Mr. William J. Cahill, Jr. October 4, 1996
Chief Nuclear Officer
Power Authority of the State of New York
123 Main Street
White Plains, NY 10601

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M95099)

Dear Mr. Cahill:

The Commission has issued the enclosed Amendment No. 234 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated March 27, 1996, as supplemented April 24, 1996, and August 15, 1996.

The proposed amendment changes would permit implementation of 10 CFR Part 50, Appendix J, Option B, with an exception to the guidelines of Regulatory Guide 1.163 for Type C testing of primary containment isolation valves in the reverse (non-accident) direction.

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

Sincerely,
/S/
Karen R. Cotton, Acting Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures: 1. Amendment No. 234 to DPR-59
2. Safety Evaluation

cc w/encls: See next page

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DATED: October 4, 1996

AMENDMENT NO. 234 TO FACILITY OPERATING LICENSE NO. DPR-59-FITZPATRICK

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

October 4, 1996

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Chief Nuclear Officer
Power Authority of the State of New York
123 Main Street
White Plains, NY 10601

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Project Directorate I-1
Division of Reactor Projects - I/II
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Docket No. 50-333

Enclosures: 1. Amendment No. 234 to DPR-59
2. Safety Evaluation

cc w/encls: See next page

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 234
License No. DPR-59

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated March 27, 1996 as supplemented April 24, 1996 and August 15, 1996, 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-59 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 234 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

S. Singh Bajwa, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 4, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 234

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
iv	iv
vi	vi
30e	30e
30f	30f
166	166
167	167
194	194
198	---
258e	258e
285	285

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4.0 BASES

A. This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL CONDITIONS (modes) for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL CONDITIONS (modes) are provided in the individual Surveillance Requirements.

B. Specification 4.0.B establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance (e.g., transient conditions or other ongoing surveillance or maintenance activities). It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with a 24 month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of this specification is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. The limit on extension of the normal surveillance interval ensures that the reliability confirmed by surveillance activities is not significantly reduced below that obtained from the specified surveillance interval. The exceptions to Specification 4.0.B are those surveillances for which the 25% extension of the interval specified does not apply. These exceptions are stated in the individual Technical Specifications. The requirements of regulations take precedence over the Technical Specifications. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended under the provisions of 4.0.B, and the surveillance requirement will be identified as an exception. An example of an exception when the test interval is not specified in the regulations is the Note in Specification 6.20, "Primary Containment Leakage Rate Testing Program," which states "The provisions of Specification 4.0.B do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program." This exception is provided because the program already includes provisions for extension of intervals.

C. This specification establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.B, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.C. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.B, was a violation of the OPERABILITY requirements of a Limiting Condition for Operation that is subject to enforcement action. Further, the failure to

4.0 BASES - Continued

C. Continued

perform a surveillance within the provisions of Specification 4.0.B is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10 CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant Technical Specifications.

If the allowable outage time limits of the ACTION requirements are less than 24 hours or a shutdown is required to comply with ACTION requirements, a 24-hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with ACTION requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of OPERATIONAL CONDITION (mode) changes imposed by ACTION requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.C is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time the surveillance is terminated.

C. Continued

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

D. This specification establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL CONDITION or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into an OPERATIONAL CONDITION or other specified condition associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION requirements, the provisions of this specification do not apply because this would delay placing the facility in a lower CONDITION of operation.

3.7 (cont'd)

(2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F, and fuel is in the reactor vessel, except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt.

4.7 (cont'd)

2. a. Perform required visual examination and leakage rate testing of the Primary Containment in accordance with the Primary Containment Leakage Rate Testing Program.

b. Demonstrate leakage rate through each MSIV is $\leq$ 11.5 scfh when tested at $\geq$ 25 psig. The testing frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

c. Once per 24 months, demonstrate the leakage rate of 10AOV-68A,B for the Low Pressure Coolant Injection system and 14AOV-13A,B for the Core Spray system to be less than 11 scfm per valve when pneumatically tested at $\geq$ 45 psig at ambient temperature, or less than 10 gpm per valve if hydrostatically tested at $\geq$ 1000 psig at ambient temperature.

Pages 167 through 175 Have Been Deleted

4.7 BASES (cont'd)

assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide additional margin between expected offsite doses and 10CFR100 guidelines.

The leakage rate testing program was originally based on NRC guidelines for development of leak rate testing and surveillance schedules for reactor containment vessels. Containment structural integrity is currently verified with visual inspections and containment leak tightness is verified by the leakage rate surveillance testing described in the JAFNPP Primary Containment Leakage Rate Testing Program.

The following are the exemptions to 10 CFR 50 Appendix J, Option A, that have been approved by the NRC, and remain applicable to Option B of 10 CFR 50, Appendix J:

1. The Type C exceptions listed on Table 4.7-2, "Ex eption to Type C Test", as of the date of issuance of Amendment 194 (July 29, 1993).

2. Valves which are sealed with fluid from a seal system, such as the liquid in the suppression chamber are not required to be Type C tested. This exemption was approved by the NRC in the original Technical Specifications (SR 4.7.A.2.c(3)).

3. The MSIVs are tested at a pressure less than P_a and $\geq$ 25 psig, with a leakage rate acceptance criteria of $\leq$ 11.5 scfh per valve. This exemption was approved by the NRC in the original Technical Specifications (Table 4.7-2).

The Program as implemented meets the requirements of Option B of 10 CFR 50 Appendix J (16) and Regulatory Guide 1.163 (13), with the exception stated in Specification 6.20. This exception applies to valves currently installed in this configuration, and does not apply to new installations. This exception is consistent with TS Table 4.7-2, previously contained in the TS, which allows reverse direction testing of valves as an exception to the requirements of the draft Appendix J, on the basis that pressurization direction was not a requirement at the time of plant design.

B. Standby Gas Treatment System and
C. Secondary Containment

Initiating reactor building isolation and operation of the Standby Gas Treatment System to maintain at least a 1/4 in. of water vacuum within the secondary containment provides an adequate test of the operation of the reactor

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6.19 POSTACCIDENT SAMPLING PROGRAM

A program shall be established, implemented, and maintained which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

A) Training of personnel,

B) Procedures for sampling and analysis,

C) Provisions for maintenance of sampling and analysis

6.20 PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate 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 exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995, as modified by the exception that Type C testing of valves not isolable from the containment free air space may be accomplished by pressurization in the reverse direction provided that testing in this manner provides equivalent or more conservative results than testing in the accident direction. If potential atmospheric leakage paths (e.g., valve stem packing) are not subjected to test pressure, the portions of the valve not exposed to test pressure shall be subjected to leakage rate measurement during regularly scheduled Type A testing. A list of these valves, the leakage rate measurement method, and the acceptance criteria, shall be contained in the Program.

A. The peak Primary Containment internal pressure for the design basis loss of coolant accident (P_a), is 45 psig.

B. The maximum allowable Primary Containment leakage rate (L_a), at P_a, shall be 1.5% of primary containment air weight per day.

C. The leakage rate acceptance criteria are:

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

2. Airlock testing acceptance criteria are:

   a. Overall airlock leakage rate is $\leq$ 0.05 L_a when tested at $\geq$ P_a,

   b. For each door seal, leakage rate is $\leq$ 120 scfd when tested at $\geq$ P_a.

3. MSIV leakage rate acceptance criteria is $\leq$ 11.5 scfh for each MSIV when tested at $\geq$ 25 psig.

D. The provisions of Specification 4.0.B do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

E. The provisions of Specification 4.0.C are applicable to the Primary Containment Leakage Rate Testing Program.

7.0 REFERENCES

(1) E. Janssen, "Multi-Rod Burnout at Low Pressure," ASME Paper 62-HT-26, August 1962.

(2) K.M. Backer, "Burnout Conditions for Flow of Boiling Water in Vertical Rod Clusters," AE-74 (Stockholm, Sweden), May 1962.

(3) FSAR Section 11.2.2.

(4) FSAR Section 4.4.3.

(5) I.M. Jacobs, "Reliability of Engineered Safety Features as a Function of Testing Frequency," Nuclear Safety, Vol. 9, No. 4, July-August 1968, pp 310-312.

(6) Deleted

(7) I.M. Jacobs and P.W. Mariott, APED Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards - April 1969.

(8) Bodega Bay Preliminary Hazards Report, Appendix 1, Docket 50-205, December 28, 1962.

(9) C.H. Robbins, "Tests of a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

(10) "Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1966, Progress Report for Period Ending December 31, 1966, ORNL-4071."

(11) Section 5.2 of the FSAR.

(12) TID 20583, "Leakage Characteristics of Steel Containment Vessel and the Analysis of Leakage Rate Determinations."

(13) Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995.

(14) Section 14.6 of the FSAR.

(15) ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III. Maximum allowable internal pressure is 62 psig.

(16) 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Option B - Performance Based Requirements", Effective Date October 26, 1995

(17) Deleted

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 234 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B "Performance-Based Requirements" to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall leakage rate performance and the performance of individual components.

By application dated March 27, 1996, as supplemented by letters dated April 24, 1996, and August 15, 1996, the New York Power Authority (the licensee or the authority) requested changes to the Technical Specifications (TSs) for James A. FitzPatrick (JAF) Nuclear Power Plant. The proposed changes would permit implementation of 10 CFR Part 50, Appendix J, Option B by referencing Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," dated September 1995. The licensee has established a "Containment Leakage Rate Testing Program" and proposed adding this program to the TS. The April 24, 1996, supplemental letter was not outside the scope of the original notice of the application in that it addressed the reverse flow type C testing of 17 valves as provided for in JAF TSs and did not affect performance based leakage testing. The August 15, 1996, supplemental letter provided clarifying information that was not outside the scope of this original notice of the application.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

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On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR Part 50 was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, was developed as a method acceptable to the NRC staff for implementing Option B. This RG states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the RG or other implementation document used by a licensee to develop a performance-based leakage rate testing program must be included, by general reference, in the plant TS. The licensee has referenced RG 1.163 in the JAF TS.

Regulatory Guide 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were attached to a letter from C. Grimes (NRC) to D. Modeen (NEI) dated November 2, 1995. These TS are to serve as a model for licensees to develop plant specific TS in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

3.1 Option B

The licensee's March 27, 1996 letter, supplemented by the April 24, 1996, and August 15, 1996, letters to the NRC proposes to establish a "Containment Leakage Rate Testing Program" and proposes to add this program to the TS. The program references RG 1.163, which specifies a method acceptable to the NRC for complying with Option B, with an exception for Type C testing of primary containment isolation valves in the reverse (non-accident) direction. This requires a change to existing TS 4.7-2 and the addition of the "Containment Leakage Rate Testing Program" to Section 6.20. Corresponding bases were also modified.

3.2 Reverse F Testing

Periodic Type C testing in the reverse (non-accident) direction for primary containment isolation valves does not expose potential atmospheric leakage paths (e.g., valve stem packing) to test pressure. Therefore, it can not be quantitatively shown that Type C test results are not affected in a non-conservative manner by testing in the reverse direction. Section 8.0 of NEI 94-01, REV. 0, requires that potential leakage paths to atmosphere be quantitatively determined. At FitzPatrick, reverse direction testing of 17 of these valves is required due the inability to isolate the valves from the containment and the lack of test connections. These valves are reverse direction tested in accordance with the current FitzPatrick TS Table 4.7-2, "Exception to Type C Tests."

Type C testing in the reverse direction for these valves provides equivalent or more conservative results than testing in the accident direction, with respect to seat leakage. With respect to the globe valves, the test pressurization is under the seat, which tends to unseat the valve. With respect to the butterfly valves, measured leakage is independent of the direction of test pressure from both a force exerted and seating surface standpoint.

Modifications have been considered by the licensee that would allow testing in the accident direction or allow potential leakage to atmosphere to be quantitatively determined. The addition of block valves and test connections to allow accident direction testing would increase design complexity, provide additional potential leakage pathways, and increase loading on piping penetrating primary containment. Valve stem packing modifications to allow

potential leakage to be quantitatively determined would increase design complexity, and provide additional potential leakage pathways.

There is a low safety significance associated with this proposal because:

1. Testing of these valves during the 1995 Integrated Leakage Rate Test (ILRT) verified that the packing glands were insignificant contributors to the overall integrated leakage rate. The 1995 as-left ILRT leakage rate was 0.0629% weight/day, which was well below the current TS acceptance criteria of 0.5% weight/day.

2. Adding the results of the 1995 As-Left Type A, B, and C tests together (approximately 2188 SCFD) results in a leakage total well below $0.6L_d$ (3216 SCFD). This conservatively shows that significant margin exists to exceeding TS or Appendix J limits.

3. Review of past ILRT results indicates that the 17 valves have not been the cause of an ILRT failure. Based on a review of the maintenance history for each valve, recurring packing or body to bonnet leaks are not expected.

4. The valve stem packing and body to bonnet gaskets are resilient materials designed to conform to sealing surfaces. The valves are installed in systems which are not normally subjected to design flows, temperatures, or pressures. During normal operation, the valve stem packing and body-to-bonnet gaskets are exposed to the primary containment atmosphere, which has a low oxygen content. Based on this, the degradation of the valve stem packing or body-to-bonnet gaskets due to continuous exposure to a harsh environment is not a significant concern.

5. From a risk perspective evaluation, the elimination of modifications that would allow testing in the accident direction or allow potential leakage to atmosphere to be quantitatively determined, can be justified using the technical bases provided for NUREG-1493. Past studies show that overall reactor accident risks are not sensitive to variations in containment leakage rate, within one or two orders of magnitude of La, the allowable containment leakage rate. This is because reactor accident risks are dominated by accident scenarios in which the containment fails or is bypassed. Such scenarios, even though they are of very low probability, dominate the predicted accident risks due to their high consequences. FitzPatrick Individual Plant Examination (IPE) results are consistent with these past technical studies.

Certain NRC sponsored studies indicate that overall plant risk is not sensitive to changes in containment leak rates. The incremental risk from leakage in the range of 1% to 10% per day is small. FitzPatrick and Peach Bottom, which were subjects of the studies, are both BWR 4 plants with MARK I containments. Similar results are expected for FitzPatrick.

The analysis described above provides justification that potential leakage paths to atmosphere for these 17 valves are unlikely to be significant and that the associated risk to public health and safety is insignificant. The Authority proposes that a soap bubble test be performed on the pressurized stem/bonnet boundaries of the 17 valves during regularly scheduled Type A testing. To provide a direct indication of the leak-tightness of the packing and body-to-bonnet gaskets, the Authority will use the acceptance criterion of zero bubbles for this test. Type C testing will be performed, as a post-work test, following work activities that affect the potential atmospheric leakage paths on any of the 17 valves. A soap bubble test will then be performed on the subject valve(s) at regularly scheduled Type A test intervals. These requirements will be contained in the Leak Rate Test Program.

3.3 Conclusions

The TS changes proposed by the licensee are in compliance with the requirements of Option B and consistent with the guidance of Regulatory Guide 1.163, dated September 1995, with the exception of reverse direction testing, as discussed above. Despite the different format of the licensee's current TS, all of the important elements of the guidance provided in the NRC letter to NEI dated November 2, 1995, are included in the proposed TS. Therefore, they are acceptable to the NRC staff. Based on the above 3.2, the licensee has provided justification that reverse direction testing, along with additional measures to ensure leak tightness of valve packing and gaskets, provide adequate assurance that the overall objectives of 10 CFR Part 50, Appendix J, will be met, and therefore, the licensee's performance based leak rate testing program is acceptable to the NRC staff.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. 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 and changes surveillance requirements. 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 (61 FR 20855). 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.

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