated thermal power.

##### 3/4.6.1.5 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 15 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

##### 3/4.6.1.6 CONTAINMENT INTERNAL PRESSURE

The limitations on primary containment to secondary containment differential pressure ensure that the primary containment peak pressure of 7.80 psig does not exceed the design pressure of 15.0 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of +0.8 psid. The limit of -0.1 to +1.0 psid for initial positive primary containment to secondary containment pressure will limit the primary containment pressure to 7.80 psig which is less than the design pressure and is consistent with the safety analysis.

##### 3/4.6.1.7 CONTAINMENT AVERAGE AIR TEMPERATURE

The limitation on containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 185°F during LOCA conditions and is consistent with the safety analysis.

##### 3/4.6.1.8 DRYWELL AND CONTAINMENT PURGE SYSTEM

The use of the drywell and containment purge lines is restricted to the 42-inch outboard and 18-inch purge supply and exhaust isolation valves. These valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations. The term sealed closed as used in this context means that the valve is secured in its closed position by deactivating the valve motor operator, and does not pertain to injecting seal water between the isolation valves by a seal water system.



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
RELATED TO AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-58  
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.  
PERRY NUCLEAR POWER PLANT, UNIT NO. 1  
DOCKET NO. 50-440

1.0 INTRODUCTION

Containment leak rate testing is necessary to demonstrate that the measured leak rate is within the acceptance criteria cited in the licensing design basis. Periodic testing of the overall containment structure along with separate leak testing of the penetrations provides assurance that post-accident radiological consequences will be within the limits of 10 CFR Part 100. The Commission's requirements regarding leak rate testing are found in Appendix J to 10 CFR Part 50.

By letter dated October 21, 1994 (PY-CEI/NRR-1650L), the licensee requested an amendment to Facility Operating License NPF-58 for the Perry Nuclear Power Plant (PNPP). The amendment would make changes to Technical Specification 3/4.6.1.2, "Primary Containment Leakage," and its associated Bases to reflect proposed exemptions to the requirements of 10 CFR Part 50, Appendix J, Option A, Sections III.A.5(b)(2), III.B.3, III.C.3, III.A.1(d), III.D.1(a), and III.D.3. The proposed exemptions were submitted by separate letter, also dated October 21, 1994 (PY-CEI/NRR-1651L).

The proposed exemptions to Appendix J, Option A, were subsequently approved by the staff in a letter dated December 4, 1995. The approved exemptions will:

- a. Exclude main steam line isolation valve leakage from inclusion in both the containment integrated leak rate (Type A) test and the combined local leak rate (Type B and C) tests, and clarify that the main steam lines are not required to be vented and drained for Type A testing;
- b. Decouple performance of the third Type A test from the shutdown for the 10-year plant inservice inspection; and
- c. Allow Type C testing to be performed at times other than during shutdown for refueling.

## 2.0 EVALUATION

### Sections III.A.5(b)(2), III.B.3, III.C.3, and III.A.1(d)

Section III.A.5(b)(2) states that the measured leakage from the containment integrated leak rate (Type A) test ( $L_{int}$ ) shall be less than 75% of the maximum allowable leakage rate ( $0.75 L_a$ ). The licensee proposed to exempt main steam line isolation valve leakage from Type A test results and consider leakage from the main steam lines separately. Sections III.B.3 and III.C.3 require that the combined leakage of valves and penetrations subject to Type B and C local leak rate testing be less than 0.6 times the maximum allowable leakage rate ( $0.6 L_a$ ). The licensee proposed to exempt main steam line isolation valve leakage from the combined leakage from Type B and C local leak rate testing and consider leakage from the main steam lines separately. Section III.A.1(d) requires that all fluid systems that would be open to containment following post-accident conditions, be vented and drained prior to conducting Type A tests. The licensee proposed that the piping between the inboard and outboard main steam line isolation valves be flooded with water when Type A tests are conducted.

In support of these exemptions, the licensee proposed modifying Technical Specification Limiting Condition for Operation (LCO) 3.6.1.2.a by inserting words to clearly state that main steam line isolation valve leakage is separate from the overall integrated leakage rate. Similar wording would also be inserted in Action 3.6.1.2.a. The licensee has also proposed to add a superscript "#" to LCO 3.6.1.2.b and to Action 3.6.1.2.b to refer to a footnote which clarifies that an Appendix J exemption is involved.

During the original staff review of the PNPP, the licensee proposed separate treatment of measured leakage past the main steam isolation valves. The licensee's radiological dose analysis assumed separate contributions from both containment leakage and main steam isolation valve leakage. This approach was reviewed and approved in the staff's Safety Evaluation Report (NUREG-0887). The PNPP Final Safety Analysis Report and technical specifications limit the maximum containment leakage to 0.20 percent per day. In addition, technical specifications limit the maximum allowable leakage of each main steam line to 25 standard cubic feet per hour.

Consistent with separate handling of main steam line isolation valve leakage, the licensee proposed that the piping between the inboard and outboard isolation valves be filled with water when Type A tests are performed. Filling these sections of pipe ensures that air does not pass through these lines thus inadvertently contributing to the Type A test results.

The methodology proposed by the licensee accounts for all containment leakage paths. Treating main steam line isolation valve leakage separately from both the integrated Type A test and the combined Type B and C tests still verifies that the total leakage is within the design limits and, therefore, meets the underlying purpose of the rule.

On this basis, the staff found that separating main steam line isolation valve leakage from Type A, B and C leak rate tests would not present an undue risk

to the public health and safety. Accordingly, the NRC approved the issuance of the subject exemption on December 4, 1995.

Based on the above, and because the licensee's proposed change to the technical specifications is consistent with the exemption approved by the NRC, the staff finds these changes acceptable.

#### Section III.D.1(a)

Section III.D.1(a) requires, in part, that "...a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant inservice inspections." The licensee proposes to perform the three Type A tests at approximately equal intervals within each 10-year period, with the third test of each set conducted as close as practical to the end of the 10-year period. However, there would be no required connection between the Appendix J 10-year interval and the inservice inspection 10-year interval.

In support of this proposed exemption, the licensee proposed modifying Technical Specification 4.6.1.2.a by deleting the sentence requiring that the two tests be performed during the same outage.

The 10-year plant inservice inspection (ISI) is the series of inspections performed every 10 years in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. The licensee performs the ISI volumetric, surface, and visual examinations of components and system pressure tests in accordance with 10 CFR 50.55a(g)(4) throughout the 10-year inspection interval. The major portion of this effort is presently being performed every refueling outage. As a result, there is no extended outage in which the 10-year ISI examinations are performed.

There is no benefit to be gained by the coupling requirement cited above in that elements of the ISI program are conducted throughout each 10-year cycle rather than during a refueling outage at the end of the 10-year cycle. Consequently, the subject coupling requirement offers no benefit either to safety or to the economical operation of the facility.

Moreover, each of these two surveillance tests (i.e., the Type A tests and the 10-year ISI program) is independent of the other and provides assurances of different plant characteristics. The Type A test assures the required leak-tightness to demonstrate compliance with the guidelines of 10 CFR Part 100. The 10-year ISI program provides assurance of the integrity of the structures, systems and components as well as verifying operational readiness of pumps and valves in compliance with 10 CFR 50.55a. There is no safety-related concern necessitating their coupling in the same refueling outage. Accordingly, the staff finds that the subject exemption request meets the underlying purpose of the rule.

On this basis, the staff found that the uncoupling of the Type A tests from the 10-year ISI program would not present an undue risk to the public health and safety. Accordingly, the NRC staff approved the issuance of the subject exemption on December 4, 1995.

Based on the above, and because the licensee's proposed change to the technical specifications is consistent with the exemption approved by the NRC, the staff finds these changes acceptable.

### Test Methodologies

The current PNPP Technical Specification 4.6.1.2 states that the containment integrated leak rate test shall be performed using the criteria specified in Appendix J to 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972 and BN-TOP-1. In addition, Specification 4.6.1.2 states that the test results shall also be reported based on the Mass Point Methodology described in ANSI/ANS N56.8-1981. The Mass Point Methodology of ANSI/ANS N56.8-1981 is intended for use when the Type A tests are greater than 24 hours in duration whereas BN-TOP-1 is to be used for durations less than 24 hours.

Subsequent to the original licensing of the PNPP, Appendix J to 10 CFR Part 50 has been revised to reflect NRC approved methodologies and updated versions of ANSI standards. However, by referencing specific standards in the technical specifications, the licensee does not have the flexibility to incorporate updated versions without processing a license amendment. Therefore, the licensee has proposed modifying Specification 4.6.1.2 to only reference BN-TOP-1 and the criteria specified in Appendix J while eliminating references to ANSI N45.4-1972 and ANSI/ANS N56.8-1981. Since Appendix J now references both ANSI N45.4-1972 and ANSI/ANS N56.8-1987, the licensee will be able to adopt the latest ANSI standards along with any future updates to Appendix J. The proposed Specification 4.6.1.2 clarifies that the provisions of BN-TOP-1 may be used for Type A tests having a duration less than 24 hours.

The proposed modification still requires the licensee to be in conformance with the criteria of Appendix J. By deleting references to specific ANSI standards, the licensee gains the additional flexibility to adopt updated standards without processing a license amendment. Since the proposed change does not alter the licensee's compliance with the requirements of Appendix J, the staff finds the proposed change to be acceptable.

### 3.0 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.

### 4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change to 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 (60 FR 42611). 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.

#### **5.0 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.

**Principal Contributor: Douglas Pickett**

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