April 1, 1996

Mr. J. P. O'Hanlon
Senior Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

SUBJECT: NORTH ANNA UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: INCREASED PRESSURIZER SAFETY VALVE LIFT SETPOINT TOLERANCE (SERIAL NO. 95-366, TS 330) (TAC NOS. M93005 AND M93006)

Dear Mr. O'Hanlon:

The Commission has issued the enclosed Amendment Nos. 200 and 181 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 July 26, 1995.

The amendments revise the Technical Specifications to increase the pressurizer safety valve lift setpoint tolerance and reduce the pressurizer high pressure reactor trip setpoint and allowable value.

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)

Gordon E. Edison, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-338
and 50-339

Enclosures:
1. Amendment No. 200 to NPF-4
2. Amendment No. 181 to NPF-7
3. Safety Evaluation

cc w/enclosures: See next page

FILENAME - A:\NA93005.AMD

OFFICE	LA:PDII-1	DRPE	PM:PDII-1	OGC	D:PDII-1
NAME	DUNNINGTON	RCLARK/cw	GEDISON	EHoller	GIMBRO
DATE	03/21/96	03/06/96	03/29/96	03/19/96	03/14/96
COPY	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

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

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. 200
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 July 26, 1995, 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. 200 , 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 its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

for

Eugene V. Imbro, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 1, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 200

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.*

Remove Pages	Insert Pages
2-5	2-5*
2-6	2-6
3/4 4-5	3/4 4-5*
3/4 4-6	3/4 4-6
3/4 4-7	3/4 4-7
B 3/4 4-2	B 3/4 4-2
B 3/4 4-2a	B 3/4 4-2a

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER High Setpoint - ≤ 109%** of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER High Setpoint - ≤ 110%*** of RATED THERMAL POWER
3.	Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4.	Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5.	Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6.	Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3 x 10^5 counts per second
7.	Overtemperature ΔT	See Note 1	See Note 3
8.	Overpower ΔT	See Note 2	See Note 3
9.	Pressurizer Pressure – Low	≥ 1870 psig	≥ 1860 psig
10.	Pressurizer Pressure – High	≤ 2360 psig	≤ 2370 psig
11.	Pressurizer Water Level – High	≤ 92% of instrument span	≤ 93% of instrument span
12.	Loss of Flow	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

* Design flow per loop is one-third of the minimum allowable Reactor Coolant System Total Flow Rate as specified in Table 3.2-1.

** The high trip setpoint for Power Range, Neutron Flux, shall be ≤ 103% RATED THERMAL POWER for the period of operation until steam generator replacement.

*** The allowable value for the high trip setpoint for Power Range, Neutron Flux, is required to be ≤ 104% RATED THERMAL POWER for the period of operation until steam generator replacement.

ISOLATED LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.5 A reactor coolant loop cold leg stop valve shall remain closed until:

a. The isolated loop has been operating on a recirculation flow of greater than or equal to 125 gpm for at least 90 minutes and the temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops.

b. The reactor is subcritical by at least 1.77 percent $\Delta k/k$.

APPLICABILITY: ALL MODES.

ACTION:

With the requirements of the above specification not satisfied, suspend startup of the isolated loop.

SURVEILLANCE REQUIREMENTS

4.4.1.5.1 The isolated loop cold leg temperature shall be determined to be within 20°F of the highest cold leg temperature of the operating loops within 30 minutes prior to opening the cold leg stop valve.

4.4.1.5.2 The reactor shall be determined to be subcritical by at least 1.77 percent $\Delta k/k$ within 30 minutes prior to opening the cold leg stop valve.

REACTOR COOLANT SYSTEM

SAFETY VALVES – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± 3% as-found and ± 1% as-left.*

APPLICABILITY: MODE 4.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient condition of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

SAFETY AND RELIEF VALVES - OPERATING

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.4.3.1 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG + 2% / - 3% average as-found with no single valve outside ± 3%, and ± 1% per valve as-left.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal temperature and pressure.

within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water induced reactivity transient.

3/4.4.2 AND 3/4.4.3 SAFETY AND RELIEF VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 380,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during hot shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, or the power operated relief valves (PORVs) will provide overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

The safety valve tolerance requirement for Modes 1-3 is expressed as an average value. That is, the as-found error (expressed as a positive or negative percentage) of each tested safety valve is summed and divided by the number of valves tested. This average as-found value is compared to the acceptable range of + 2% to - 3%. In addition, no single valve is allowed to be outside of ± 3%.

An average tolerance of + 2% / - 3% was confirmed to be adequate for Modes 1-3 accident analyses. For the overpressure events, the analyses considered several combinations of valve tolerance with the arithmetic average of the three valves' tolerance equal to + 2% (with no valve outside of ± 3%). The case of a + 2% tolerance on each of the three valves provided the most limiting results. The - 3% tolerance is limiting for the DNB acceptance criterion.

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

a) Manual control of PORVs to control reactor coolant system pressure. This is a function that may be used to mitigate certain accidents and for plant shutdown.

b) Maintaining the integrity of the reactor coolant pressure boundary. This function is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

c) Manual control of the block valve to (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item a, above), and (2) isolate a PORV with excessive seat leakage (Item b, above).

d) Automatic control of PORVs to control reactor coolant system pressure. This function reduces challenges to the code safety valves for overpressurization events.

e) Manual control of a block valve to isolate a stuck-open PORV.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their functions. Specification 4.4.3.2.1 addresses the PORVs and Specification 4.4.3.2.2 addresses the block valves. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the ACTION requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status.

Surveillance Requirement 4.4.3.2.1.b.2 provides for the testing of the mechanical and electrical aspects of control systems for the PORVs.

Testing of PORVs in HOT STANDBY or HOT SHUTDOWN is required in order to simulate the temperature and pressure environmental effects on PORVs. Testing at COLD SHUTDOWN is not considered to be a representative test for assessing PORV performance under normal plant operating conditions.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system.

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

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. 181
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 July 26, 1995, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 181 , 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 its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

RB Cotton for

Eugene V. Imbro, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 1, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 181

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.*

Remove Pages	Insert Pages
2-6	2-6
3/4 4-5	3/4 4-5*
3/4 4-6	3/4 4-6
3/4 4-7	3/4 4-7
B 3/4 4-2	B 3/4 4-2
B 3/4 4-2a	B 3/4 4-2a

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER High Setpoint - ≤ 109% of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER High Setpoint - ≤ 110% of RATED THERMAL POWER
3.	Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4.	Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5.	Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6.	Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3 x 10^5 counts per second
7.	Overtemperature ΔT	See Note 1	See Note 3
8.	Overpower ΔT	See Note 2	See Note 3
9.	Pressurizer Pressure – Low	≥ 1870 psig	≥ 1860 psig
10.	Pressurizer Pressure – High	≤ 2360 psig	≤ 2370 psig
11.	Pressurizer Water Level – High	≤ 92% of instrument span	≤ 93% of instrument span
12.	Loss of Flow	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

* Design flow per loop is one-third of the minimum allowable Reactor Coolant System Total Flow Rate as specified in Table 3.2-1.

LIMITING CONDITION FOR OPERATION

3.4.1.5 A reactor coolant loop cold leg stop valve shall remain closed until:

a. The isolated loop has been operating on a recirculation flow greater than or equal to 125 gpm for at least 90 minutes and the temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops.

b. The reactor is subcritical by at least 1.77 percent Δk/k.

APPLICABILITY: ALL MODES.

ACTION:

With the requirements of the above specification not satisfied, suspend startup of the isolated loop.

SURVEILLANCE REQUIREMENTS

4.4.1.5.1 The isolated loop cold leg temperature shall be determined to be within 20°F of the highest cold leg temperature of the operating loops within 30 minutes prior to opening the cold leg stop valve.

4.4.1.5.2 The reactor shall be determined to be subcritical by at least 1.77 percent Δk/k within 30 minutes prior to opening the cold leg stop valve.

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± 3% as-found and ± 1% as-left.*

APPLICABILITY: MODE 4.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient condition of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

SAFETY AND RELIEF VALVES – OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3.1 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG + 2% / - 3% average as-found with no single valve outside ± 3%, and ± 1% per valve as-left.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal temperature and pressure.

3/4.4.2 AND 3/4.4.3 SAFETY AND RELIEF VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 380,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during hot shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, or the power operated relief valves (PORVs) will provide overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

The safety valve tolerance requirement for Modes 1-3 is expressed as an average value. That is, the as-found error (expressed as a positive or negative percentage) of each tested safety valve is summed and divided by the number of valves tested. This average as-found value is compared to the acceptable range of + 2% to - 3%. In addition, no single valve is allowed to be outside of ± 3%.

An average tolerance of + 2% / - 3% was confirmed to be adequate for Modes 1-3 accident analyses. For the overpressure events, the analyses considered several combinations of valve tolerance with the arithmetic average of the three valves' tolerance equal to + 2% (with no valve outside of ± 3%). The case of a + 2% tolerance on each of the three valves provided the most limiting results. The - 3% tolerance is limiting for the DNB acceptance criterion.

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

a) Manual control of PORVs to control reactor coolant system pressure. This is a function that may be used to mitigate certain accidents and for plant shutdown.

b) Maintaining the integrity of the reactor coolant pressure boundary. This function is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

c) Manual control of the block valve to (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item a, above), and (2) isolate a PORV with excessive seat leakage (Item b, above).

d) Automatic control of PORVs to control reactor coolant system pressure. This function reduces challenges to the code safety valves for overpressurization events.

e) Manual control of a block valve to isolate a stuck-open PORV.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their functions. Specification 4.4.3.2.1 addresses the PORVs and Specification 4.4.3.2.2 addresses the block valves. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the ACTION requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status.

Surveillance Requirement 4.4.3.2.1.b.2 provides for the testing of the mechanical and electrical aspects of control systems for the PORVs.

Testing of PORVs in HOT STANDBY or HOT SHUTDOWN is required in order to simulate the temperature and pressure environmental effects on PORVs. Testing at COLD SHUTDOWN is not considered to be a representative test for assessing PORV performance under normal plant operating conditions.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE ensures that the plant will be able to establish natural circulation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 200 AND 181 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 July 26, 1995, the Virginia Electric and Power Company (the licensee) proposed an amendment to the North Anna Units 1 and 2 (NA-1&2) Technical Specifications (TS). Specifically, the proposed amendment requested the following changes to the TSs:

Table 2.2-1, Reactor Trip system Instrumentation Trip Setpoints, Item 10, Pressurizer Pressure - High

Revise the existing trip setpoint from ≤ 2385 psig to ≤ 2360 psig.

Revise the allowable value from ≤ 2395 psig to ≤ 2370 psig.

TS 3.4.2, Reactor Coolant System Safety Valves - Shutdown

Revise the safety valve lift setpoint tolerance from ±1% to ±3% as-found and ±1% as-left*.

TS 3.4.3.1, Reactor Coolant System Safety and Relief Valves - Operating

Revise the safety valve lift setpoint tolerance from ±1% to +2%/-3% average as-found with no single valve outside ±3%, and ±1% per valve as-left*.

Bases for TS 3/4.4.2 and 3/4.4.3, Reactor Coolant System Safety and Relief Valves

Add or modify the following paragraphs to the 3/4.4.2 and 3/4.4.3 bases sections:

Page B 3/4 4-2

The safety valve tolerance requirement for Modes 1-3 is expressed as an average value. That is, the as-found error (expressed as a positive or negative percentage) of each tested safety valve is summed and divided by the number of valves tested. This average as-found value is compared to the acceptable range of +2% to -3%. In addition, no single valve is allowed to be outside of ±3%.

An average tolerance of +2%/-3% was confirmed to be adequate for Modes 1-3 accident analyses. For the overpressure events, the analyses considered several combinations of valve tolerance with the arithmetic average of the three valves' tolerance equal to +2% (with no valve outside of ±3%). The case of a +2% tolerance on each of the three valves provided the most limiting results. The -3% tolerance is limiting for the DNB acceptance criterion.

Page B 3/4 4-2a

The OPERABILITY of the PORVs and block values is determined on the basis of their being capable of performing the following functions:

a) Manual control of PORVs to control reactor coolant system pressure. This is a function that may be used to mitigate certain accidents and for plant shutdown.

b) Maintaining the integrity of the reactor coolant pressure boundary. This function is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

2.0 BACKGROUND

Three code safety valves are installed on each unit's pressurizer. The valves have a nominal lift setpoint of 2485 psig and function to protect the reactor coolant system from overpressure.

The licensee stated that the PSVs have a history of drifting outside the currently allowed tolerance of ±1%, resulting in TS violations. Because up to a ±3% tolerance is permitted by ASME Code Section III, Division 1, Subsection NB, Part 7513, for code safety valves, a project was initiated to justify an increase in the PSV tolerance to reduce the number of TS violations. The analyses and evaluation described support the proposed PSV setpoint tolerance increase.

The licensee stated that the proposed TS changes do not affect the nominal lift setpoint of the pressurizer safety valves, nor the as-left tolerance requirement. Only the allowable as-found tolerance about the existing lift setpoint is to be changed.

To ensure acceptable analysis results with the increased as-found PSV tolerance, a concurrent reduction in the pressurizer high pressure reactor trip TS setpoint is also proposed by the licensee. This reduction provides a faster response of the reactor protection system to over pressure events without significantly impacting existing operating margin.

3.0 EVALUATION

VEPCO made analysis of the Loss of Load, Locked Rotor, and Rod Withdrawal events to demonstrate that increasing the at-power PSV lift setpoint tolerance to +2%/-3% average as-found with no single valve outside ±3% as-found and ±1% per valve as-left does not result in a transient pressure in excess of the overpressure safety limit. The transient analyses were performed with the RETRAN system transient analysis code. The evaluation of these events is summarized below.

3.1 Loss of Load

The Loss of Load analysis was performed to establish that a Loss of Load event would not result in primary side pressures beyond the limit of 2750 psia nor secondary side pressure beyond the limit of 1210 psia when the pressurizer safety valve lift setpoint tolerance is increased to 2%. The maximum primary side (cold leg) pressure was determined to be 2740 psia which is below the overpressure safety limit (110% of design pressure) of 2750 psia. The peak secondary pressure was 1181 psia which is below the acceptance criterion of 1210 psia. We find the results to be acceptable as the pressure values fall within the overpressure safety limits.

3.2 Locked Rotor Analysis

The Locked Rotor Analysis was performed in order to determine if an increased PSV value lift setpoint tolerance would result in an overpressurization of the primary side during a postulated Locked Rotor transient. The transient analysis using the RETRAN code for the Locked Rotor event with a 2% average PSV setpoint tolerance calculated a peak primary (cold leg) pressure of 2739 psia. This value is below the primary safety limit of 2750 psia. The maximum secondary side pressure was determined to be 1186 psia, which below the overpressure limit of 1210 psia. We find the results to be acceptable as the pressure values fall within the overpressure safety limits.

3.3 Rod Withdrawal Events

The licensee stated that recent reanalyses of the Rod Withdrawal at Power (RWAP) and Rod withdrawal from Subcritical (RWSC) events revealed that these events may result in significant pressurization of the RCS, particularly those cases initiated from low power.

The impact of a 3% PSV lift setpoint tolerance (bounding the 2% average tolerance) on RWAP results was quantified by the licensee. The limiting case was initiated from 8% power, and assumed a 30 pcm/sec reactivity insertion rate, a 3% PSV lift setpoint tolerance, a water loop seal (additional opening delay), and a -1.4 pcm/°F full power Doppler temperature coefficient. This case resulted in maximum RCS pressure of 2725 psia.

Similarly, the impact of 3% PSV lift setpoint tolerance on RWSC results was quantified. A case which assumed a 100 pcm/sec reactivity insertion rate, a 3% PSV lift setpoint tolerance, and a water loop seal was run. The peak RCS pressure in the analysis was 2587 psia.

The results for both the RWSC and RWAP are below 110% of the RCS design pressure, and are therefore acceptable.

3.4 DNB Considerations

An increased negative PSV lift setpoint tolerance potentially reduces the system pressure experienced at the point of minimum Departure from Nucleate Boiling Ratio (DNBR). Therefore the effect of a -3% PSV setpoint tolerance on the DNBR results of affected transients was evaluated by examining the North Anna USFAR Chapter 15 safety analysis results.

Of the affected transients, only the DNBR results of the Locked Rotor event were found to be potentially adversely affected by the increased negative tolerance. A conservative maximum impact on the Locked Rotor analysis was quantified by the licensee and the DNBR acceptance criteria was found to be met. We therefore find this acceptable.

3.5 Operational Margin Considerations

The licensee stated that the proposed setpoint tolerances were chosen such that an inadvertent opening of the safety valves during normal operation would not occur. The proposed high primary pressure trip setpoint is 2370 psig with an uncertainty of 18.72 psi. The nominal setpoint plus uncertainty is, therefore, 2389 psig. Because the nominal PSV lift setpoint minus 3% tolerance is 2425 psia, a reactor trip will occur before the PSVs open. It is therefore concluded that the proposed setpoint tolerance change will not present any operational considerations.

3.6 Mode 4 Considerations

The licensee calculated the shutdown overpressure protection requirements. The analysis used a tolerance of +3% on the pressurizer safety valve. Tolerance in the negative direction provides additional margin. The analysis showed that for two charging pumps injecting at double the flow of a single pump, two PSVs provide adequate overpressure protection. Therefore, for the case of one operable charging pump, as required in Mode 4 and below, one PSV will provide adequate overpressure protection with a tolerance of up to +3%. Therefore, the Mode 4 requirement (i.e., TS 3.4.2) is specified as ±3%, which we find to be acceptable.

3.7 Summary

The Loss of Load, Locked Rotor, and Rod Withdrawal event analyses demonstrate that increasing the at-power PSV lift setpoint tolerance to +2%/-3% <u>average</u> as-found with no single valve outside ±3% as-found and ±1% per valve as-left does not result in a transient pressure in excess of the overpressure safety limit. In addition, the licensee's analysis has shown that the increased setpoint tolerance does not adversely impact the DNBR results of an North Anna UFSAR Chapter 15 transient analyses. Mode 4 overpressure protection has been shown to be adequate with one PSV with a tolerance of ±3%. The reduction in the pressurizer high pressure reactor trip setpoint ensures that the analysis results for the loss of external load accident continue to meet the acceptance criteria with the higher PSV tolerance. The increased PSV setpoint tolerances and reduction of the high pressurizer pressure reactor trip setpoint do not present any operational considerations which would significantly impact the performance of the plant during normal operation or during postulated accident conditions. Each pertinent safety criteria was evaluated by the licensee for the proposed TS changes, and all were found to be acceptable. Our review has found that these proposed changes are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC 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 issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (60 FR 45189). Accordingly, these 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.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: H. Balukjian

Date: April 1, 1996

DATED: April 1, 1996

AMENDMENT NO. 200 TO FACILITY OPERATING LICENSE NO. NPF-4-NORTH ANNA UNIT 1
AMENDMENT NO. 181 TO FACILITY OPERATING LICENSE NO. NPF-7-NORTH ANNA UNIT 2

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