

December 14, 1988

Docket Nos. 50-338
and 50-339

DISTRIBUTION
See attached page

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
UPPER LIMIT TEMPERATURE (TAC NOS. 67535 AND 67536)

The Commission has issued the enclosed Amendment Nos. 110 and 96 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 March 2, 1988, as supplemented August 5, 1988. The amendments are effective as of the date of issuance and shall be implemented within 14 days.

The amendments revise the NA-1&2 TS containment air temperature upper limit from 105°F to 120°F and the volume of water available from the refueling water storage tank for the quench spray system is redefined and reduced to permit the use of wide range level instrumentation for TS surveillance.

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. 110 to NPF-4
2. Amendment No. 96 to NPF-7
3. Safety Evaluation

cc w/enclosures:
See next page

[AMEND 67535/36]

*See previous concurrence

LA:PDII-2*
DMiller
11/22/88

PM:PDII-2
LEROTE
12/14/88

D:PDII-2
Hjerrow
12/14/88

PSB:NRR*
JCraig
11/22/88

OGC-WF*
SHLewis
11/22/88

PSB:NRR
CYang Li
11/23/88

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DATED: December 14, 1988

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

Docket File

NRC & Local PDRs

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ARM/LFMB

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OFO
||
||

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Virginia Electric & Power Company

North Anna Power Station
Units 1 and 2

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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. 110
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 March 2, 1988, as supplemented August 5, 1988, 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.

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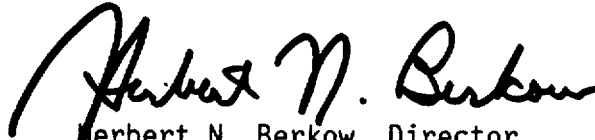
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.D.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 110, 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 14 days.

FOR THE NUCLEAR REGULATORY COMMISSION



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: December 14, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 110

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.

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EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

- a. A contained borated water volume of between 466,200 and 487,000 gallons.
- b. Between 2300 and 2400 ppm of boron, and
- c. A solution temperature between 40°F and 50°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

- a. At least once per 7 days by:
 - 1. Verifying the contained borated water volume in the tank, and
 - 2. Verifying the boron concentration of the water.
- b. At least once per 24 hours by verifying the RWST temperature.

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:
 1. 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
 2. All equipment hatches are closed and sealed
- b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

* 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. $\leq L_a$, 0.1 percent by weight of the containment air per 24 hours at P_a , ≥ 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^a pressurized to P_a , ≥ 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 + 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 5 to 7 refueling outage. Following this outage, P_a shall be 44.1 psig.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- b. If any periodic Type A test fails to meet $.75 L_a$, 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 $.75 L_a$, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet $.75 L_a$ 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:
1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within $0.25 L_a$.
 2. Has a duration sufficient to establish accurately the change in leakage between the Type A test and the supplemental test.
 3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at $P_a \geq 44.1$ psig.
- d. Type B and C tests shall be conducted with gas at $P_a, \geq 44.1$ psig, at intervals no greater than 24 months except for tests involving:
1. Air locks,
 2. Penetrations using continuous leakage monitoring systems
- e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.
- f. Type B test for penetrations employing a continuous leakage monitoring system shall be conducted at $P_a \geq 44.1$ psig at intervals no greater than once per 3 years.
- g. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.
- h. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

- a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and
- b. An overall air lock leakage rate of less than or equal to $0.05 L_a$ at P_a greater than or equal to 44.1 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

- a. With one containment air lock door inoperable:
 1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.+
 2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.
 3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
 4. The provisions of Specification 3.0.4 are not applicable.
- b. With a containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

- a. *Within 72 hours following closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying that the seal leakage is less than $0.01 L_a$ as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a pressure of greater than or equal to 44.1 psig.
- b. At least once per 6 months by conducting an overall air lock leakage test at greater than or equal to P_a , 44.1 psig, and by verifying that the overall air lock leakage rate is within its limit#, and
- c. At least once per 18 months during shutdown by verifying that only one door in each air lock can be opened at a time.

+Entry to repair the inner air lock door, if inoperable, is allowed.

*Exempt to Appendix "J" of 10 CFR Part 50.

#The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal air partial pressure shall be maintained > 9.0 psia and within the acceptable operation region on Figure 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal air partial pressure < 9.0 psia or above the applicable limit shown on Figure 3.6-1, restore the internal air partial pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal air partial pressure shall be determined to be within the limits at least once per 12 hours.

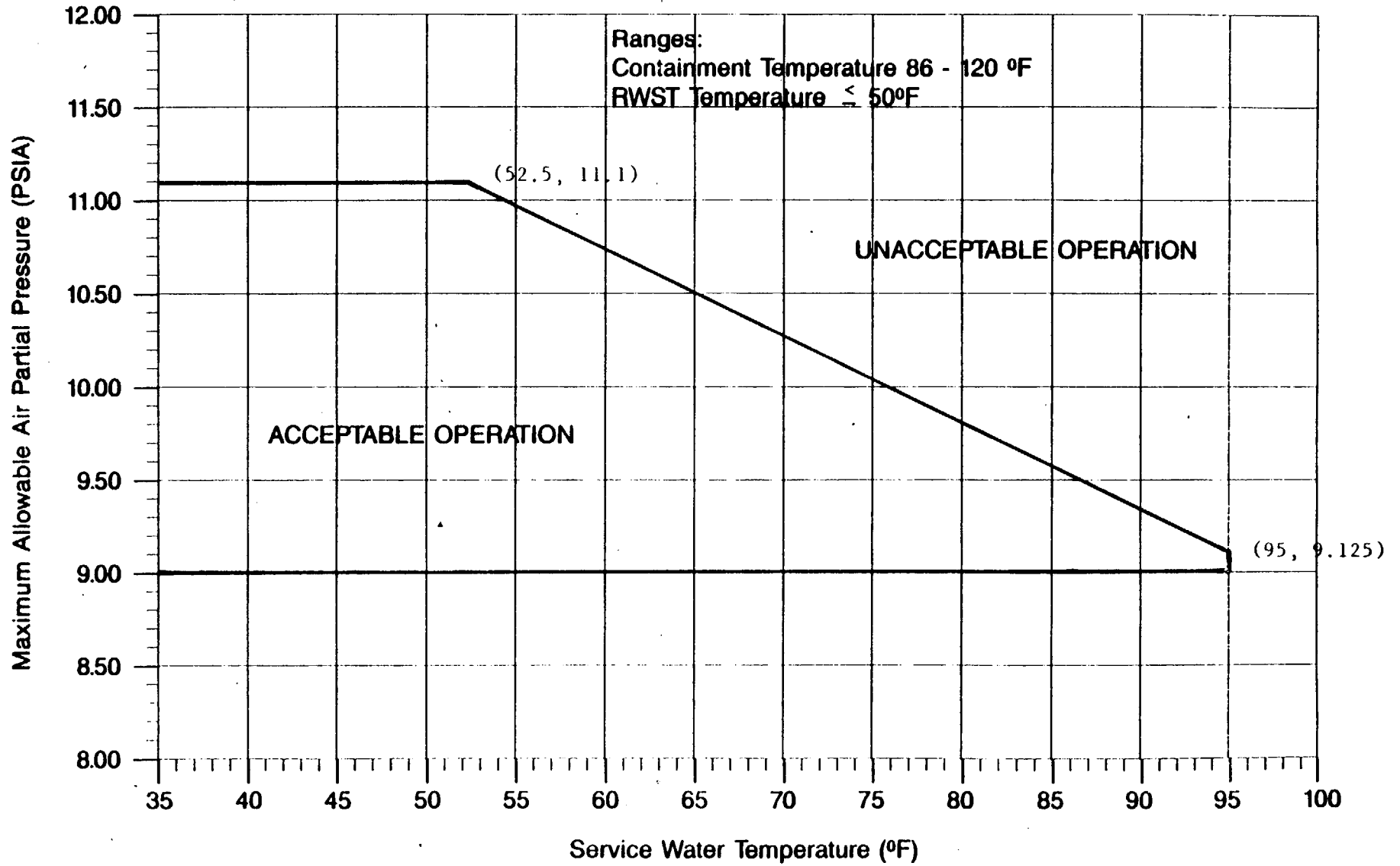


Figure 3.6-1

Containment Air Partial Pressure Versus Service Water Temperature

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be maintained $\geq 86^\circ\text{F}$ and $\leq 120^\circ\text{F}$.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature $> 120^\circ\text{F}$ or $< 86^\circ\text{F}$, restore the average air temperature to within the limit within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The primary containment average air temperature shall be the weighted average of at least the minimum number of temperatures at the following locations and shall be determined at least once per 24 hours:

<u>Location</u>		<u>Weight Factor(WF)</u>	<u>Min. No. of Temperature</u>
a. Containment dome	Elev. ~ 390	0.09604	1
b. Inside crane wall	Elev. ~ 329	0.04846	2
c. Annulus	Elev. ~ 329	0.02256	2
d. Annulus	Elev. ~ 238	0.04972	1
e. Cubicles	Elev. ~ 268	0.06785 (.07513)*	2

4.6.1.5.2 The average containment air temperature shall be determined by the following relationship:

$$T_{\text{containment}} = \frac{1.0}{\left[\sum_{i=1}^n \frac{W_{Fi}}{T_i} \right]} \text{ where}$$

W_{Fi} is the weight factor for the temperature T_i , of the i^{th} temperature measurement.

*Weight factor to be used for pressurizer cubicle at Elev. 268.

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EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for quench spray and between 7.7 and 9.0 for the solution recirculated within the containment after a LOCA. This pH minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

An RWST wide range level instrument loop uncertainty was included in the safety analysis and therefore need not be considered by the operator.

3/4.6 CONTAINMENT SYSTEMS

BASES

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.

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 Q is further limited to $\leq 0.75 L$ 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,

CONTAINMENT SYSTEMS

BASES

- 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 OPERABILITY 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. 96
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 March 2, 1988, as supplemented August 5, 1988, 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. 96, 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 14 days.

FOR THE NUCLEAR REGULATORY COMMISSION



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: December 14, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 96

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.

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EMERGENCY CORE COOLING SYSTEMS

HEAT TRACING

LIMITING CONDITION FOR OPERATION

3.5.4.2 At least two independent channels of heat tracing shall be OPERABLE for the boron injection tank and for the heat traced portions of the associated flow paths.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With only one channel of heat tracing on either the boron injection tank or on the heat traced portion of an associated flow path OPERABLE, operation may continue for up to 30 days provided the tank and flow path temperatures are verified to be greater than or equal to 115°F at least once per 8 hours; otherwise, be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.2 Each heat tracing channel for the boron injection tank and associated flow path shall be demonstrated OPERABLE:

- a. At least once per 31 days by energizing each heat tracing channel and
- b. At least once per 24 hours by verifying the tank and flow path temperatures to be greater than or equal to 115°F. The tank temperature shall be determined by measurement. The flow path temperature shall be determined by either measurement or recirculation flow until establishment of equilibrium temperatures within the tank.

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

- a. A contained borated water volume of between 466,200 and 487,000 gallons.
- b. Between 2300 and 2400 ppm of boron, and
- c. A solution temperature between 40°F and 50°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

- a. At least once per 7 days by:
 1. Verifying the contained borated water volume in the tank, and
 2. Verifying the boron concentration of the water.
- b. At least once per 24 hours by verifying the RWST temperature.

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.

* 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 40.5 psig until completion of the Cycle 6 to 7 refueling outage. ^a Following this outage, P_a shall be 44.1 psig.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- b. If any periodic Type A test fails to meet $.75 L_a$, 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 $.75 L_a$, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet $.75 L_a$ 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:
 - 1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within $0.25 L_a$
 - 2. Has a duration sufficient to establish accurately the change in leakage between the Type A test and the supplemental test.
 - 3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at P_a , greater than or equal to 44.1 psig.
- d. Type B and C tests shall be conducted with gas at P_a , greater than or equal to 44.1 psig, at intervals no greater than 24 months except for tests involving:
 - 1. Air locks,
 - 2. Penetrations using continuous leakage monitoring systems
- e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.
- f. Type B test for penetrations employing a continuous leakage monitoring system shall be conducted at P_a , greater than or equal to 44.1 psig, at intervals no greater than once per 3 years.
- g. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.
- h. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

- a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and
- b. An overall air lock leakage rate of less than or equal to $0.05 L_a$ at P_a , greater than or equal to 44.1 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

- a. With one containment air lock door inoperable:
 1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.+
 2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.
 3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
 4. The provisions of Specification 3.0.4 are not applicable.
- b. With a containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

- a. *Within 72 hours following closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying that the seal leakage is less than $0.01 L_a$ as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a pressure of greater than or equal to 44.1 psig.

+Entry to repair the inner air lock door, if inoperable, is allowed.

*Exempt to Appendix "J" of 10 CFR Part 50.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- b. At least once per 6 months by conducting an overall air lock leakage test at greater than or equal to P_a , 44.1 psig, and by verifying that the overall air lock leakage rate is within its limit[#], and
- c. At least once per 18 months during shutdown by verifying that only one door in each air lock can be opened at a time.

[#]The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal air partial pressure shall be maintained greater than or equal to 9.0 psia and within the acceptable operation on Figure 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal air partial pressure less than 9.0 psia or above the applicable limit shown on Figure 3.6-1, restore the internal air partial pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal air partial pressure shall be determined to be within the limits at least once per 12 hours.

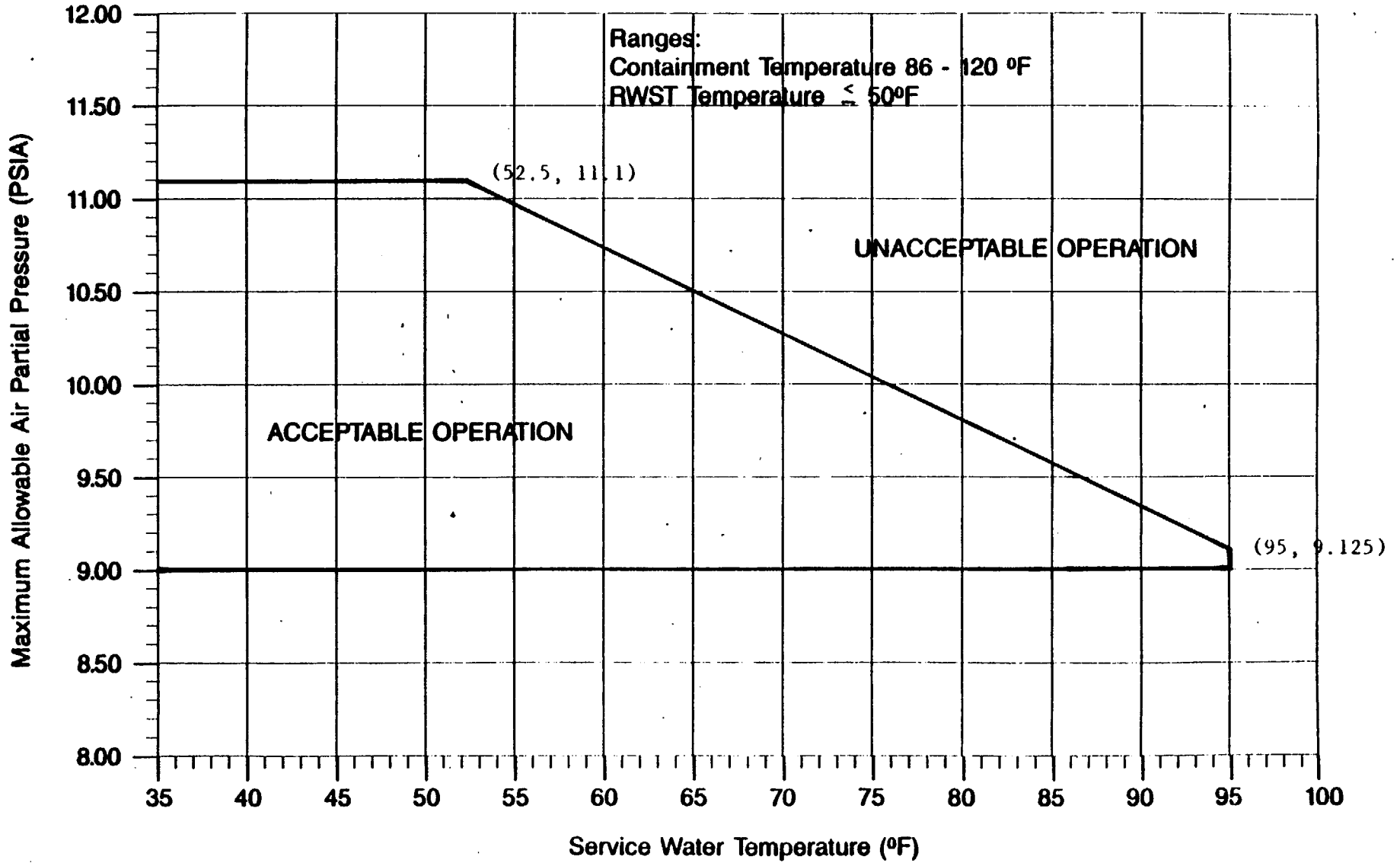


Figure 3.6-1

Containment Air Partial Pressure Versus Service Water Temperature

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be maintained greater than or equal to 86°F and less than or equal to 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature greater than 120°F or less than 86°F, restore the average air temperature to within the limit within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The primary containment average air temperature shall be the weighted average of at least the minimum number of temperatures at the following locations and shall be determined at least once per 24 hours:

<u>Location</u>		<u>Weight Factor(WF)</u>	<u>Min. No. of Temperature Detectors</u>
a. Containment dome	Elev. ~ 390	0.04789	1
b. Inside crane wall	Elev. ~ 329	0.09373	2
c. Annulus	Elev. ~ 329	0.02283 (0.02935)*	2
d. Annulus	Elev. ~ 238	0.08309	1
e. Cubicles	Elev. ~ 268	**	1

4.6.1.5.2 The average containment air temperature shall be determined by the following relationship:

$$T_{\text{containment}} = \frac{1.0}{\sum_{i=1}^n \frac{WF_i}{T_i}} \quad \text{where}$$

WF_i is the weight factor for the temperature T_i , of the i^{th} temperature measurement.

* Weight factor to be used for pressurizer cubicle at Elev. 268.

**Weight factor to be used for cubicles A=0.03932, B=0.03597., C=0.03619

EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for quench spray and between 7.7 and 9.0 for the solution recirculated within the containment after a LOCA. This pH minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

An RWST wide range level instrument loop uncertainty was included in the safety analysis and therefore need not be considered by the operator.

CONTAINMENT SYSTEMS

BASES

- 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. 110 AND 96 TO

FACILITY OPERATING LICENSE NO. 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 March 2, 1988 and as supplemented by letter dated August 5, 1988, the Virginia Electric and Power Company (the licensee) requested changes to the Technical Specifications (TS) for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). The proposed changes would revise the NA-1&2 TS by increasing the maximum allowable temperature inside containment from the present value of 105°F to 120°F. In addition, the volume of water available from the Refueling Water Storage Tank (RWST) for the quench spray system would be redefined and reduced to permit the use of wide range level instrumentation for TS surveillance.

NA-1&2 currently operate within an allowable containment temperature range of 86°F to 105°F. The upper limit temperature (105°F) is approached during the summer months due to high ambient and service water temperatures because there is no practical way to reduce the temperature in a large enclosed volume or a large body of water in short periods of time. The NA-1&2 TS 3.6.1.5 Action Statements require that the containment temperature be restored to within its maximum (105°F) limit within 8 hours or be in at least Hot Standby within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. In order to address these matters, the licensee has been required to implement manpower-intensive procedures during the summer months to prevent exceeding the upper temperature TS limit of 105°F. Therefore, the licensee has submitted a safety evaluation to justify the proposed change which would increase the maximum containment temperature from 105°F to 120°F.

The containment temperature limit is set by performing the necessary transient analyses to ensure that the containment design criteria are met following a design basis accident. Temperature is a significant initial condition for these analyses. Another important analysis input is the volume of water from the RWST available for the quench spray system. Also, an analysis addressing the inadvertent operation of the quench spray system for a containment bulk air temperature of 120°F is required. In addition, the effects on environmentally qualified

electrical equipment (10 CFR 50.49) inside containment must be addressed for the revised upper limit containment temperature and resulting pressure conditions. Finally, containment leak rate tests must be evaluated and compared with required test pressures based on the proposed containment temperature of 120°F.

The NA-1&2 plant design basis was reviewed to determine which transients are impacted by the increased service water temperature and containment bulk temperature. The containment design is based on two Condition IV transients: the Loss of Coolant Accident (LOCA) and the Main Steam Line Break (MSLB). As a result of the worst-case LOCA or MSLB, containment integrity is assured if the following three conditions are satisfied: (1) the peak calculated containment pressure is less than the design pressure of 45 pounds per square inch gauge (psig), (2) the containment is depressurized to subatmospheric within 1 hour of the accident, and (3) once depressurized, the containment is maintained at a pressure less than atmospheric for the duration of the accident.

The Condition II and III transients either do not breach the reactor coolant pressure boundary or are less severe than the LOCA and MSLB transients analyzed as part of the containment design. For the analyses reported in the licensee's submittal several LOCA scenarios were analyzed. The MSLB analysis includes a spectrum of break sizes and power levels which was necessary because of a change in the methodology. The staff's discussion and evaluation of these matters is provided below.

2.0 DISCUSSION

2.1 LOCA Analysis

The containment response to a LOCA was reported in the original NA-1&2 Final Safety Analysis Report (FSAR). Since that time the LOCA has been reanalyzed as a result of changes in the operating conditions of the reactor coolant system. In the recent past the core average temperature was increased a total of 7.5°F and the core power was uprated to 2893 Mwt. In each instance the limiting LOCA break was analyzed to be certain that the containment design criteria were met with the revised operating conditions.

The latest LOCA analysis has used input parameters which support changes in the containment safety analysis for air and service water temperature. The maximum containment temperature was increased from 105°F to 120°F and the maximum service water temperature from 95°F to 97°F. Also, a 2 percent initial power uncertainty was included in the LOCA analysis.

In addition, a lower limit on the RWST water volume was assumed in the LOCA analysis. This lower value would permit the use of wide range level instrumentation for TS surveillance and would mitigate operator actions needed to account for instrument level uncertainty. This lower volume (452,327 gallons) corresponds to a tank level which is 7.5 percent of level instrument span below the upper TS limit of 487,000 gallons.

The analyses considered the fact that a portion of the tank volume (approximately 17,000 gallons) is not available for safety injection and containment spray because of the physical configuration of the tank and pump suction piping. Therefore, the lower limit of water volume (452,327 gallons) was further reduced by 17,000 gallons to a value of 435,361 gallons for use as an input parameter for the LOCA analyses.

The proposed TS lower limit value was determined by adding an appropriate allowance for instrument level measurement uncertainty to the lower RWST water volume discussed above. This uncertainty corresponds to 3 percent of level instrument span. This uncertainty allowance of 13,873 gallons when added to the lower water volume of 452,327 gallons provides a minimum TS limit of 466,200 gallons.

The LOCA analysis consists of a peak pressure, a depressurization analysis and a Net Positive Suction Head Analysis (NPSHA) for the containment spray pumps and the Low Head Safety Injection (LHSI) pumps. A spectrum of analyses were performed to determine the limiting break size and location. The depressurization analysis was performed for the limiting break which is a double-ended rupture at the reactor coolant pump suction (PSDER). The limiting break size and location for the containment spray pump NPSHA was also determined from a spectrum analysis. Therefore, only the double-ended rupture of the hot leg was considered for the recirculation spray pumps. Similarly, the PSDER with minimum Engineered Safety Features (ESF) gives the limiting NPSHA for the LHSI pumps. Each of these scenarios were analyzed with the LOCTIC computer code. The LOCTIC computer code was used in the design basis analysis for NA-1&2 to calculate temperature and pressure of the containment atmosphere as a function of time following a LOCA or a main steam line break (MSLB) inside containment. The LOCTIC computer code is described in Section 6.2.1.1.1.2 of the NA-1&2 Updated Final Safety Analysis Report (UFSAR).

Based on the LOCA analysis, which considered a spectrum of break sizes and locations and different single failure scenarios, the peak containment pressure was found to be 44.1 psig. The length of time to subatmospheric conditions was found to be 3310 seconds. The maximum subatmospheric peak was found to be -0.02 psig. The NPSHA showed 5.8 ft and 2.5 ft margin for the outside and inside recirculation-spray pumps, respectively and 0.1 ft margin for the LHSI pumps.

2.2 Main Steam Line Break (MSLB)

The other design basis event for containment design is the MSLB inside containment. While less mass is present in a steam generator than the reactor coolant system, the fluid enthalpy is much higher so there is no qualitative way to determine which transient, the LOCA or the MSLB, presents more of a challenge to the containment structure and containment safety systems.

The traditional MSLB analysis for containment response considers only the double-ended rupture₂(DER) of the main steam line upstream of the flow restrictor (i.e., 4.6 ft) at hot standby conditions. The hot zero power condition is

more limiting because the specific enthalpy of the fluid in the steam generator is nearly the same as the full power value and the total mass is highest, and therefore, the total energy is highest at the no load condition.

The analysis presented by the licensee for a containment temperature of 120°F differs from that of the traditional approach in one significant way which makes it necessary to consider a spectrum of MSLB scenarios. The mass and energy release data were calculated using the LOFTRAN code which has an NRC-approved entrainment model for steam line break analysis. The LOFTRAN code was found acceptable by the staff in the NRC Safety Evaluation Report transmitted to Westinghouse by letter dated May 27, 1986. The use of an entrainment model requires a spectrum analysis because only larger break sizes contain water entrained with the steam. The amount of entrained water decreases as the break size decreases. The mass and energy data from LOFTRAN were input to the LOCTIC computer code to determine the containment pressure and temperature responses.

The spectrum of breaks necessary to bound the effects of break size and power level on the mass and energy released from a ruptured steam line were defined on the basis of extensive analysis. The postulated break area can have competing effects on blowdown results. Larger breaks are more likely to have water entrainment; however, these breaks also result in earlier protection signal generation. Therefore, for power levels of 102%, 70%, 30% and 0% of nominal full power, five critical break sizes were defined and their characteristics quantified. The break areas analyzed were defined as follows:

- ° A full double-ended rupture at the outlet of one steam generator nozzle;
- ° A full double-ended rupture downstream of the flow restrictor in one steam line;
- ° A small double-ended rupture at the steam generator nozzle having an area just larger than that at which water entrainment occurs;
- ° A small double-ended rupture at the steam generator nozzle having an area just small enough to preclude entrainment; and
- ° A small split rupture that will neither generate a steam line isolation signal from the Westinghouse Solid State Protection System nor result in water entrainment in the break effluent.

The double-ended break sizes were assumed to occur at the outlet of one steam generator and downstream of the flow restrictor. Flow restrictors₂ in the steam line limit the effective area of a full DER to a maximum of 1.4 ft² per steam generator if the break occurs downstream of the restrictors. Upstream, the outlet nozzles of the steam generator limit the effective break area to 4.6 ft².

The key reactor coolant system (RCS) variables are initial power level, RCS pressure, RCS temperature, and RCS loop flow. For this analysis, the standard 2% uncertainty on power level was used. The thermal design flow was assumed along with the nominal RCS pressure of 2250 psia. The RCS average temperature was assumed to be 4°F above the nominal value to account for measurement and control system uncertainties.

The core kinetic parameters were chosen to simulate end-of-cycle conditions with the most reactive rod stuck out of the core. These assumptions maximize the positive reactivity insertion due to moderator feedback during cooldown. Additionally, minimum safety injection was assumed to restrict the flow of borated water to a rate corresponding to the operation of one charging pump. The safety injection lines downstream of the boron injection tank were assumed to have a zero boron concentration. These assumptions minimize the magnitude of the negative reactivity inserted.

The feedwater flow rate was conservatively modeled by assuming an increase in response to the steam line break. For split breaks and small double-ended ruptures, feedwater flow was increased proportionally to the steam line flow increase. For the large double-ended rupture cases the feedwater flow was instantaneously ramped to a maximum of 220% of nominal full feedwater flow in response to the decreasing steam generator pressure.

Various system component failures were evaluated to determine which failure results in the largest increase in releases to the containment. The failure of one safe-guards train to operate was assumed in the analysis along with the failure of the non-return valve in the steam line with the faulted steam generator. The safeguards train failure reduced the boron delivery to the core while the non-return valve failure allowed the steam generators to blowdown until the main steam isolation valves on the intact loops were isolated. Since the main steam trip valves at NA-1&2 do not prevent reverse flow, the nonisolatable volume in the main steam continues to blowdown even after steam line isolation occurs.

Containment response calculations were performed using the LOCTIC computer code to determine pressure and temperature response to a main steam line break inside containment. LOCTIC input included the conservative assumptions as follows:

- ° Eight percent partial revaporization of condensate containment thermodynamic modes,
- ° Failure of the nonreturn valve on the broken steam line to close (the most limiting single failure),
- ° Minimum quench spray failure initiation 60 seconds after containment depressurization actuation (CDA),
- ° Initial containment conditions that yield a maximum pressure and temperature including: air partial pressure, bulk temperature and dewpoint, and
- ° Auxiliary feedwater initiation immediately (0 sec) after the break with a rate of 900 gpm to the affected steam generator.

Mass and energy release rates were generated for the spectrum of breaks required to envelope the effects of break size and power level on the mass and energy released from a ruptured steam line. In most cases, the transients are characterized by a rapid increase in mass flow rate and energy flow rate and lasting a few seconds before beginning to decrease exponentially. The mass flow rate is largest for the 4.6 ft DER breaks and the hot zero power cases. The energy release rate is largest for the 4.6 ft break and for the 102% power cases.

Single failure considerations were applied to the containment response calculations in a manner consistent with the mass and energy release calculations. In particular, the failure of one emergency bus was assumed. For the containment analysis this assumption results in the failure of one inside recirculation spray pump, one outside recirculation spray pump and one quench spray pump. The single failure assumptions made in the LOFTRAN analyses are inherent in the mass and energy data. In effect, two single failures have been included in the main steam line break results: the failure of the non-return valve and the failure of one emergency bus. The non-return valve failure affects the early portion of the transient while the emergency bus failure primarily affects the later stages of the transient.

The loss of offsite power was also treated in a conservative fashion. The mass and energy release calculations assume the reactor coolant pumps remain functional (i.e., no loss of offsite power) to provide the maximum heat transfer from the primary to the secondary system. The containment response calculations provide for the loss of offsite power by assuming containment spray delays representative of a loss of offsite power condition. This approach represents another inherent conservatism in the analysis because the offsite power must affect all of the plant systems consistently.

As noted above, the MSLB scenarios involved several combinations of break size, break type and power level. In all, 20 cases were run for the peak temperature cases and then repeated for the peak pressure cases. The large number of cases were required through the use of an entrainment model which makes it impossible to qualitatively determine the limiting break size and power level. As analyzed, the limiting steam line break for both containment temperature and pressure is the DER of the 30-inch (4.6 ft) line upstream of the flow restrictor at hot zero power. The peak pressure was found to be 44.9 psig and peak containment atmosphere temperature was found to be 357.4°F.

This peak containment temperature of 357.4°F is higher than the containment design temperature of 280°F. In addition, this temperature of 357.4°F is substantially lower than the value of 442°F previously reported in the NA-1&2 FSAR. This lower value is a result of the entrainment assumptions used in the analysis supporting a maximum containment temperature of 120°F. The staff's NA-1&2 SER (related to operation), Supplement 3 (1976), addressed the effect of the peak containment atmosphere exceeding the design temperature on non-electrical equipment inside the containment and on the containment structure itself. On the basis of the staff's review, the staff concluded that the large heat capacity of the containment structure and the relatively short period of time that the peak containment atmosphere temperature is above the containment design temperature precludes any

adverse effects occurring on either non-electrical equipment or the containment structure. Based on the reasons above, the staff has determined that a peak containment atmosphere temperature of 357.4°F will have no adverse effects on either the non-electrical equipment or the containment structure. In addition, the transient peak containment temperature of 357.4°F is significantly lower than the previously reported value of 442°F specified in the NA-1&2 FSAR for the MSLB design basis analysis.

Containment Pressure Analysis For

2.3 Inadvertent Operation of Quench Spray System

An analysis of the inadvertent operation of the quench spray system was performed as part of the analysis supporting a maximum containment temperature of 120°F. The acceptance criteria and method of analysis for the inadvertent operation of the quench spray system are described in Section 6.2.6.3 of the NA-1&2 UFSAR. Supporting calculations for the current containment average air temperature limit of 105°F are presented in UFSAR Table 6.2.-75. Identical calculations in support of the 120°F limit indicate a value of 7.8 psig which provides an adequate margin above the minimum design containment pressure limit of 5.5 psig.

2.4 Equipment Qualification

The effects of the proposed changes in allowable containment temperature and pressure on the environmental qualification of electrical equipment have been evaluated as required by 10 CFR 50.49. The evaluation addressed the following environmental qualification parameters: equipment operation time, accident environmental conditions and effects on equipment qualification, and equipment aging effects including service life and maintenance schedule. An average maximum temperature of 105°F within containment has been used for determining thermal age degradation of safety-related electrical equipment inside containment. Periodic test procedures are currently in place to monitor containment average air temperature and to use the information, as necessary, to adjust equipment service lifetimes and maintenance schedules, as appropriate. These procedures involve the evaluation of containment average air temperature for comparison against an annual operating time versus a containment average air temperature standard. This standard permits quick determination of whether or not the containment operating air temperature history profile is potentially more or less severe in terms of equipment aging than continuous operation of 105°F. The Arrhenius methodology was used to determine the equivalent thermal age degradation associated with operation at different temperatures above and below 105°F. The table below provides the standard for annual operating time versus the containment average air temperature range (°F).

Annual Operating Time
Versus
Containment Average Air Temperature
Standard*

Daily Peak Average Temperature Range (°F)	Number of Days
115 < T ≤ 120	5
112 < T ≤ 115	5
107 < T ≤ 112	5
103 < T ≤ 107	6
100 < T ≤ 103	36
95 < T ≤ 100	158
T ≤ 95	150

*This standard equates to an Equivalent Arrhenius Temperature of 105°F for a period of 365 days.

Based on the above, the containment average air temperature would not be expected to exceed 105°F for more than a few days in any 12 month period. This is based on past NA-1&2 operating experience. In addition, any days of operation above 105°F would be compensated for by many more days of operation at air temperatures below 100°F as shown by past operating experience. Further, even if the standard were to be exceeded by a small amount during a 12 month period, the effects on service lifetimes and maintenance schedules would be small and easily managed within existing preventive maintenance and outage schedules.

2.5 Containment Leak Rate Test Evaluation

The evaluation supporting the NA-1&2 TS change for a maximum containment air temperature of 120°F determined that peak containment pressure following a LOCA is 44.1 psig. Therefore, the licensee proposed that the NA-1&2 TS 3.6.1.2 would increase the test pressure for the integrated and local leak rate tests from the current value of 40.6 psig to 44.1 psig. The containment leak rate tests are required to be performed every 24 months during cold shutdown. The licensee has requested that the proposed change to TS 3.6.1.2 be made effective following startup from the next refueling outage for both NA-1&2. The licensee has evaluated the adequacy of the most recent leak rate testing for NA-1&2. The integrated leak rate test for NA-1 was performed at a pressure (43.7 psig) less than the 44.1 psig. The integrated leak rate test for NA-2 was performed at 44.2 psig. In addition, certain local leak rate tests for both NA-1&2 may have been performed at pressures slightly less than 44.1 psig. The licensee calculated the projected leak rates associated with higher pressure (44.1 psig) from the measured leak rates and found them to be within the proposed TS 3.6.1.2 limit. Based on the above calculation, the staff finds the proposed change to TS 3.6.1.2 to be acceptable.

3.0 PROPOSED TECHNICAL SPECIFICATION CHANGES

Several NA-1&2 TSs need to be changed to incorporate the results of the analyses which justify operation at the increased containment temperature (120°F) and modified RWST level. Each change is discussed separately below.

3/4 6.1.2 Containment Leakage Rates

The limiting condition for operation (LCO) and the surveillance requirement (SR) have been modified to account for the new maximum pressure following a LOCA. The peak pressure resulting from the LOCA analyses is 44.1 psig for a pump suction double-end rupture. This result is incorporated in the LCO and SR.

3/4 6.1.3 Containment Air Lock

The LCO and SR have been modified to account for the new maximum pressure following a LOCA. The peak pressure resulting from the LOCA analysis is 44.1 psig for a pump suction double-end rupture. This result is incorporated in the LCO and SR.

3/4 6.1.4 Internal Air Partial Pressure

The containment internal air partial pressure limit has been revised based on the analytical results justifying operation at the increased containment temperature. The minimum limit is set by the structural criteria for the containment mat. The revised diagonal limit line in TS Figure 3.6-1 is set by the LOCA depressurization analysis. The MSLB analysis sets the horizontal limit line at 11.1 psia. All of the limit lines have been conservatively changed to reflect the containment vacuum and leakage monitoring pressure instrumentation loop uncertainty which is used by the operator for periodic surveillance.

3/4 6.1.5 Containment Air Temperature

One goal of the analyses effort was to increase the maximum containment air temperature from 105°F to 120°F. Since all of the analyses showed acceptable results, the temperature increase was proposed. Therefore, the containment air temperature instrumentation uncertainty was also accommodated.

B3/4.5.5 Refueling Water Storage Tank

The basis for the RWST TS has been modified to state that instrument uncertainty was included in the safety analysis. The operator does not have to account for level uncertainty during surveillance.

B3/4.6.1.4 and B3/4.6.1.5 Internal Pressure and Temperature

The bases for the containment internal pressure and temperature TS has been modified to state that instrument uncertainty was considered in the safety analysis. The operator does not have to account for these uncertainties during surveillance.

B3/4.6.1.6 Containment Structural Integrity

The basis for the structural integrity specification was modified because it contained reference to the numerical value of the peak LOCA temperature. Rather than list the revised value of 44.1 psig the numerical value has been removed. The magnitude of the value is not significant in the context of this specification since the discussion relates to the way containment integrity is maintained.

4.0 SUMMARY

Based on the above, the results indicate that the containment design criteria are not violated at the initial conditions of 120°F containment temperature, 97°F service water temperature and the lower volume of RWST water. Accident consequences are not increased by the proposed TS changes. Instrument uncertainties have been considered in the safety analyses which provides further evidence that accident consequences are not increased by these changes.

The results of the analyses show that none of the containment design bases are violated. That is, the following inequalities remain valid:

- ° Peak pressure: 44.1 psig (LOCA), 44.9 psig (MSLB) < 45 psig
- ° Depressurization to subatmospheric: 3310 seconds < 3600 seconds
- ° Maintain subatmospheric pressure: -0.02 psig < 0.0 psig

Therefore, the containment design criteria specified in the introduction are: (1) peak pressure < 45 psig, (2) depressurize to subatmospheric < 3600 seconds, and (3) maintain pressure subatmospheric for duration of the accident are not violated and a TS change governing containment air temperature and minimum RWST volume is technically acceptable.

Appropriate analysis for input parameters for LOCA analysis has been made to support the RWST wide range level instrumentation for TS surveillance. Additionally, the NPSHA exceeds the Net Positive Suction Head Required (NPSHR) for the recirculation spray pumps and the LHSI pumps. The peak containment temperature from the steam line break is substantially lower than previously reported, even with the revised initial conditions, because of the entrainment assumptions. The inadvertent operation of the quench spray system provides adequate margin above the minimum design containment pressure limit of 5.5 psia. Operability of the electrical equipment within the containment at temperatures up to 120°F has also been considered and found to be acceptable with procedures in place to ensure that maintenance schedules can be adjusted to maintain established equipment lifetimes with an allowable containment temperature up to 120°F. The maximum service temperature, while analyzed at 97°F, will not be changed at this time since the service water system modifications recently implemented can maintain the reservoir temperature below the currently allowable TS value of 95°F. Finally, projected leak rates associated with the higher LOCA pressure (44.1 psig) from the measured leak rates are within the proposed TS limits.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on December 9, 1988 (53 FR 49805). Accordingly, based on the environmental assessment, the Commission has determined that the issuance of these amendments will not have a significant effect on the quality of the human environment.

6.0 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: December 14, 1988

Principal Contributor:

Leon Engle

UNITED STATES NUCLEAR REGULATORY COMMISSION
VIRGINIA ELECTRIC AND POWER COMPANY, ET AL
DOCKET NOS. 50-338 AND 50-339
NOTICE OF ISSUANCE OF AMENDMENTS TO
FACILITY OPERATING LICENSES

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. and to Facility Operating License Nos. NPF-4 and NPF-7, to the Virginia Electric and Power Company (the licensee), which revised the Technical Specifications for operation of the North Anna Power Station, Units 1 and 2 (NA-1&2), located in Louisa County, Virginia. The amendments were effective as of the date of their issuance.

The amendments revised the NA-1&2 TS containment air temperature upper limit from 105°F to 120°F and the volume of water available from the refueling water storage tank for the quench spray system was redefined and reduced to permit the use of wide range level instrumentation for TS surveillance.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notice of Consideration of Issuance of Amendments and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on May 12, 1988 (53 FR 16921).

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Also in connection with this action, the Commission prepared an Environmental Assessment and Finding of No Significant Impact, which was published in the FEDERAL REGISTER on December 9, 1988 (53 FR 49805).

For further details with respect to the action, see (1) the application for amendment dated March 2, 1988, as supplemented August 5, 1988, (2) Amendment Nos. 110 and 96 to Facility Operating License Nos. NPF-4 and NPF-7, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street, N.W., Washington, D.C., and at the Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901.

Dated at Rockville, Maryland this 14th day of December 1988.

FOR THE NUCLEAR REGULATORY COMMISSION



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Office of Nuclear Reactor Regulation