May 8, 1989

Docket Nos. 50-338
and 50-339

DISTRIBUTION
See attached sheet

Mr. W. R. Cartwright
Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Dear Mr. Cartwright:

SUBJECT: NORTH ANNA UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CONTAINMENT ISOLATION VALVES (TAC NOS. 56348 AND 56349)

The Commission has issued the enclosed Amendment Nos. 116 and 99 to Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). The amendments revise the Technical Specifications (TS) in response to your letter dated October 19, 1984.

The amendments add additional surveillance requirements for the butterfly-type containment isolation valves in the containment purge lines and the containment vacuum ejector lines. These valve leakage tests are to be performed in addition to the tests of the valves required by Appendix J to 10 CFR Part 50.

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

Sincerely,

Original signed by

Leon B. Engle, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 116 to NPF-4
2. Amendment No. 99 to NPF-7
3. Safety Evaluation

cc w/enclosures:
See next page

[AMEND 56348/56349]

*SEE PREVIOUS CONCURRENCE

*LA:PDII-2	PM:PDII-2	D:PDII-2	*OGC
DMiller	LEngle:bd	HBerkow	EReis
04/10/89	05/5/89	05/5/89	04/18/89

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DATED: May 8, 1989

AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. NPF-4-NORTH ANNA UNIT 1
AMENDMENT NO. 99 TO FACILITY OPERATING LICENSE NO. NPF-7-NORTH ANNA UNIT 2

Docket File
NRC & Local PDRs
PDII-2 Reading
S. Varga, 14/E/4
G. Lainas, 14/H/3
H. Berkow
D. Miller
L. Engle
OGC-WF
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
B. Grimes, 9/A/2
T. Meek (8), P1-137
Wanda Jones, P-130A
E. Butcher, 11/F/23
J. S. Guo, 8/D/1
ACRS (10)
GPA/PA
ARM/LFMB
B. Wilson, R-II

cc: Plant Service list

DF01
1/1

Mr. W. R. Cartwright
Virginia Electric & Power Company

North Anna Power Station
Units 1 and 2

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Mineral, Virginia 23117

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. NPF-4

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated October 19, 1984, 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-7 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 116 , 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 issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 8, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 116

TO FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Page

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B3/4 6-1

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3.1., and

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

c. After each closing of the equipment hatch, by leak rate testing the equipment hatch seals with gas at P_a, greater than or equal to 40.6 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

d. Each time containment integrity is established after vacuum has been broken by pressure testing the butterfly isolation valves in the containment purge lines and the containment vacuum ejector line.

*Except valves, blind flanges and deactivated automatic valves which are located inside the containment and are locked sealed or otherwise sealed in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:**

a. An overall integrated leakage rate of:

1. $\leq L_a$, 0.1 percent by weight of the containment air per 24 hours at P_a, $\geq$ 44.1 psig, or

b. A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves subject to Type B and C tests, when pressurized to P_a, $\geq$ 44.1 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or (b) with the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding 0.60 L_a, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The 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 50 using the methods and provisions of either ANSI N45.4-1972 for leakage rate point data analysis or ANSI/ANS-56.8-1987 for mass point data analysis with a minimum test duration of 24 hours.**

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 $\pm$ 10 month intervals during shutdown at $P_a \geq$ 44.1 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.*

*The third test of the first 10-year service period shall be conducted during the 1989 Refueling/10-Year ISI Outage.

**For Specification 3/4.6.1.2 only, P_a shall be 40.6 psig until completion of the Cycle 7 to 8 refueling outage. Following this outage, P_a shall be 44.1 psig.

3/4.6.1 CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

Leakage integrity tests in the containment purge lines and the containment vacuum ejector system lines is to identify excessive degradation of the resilient seats of these valves. These tests will be performed in addition to the Type C tests required by 10 CFR Part 50, Appendix J and will not relieve the responsibility to conform with Appendix J.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage tests ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50. Due to the increased accuracy of the mass-point method for containment integrated leakage testing, the mass-point method referenced in ANSI/ANS 56.8-1987 can be used in lieu of the methods described in ANSI N45.4-1972.

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 CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 and 3/4.6.1.5 INTERNAL PRESSURE AND TEMPERATURE

The limitations on containment internal pressure and average air temperature ensure that

1) The containment pressure is prevented from reaching the containment lower design pressure of 5.5 psia for an inadvertent containment spray actuation,

2) That the peak clad fuel temperature will remain less than 2200°F for a LOCA and

3) That for either a LOCA or MSLB;

   a) The peak containment pressure will be limited to the upper containment design pressure of 45 psig,

   b) The containment internal pressure can be returned sub-atmospheric within 60 minutes, and

   c) Safety related equipment within the containment will not experience temperatures greater than those to which they have previously been qualified.

   d) It is a design criteria that the containment internal pressure remain subatmospheric after 60 minutes.

The limits shown in Figure 3.6-1 and Specification 3.6.1.5 are consistent with the assumptions of the accident analyses which included consideration of instrument loop uncertainties.

3/4.6.1.6 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 facility. Structural integrity is required to ensure that the containment will withstand the design pressure of 45 psig. The visual examination of the concrete and liner and the Type A leakage tests are sufficient to demonstrate this capability.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT QUENCH AND RECIRCULATION SPARY SYSTEMS

The OPERABLILITY of the containment spray systems ensures that containment depressurization and subsequent return to subatmospheric pressure will occur in the event of a LOCA. The pressure reduction and resultant termination of containment leakage are consistent with the assumptions used in the accident analyses.

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 99
License No. NPF-7

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated October 19, 1984, 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.D.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 99 , 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 issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 8, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 99

TO FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Page

3/4 6-1
B3/4 6-1

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3.1., and

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

c. After each closing of the equipment hatch, by leak rate testing the equipment hatch seals with gas at Pa, greater than or equal to 40.6 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

d. Each time containment integrity is established after vacuum has been broken by pressure testing the butterfly isolation valves in the containment purge lines and the containment vacuum ejector line.

*Except valves, blind flanges and deactivated automatic valves which are located inside the containment and are locked sealed or otherwise sealed in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:**

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.1 percent by weight of the containment air per 24 hours at P_a, greater than or equal to 44.1 psig, or

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a, greater than or equal to 44.1 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or (b) with the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a and the combined leakage rate for all penetrations subject to Type B and C tests to less than or equal to 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The 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 50 using the methods and provisions of either ANSI N45.4-1972 for leakage rate point data analysis or ANSI/ANS-56.8-1987 for mass point data analysis with a minimum test duration of 24 hours.**

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at P_a greater than or equal to 44.1 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.*

*The second test of the first 10-year service period shall be conducted during the 1989 Refueling Outage.

**For Specification 3/4.6.1.2 only, P_a shall be 40.6 psig until completion of the Cycle 6 to 7 refueling outage. Following this outage, P_a shall be 44.1 psig.

3/4.6.1 CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

Leakage integrity tests in the containment purge lines and the containment vacuum ejector system lines is to identify excessive degradation of the resilient seats of these valves. These tests will be performed in addition to the Type C tests required by 10 CFR Part 50, Appendix J and will not relieve the responsibility to conform with Appendix J.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50. Due to the increased accuracy of the mass-point method for containment integrated leakage testing, the mass-point method referenced in ANSI/ANS 56.8-1987 can be used in lieu of the methods described in ANSI N45.4-1972.

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 CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 and 3/4.6.1.5 INTERNAL PRESSURE AND TEMPERATURE

The limitations on containment internal pressure and average air temperature ensure that

1) The containment pressure is prevented from reaching the containment lower design pressure of 5.5 psia for an inadvertent containment spray actuation,

2) That the peak clad fuel temperature will remain less than 2200°F for a LOCA and

3) That for either a LOCA or MSLB:

    a) The peak containment pressure will be limited to the upper containment design pressure of 45 psig,

    b) The containment internal pressure can be returned subatmospheric within 60 minutes, and

    c) Safety related equipment within the containment will not experience temperatures greater than those to which they have previously been qualified.

    d) It is a design criteria that the containment internal pressure remain subatmospheric after 60 minutes.

The limits shown in Figure 3.6-1 and Specification 3.6.1.5 are consistent with the assumptions of the accident analyses which included consideration of instrument loop uncertainties.

3/4.6.1.6 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 facility. Structural integrity is required to ensure that the containment will withstand the design pressure of 45 psig. The visual examination of the concrete and liner and the Type A leakage tests are sufficient to demonstrate this capability.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 116 AND 99 TO

FACILITY OPERATING LICENSE NOS. NPF-4 AND NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

NORTH ANNA POWER STATION, UNITS NO. 1 AND NO. 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By letter dated October 19, 1984, Virginia Electric and Power Company (the licensee) requested changes to the North Anna, Units 1 and 2 Technical Specifications (TS) to add additional surveillance requirements for the butterfly-type containment isolation valves in the containment purge lines and the containment vacuum ejector lines. The licensee proposed to test these valves each time containment integrity is established after containment vacuum has been broken. The licensee stated that the proposed TS change is part of the implementation of provisions of NUREG-0737, Item II.E.4.2, "Containment Isolation Dependability" and the guidance developed as part of Multi-Plant Action (MPA) B-24, "Containment Purging During Normal Plant Operation." The following evaluation addresses the licensee's TS changes.

2.0 EVALUATION

The licensee proposed to add TS 4.6.1.1.d (surveillance requirements) to the TS for both units to read as follows:

> "Primary containment integrity shall be demonstrated each time containment integrity is established after vacuum has been broken by pressure testing the butterfly isolation valves in the containment purge lines and the containment vacuum ejector line."

The purpose of the leakage integrity tests for these butterfly-type valves is to identify excessive degradation of the resilient seats of these valves. Excessive leakage past the resilient seat of a butterfly valve is typically caused by severe environmental conditions and/or wear due to frequent use. Consequently, the leak testing frequency for these valves should be keyed to the occurrence of severe environmental conditions and the use of the valves, rather than to the current containment leak rate testing requirements of 10 CFR Part 50, Appendix J. These tests are not the Type C tests of valves set out in 10 CFR Part 50, Appendix J. These valve leakage tests are to be performed in addition to the Type C tests of the valves required by Appendix J.

Each unit of the North Anna plant has seven butterfly containment isolation valves to be leak tested in accordance with the proposed surveillance requirements of TS 4.6.1.1.d. They include one 8-inch valve inside containment for the containment vacuum ejector line, two 36-inch valves inside containment and two 36-inch valves outside containment for the containment purge supply and exhaust lines, one 18-inch valve outside containment for containment purge alternate supply units, and one 8-inch valve outside containment for the containment purge bypass line. Branch Technical Position (BTP) CSB 6-4, Item B.4 specifies that provisions should be made for testing the availability of the isolation function and the leakage rate of these valves during reactor operation, but it does not address test frequency. The criteria of MPA B-24 states that leakage integrity tests be performed for the containment isolation valves with resilient seals in (1) the active purge system (i.e., those which may be operated during reactor operation) at least once every 3 months, and (2) the passive purge system (i.e., those which must be administratively controlled closed during reactor operation) at least once every 6 months. Rather than testing the passive purge system valves at 6-month intervals, the licensee proposed to test these valves when containment integrity is re-established after containment vacuum is broken.

The staff notes that the licensee's proposed test frequency deviates from the MPA B-24 recommended test interval. However, the staff finds the proposed test frequency to be consistent with the intent of MPA B-24 for the following reasons:

(1) The valve leak tests require personnel entry into the containment because some of the test connections are located inside containment. In addition, a blind flange must be installed on the lines inside containment to permit these valves to be tested individually in the appropriate direction. Therefore, personnel radiological exposure considerations make the leak testing impractical during power operation.

(2) It is not reasonable to require a plant shutdown every 6 months to satisfy the MPA B-24 test interval guideline.

(3) Since the leakage integrity tests are conducted following each cycling of the isolation valves, significant additional valve degradation is not expected while the valves are not used.

(4) The proposed valve leak tests are performed in addition to the required Appendix J Type C test to verify possible valve degradation.

The licensee also proposed changes to the corresponding TS bases to justify the proposed TS changes. The staff concurs with the licensee's proposed TS changes and bases since they meet the intent of the staff guidelines for maintaining integrity of the containment isolation valves.

Finally, the licensee also proposed revising the format of the Unit 1 TS to correspond to the Unit 2 TS. The staff agrees with this proposed change and finds it acceptable.

3.0 CONCLUSION

The staff has reviewed the licensee's proposed changes to TS 4.6.1.1 regarding the frequency of leak testing for the butterfly type isolation valves in the containment purge and vacuum systems. Based on its review, the staff concludes that the changes meet the intent of the guidelines of MPA B-24 and NUREG-0737, Item II.E.4.2, and therefore are acceptable.

ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to surveillance requirements. The staff has determined that the amendments involve 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 published a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet 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 amendments.

CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 8, 1989

Principal Contributor:

J. S. Guo