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

November 20, 2012

Mr. David A. Heacock
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS TO ELIMINATE THE STEAM GENERATOR WATER LEVEL LOW COINCIDENT WITH STEAM FLOW/FEEDWATER FLOW MISMATCH REACTOR TRIP (TAC NOS. ME8566 AND ME8567)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 268 and 249 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Units 1 and 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 2, 2012, as supplemented by letter dated August 6, 2012.

The amendments revise the TSs by deleting the Steam Generator Water Level Low Coincident with Steam Flow/Feedwater Flow Mismatch Reactor Trip Function from the TS Table 3.3.1-1, Item 15.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Dr. V. Sreenivas, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 268 to NPF-4
2. Amendment No. 249 to NPF-7
3. Safety Evaluation

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 268
Renewed 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 April 2, 2012, as supplemented by letter dated August 6, 2012, 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 paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-4, as indicated in the attachment to this license amendment, and is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 268, are hereby incorporated in the renewed 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 before exiting the fall 2013 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-4
and the Technical Specifications

Date of Issuance: November 20, 2012

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 249
Renewed 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 April 2, 2012, as supplemented by letter dated August 6, 2012, 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 paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-7, as indicated in the attachment to this license amendment, and is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 249, are hereby incorporated in the renewed 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 before exiting the spring 2013 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-7
and the Technical Specifications

Date of Issuance: November 20, 2012

ATTACHMENT

TO LICENSE AMENDMENT NO. 268

RENEWED FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

AND

TO LICENSE AMENDMENT NO. 249

RENEWED FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the Licenses and the Appendix "A" Technical Specifications (TSs) with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
Licenses	Licenses
License No. NPF-4, page 3	License No. NPF-4, page 3
License No. NPF-7, page 3	License No. NPF-7, page 3
TSs	TSs
3.3.1-15	3.3.1-15

(2) Pursuant to the Act and 10 CFR Part 70, VEPCO to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30,40, and 70, VEPCO to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material, without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or component; and

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, VEPCO to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

VEPCO is authorized to operate the North Anna Power Station, Unit No.1, at reactor core power levels not in excess of 2940 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 268 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material, without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30,40, and 70, VEPCO to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations as set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

VEPCO is authorized to operate the facility at steady state reactor core power levels not in excess of 2940 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 249 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the condition or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the renewed license supported by a favorable evaluation by the Commission:

a. If VEPCO plans to remove or to make significant changes in the normal operation of equipment that controls the amount of radioactivity in effluents from the North Anna Power Station, the

Table 3.3.1-1 (page 3 of 5)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
15. Deleted					
16. Turbine Trip					
a. Low Auto Stop Oil Pressure	1(h)	3	N	SR 3.3.1.10 SR 3.3.1.15	≥ 40 psig
b. Turbine Stop Valve Closure	1(h)	4	N	SR 3.3.1.10 SR 3.3.1.15	≥ 0% open
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1, 2	2 trains	O	SR 3.3.1.14	NA
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2(d)	2	Q	SR 3.3.1.11 SR 3.3.1.13	≥ 3E-11 amp
b. Low Power Reactor Trips Block, P-7	1	1 per train	R	SR 3.3.1.5	NA
c. Power Range Neutron Flux, P-8	1	4	R	SR 3.3.1.11 SR 3.3.1.13	≤ 31% RTP
d. Power Range Neutron Flux, P-10	1, 2	4	Q	SR 3.3.1.11 SR 3.3.1.13	≥ 7% RTP ≤ 11% RTP
e. Turbine Impulse Pressure, P-13	1	2	R	SR 3.3.1.10 SR 3.3.1.13	≤ 11% turbine power
19. Reactor Trip Breakers(i)	1, 2	2 trains	P	SR 3.3.1.4	NA
	3(a), 4(a), 5(a)	2 trains	C	SR 3.3.1.4	NA
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1, 2	1 each per RTB	S	SR 3.3.1.4	NA
	3(a), 4(a), 5(a)	1 each per RTB	C	SR 3.3.1.4	NA
21. Automatic Trip Logic	1, 2	2 trains	O	SR 3.3.1.5	NA
	3(a), 4(a), 5(a)	2 trains	C	SR 3.3.1.5	NA

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(h) Above the P-8 (Power Range Neutron Flux) interlock.

(i) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 268

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-4

AND

AMENDMENT NO. 249

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By application dated April 2, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12094A307), as supplemented by letter dated August 6, 2012 (ADAMS Accession No. ML12220A355), Virginia Electric and Power Company (the licensee) submitted a request for changes to the North Anna Power Station, Units 1 and 2 (NAPS 1 and 2), Technical Specifications (TSs). The requested change would eliminate the Steam Generator Water Level Low Coincident with Steam Flow/Feedwater Flow Mismatch Reactor Trip Function from each of the respective TSs. The request for change is based upon the installation of new equipment which is intended to eliminate the circumstance that warranted inclusion of function 15 into TS Table 3.3.1-1.

2.0 REGULATORY EVALUATION

The following regulatory bases and guidance documents pertain to the proposed TS change.

Title 10 of the *Code of Federal Regulations* (10CFR), Part 50, Section 50.36(c)(2)(i), states the following:

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

10 CFR 50.36(c)(2)(ii) states the following:

> A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:
>
> (A) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
> (B) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
> (C) *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
> (D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

10 CFR 50.36(c)(3) states the following:

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

10 CFR 50.55a(h)(2) states the following:

> *Protection systems.* For nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, protection systems must meet the requirements stated in either IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations, . . ."

3.0 <u>TECHNICAL EVALUATION</u>

As described in the license amendment request (LAR) (ADAMS Accession No. ML12094A307), the current steam generator level sensor configuration of the North Anna units includes three sensors that supply data on Steam Generator level to the protection system. One of these sensors also supplies data to the unit control system. This particular configuration triggers Clause 4.7.3 of IEEE 279, since a failure of the sensor that feeds both control and protection systems would be considered the initiating event (due to a potential control system response to the failure) and, thus, the licensee is required to consider a failure of a second sensor that provides data to the protection system. The licensee's approach to addressing this situation was to include function 15 in TS Table 3.3.1-1, which requires a trip based upon a (single) low steam generator level sensor indication combined with a mismatch between steam and feedwater flow. The staff reviewed Sections 7.2.1.1.5 and 7.2.2.3.5, Item 1.c of the North Anna Updated Final Safety Analysis Report (UFSAR) (Revision 47), and the description of the steam generator related trip conditions is consistent with the LAR.

The LAR (ADAMS Accession No. ML12094A307) describes installation of a median signal selector (MSS) which would accept all three-steam generators level sensor readings and pass the "middle" value to the control system. The licensee's amendment proposed that this new equipment would eliminate the initiating condition (i.e., a single failure of a sensor that causes a control system reaction) that necessitated function 15 in TS Table 3.3.1-1.

IEEE 279-1971: Clause 4.7.3

In the current North Anna plant configuration, one of the three steam generator level indicators is used to supply data to both the protection and feedwater control system. [The other two steam generator level indicator sensors only supply the data to the protection system (and not the control system).] In the event that the sensor used for both control and protection was to fail high, the control system would respond by reducing feedwater flow to the steam generator. In this instance, Clause 4.7.3 of IEEE Standard 279-1971 specifies that the design should still be capable of protecting against the condition even when degraded by a second (postulated) random failure. The proper response for the protection system in this case (i.e., reduced feedwater flow culminating in a low-low steam generator level) would be a reactor trip (i.e., function 14 in North Anna TS Table 3.3.1-1).

However, if the second failure postulated (per Clause 4.7.3) involves one of the other steam generator level sensors "failed in place," the low steam generator level trip would not actuate – i.e., two out of three voting with one sensor failed high, one failed "in place" and one correctly registering a low level. To comply with the IEEE 279 clause, an additional trip was incorporated into the North Anna TS - Steam Generator Water Level Low Coincident with Steam Flow/Feedwater Flow Mismatch Reactor Trip Function, which is function 15 in TS Table 3.3.1-1.

The proposed change to the North Anna units introduces a MSS into the logic of the feedwater control system. With the proposed change all three steam generator level signals would pass through the protection system and each signal would pass into the new MSS. The MSS selects the "middle" of the three signals provided to it and passes that data along for use in feedwater control. The licensee's amendment proposes that, with the new MSS installed, a single failure of a sensor will not result in a feedwater control event. Without the initial sensor failure initiating the transient event, a second single random failure need not be considered per IEEE 279-1971, Clause 4.7.3.

The staff considered the instances where a single steam generator level indication channel either fails high or fails low. In either case, the MSS signal selector would reject both the high and low signals and relay the "middle" signal to feedwater control. With the use of the MSS, the failed high or low signal would not be the signal selected for control and, thus, would not initiate the transient that function 15 in TS Table 3.3.1-1 is designed to protect against. In the instance of a sensor failed in place, no transient would be initiated even if it happens to be the "middle" signal.

The staff also considered the potential for the MSS to "mask" sensor failures, since a failed sensor may not immediately cause a change in plant operations. The licensee confirmed in their response to staff request for additional information (RAI #2) (ADAMS Accession No. ML12220A355) that their adherence to the TS Surveillance Requirement 3.3.1.1 as mandated by function 14 (Steam Generator Water Level – Low Low) of Table 3.3.1-1 remains unchanged by the addition of the MSS. In accordance with North Anna Surveillance Requirement (SR) 3.3.1.1, a formal Channel Check of the three steam generator level sensor readings is performed at least every 12 hours.

The LAR also describes the failure mechanisms of the MSS. All of the failure modes can be identified via on-line testing. Functional and calibration tests will be performed quarterly and every 18 months, respectively. These functional and calibration tests will be documented in the North Anna Technical Requirements Manual (TRM). Capturing these controls in the TRM is appropriate, as the MSS is strictly part of the control system and does not perform a protection function, and as MSS failure is not an accident indicator.

Failure of the MSS power supply will result in operation analogous to the current plant configuration – i.e., passing of a single steam generator level signal for feedwater control. In the event of this failure, a control room alarm will annunciate and the TRM will preclude plant operations with automatic steam generator water level control.

Although not specifically requested to be reviewed in the LAR, North Anna UFSAR, Section 7.2.2.3.5, Item 2 cites an exception that was taken to Clause 4.7.3 in the event that the single steam generator level channel used for feedwater control (in the current configuration) fails in a specific manner. Based on its review of the proposed modification, the staff believes that the proposed modification should significantly reduce the potential for the postulated transient described in North Anna, UFSAR Section 7.2.2.3.5, Item 2.

For the reasons described above, the staff agrees that the installation of the MSS alleviates the potential for a single steam generator sensor failure causing a transient in the feedwater flow to the steam generators. The staff concludes that this change both enhances the safety of the plant and eliminates the need for function 15 (Steam Generator Water Level Low Coincident with Steam Flow/Feedwater Flow Mismatch Reactor Trip Function) in TS Table 3.3.1-1.

IEEE 279-1971: Clause 4.7.2

Clause 4.7.2 of IEEE 279-1971 requires that any signals routed from protection system equipment for control systems use shall be via isolation devices, which are to be classified as part of the protection system. The staff inquired about how the licensee was providing isolation between the protection system and the control system.

As stated in the LAR and clarified in the licensee's response to request for supplemental information (RAI #1) (ADAMS Accession No. ML12220A355), the Westinghouse 7300 Process Protection Racks of the reactor protection system contain qualified isolation devices for each of the three channels for each steam generator. The staff reviewed a figure depicting the signal routing and isolation device that the licensee provided with the request for additional information (RAI) response, and is itconsistent with the textual description provided.

The staff reviewed North Anna UFSAR, Section 7.7.2.1 which describes the separation of protection and control systems, and the description is consistent with the LAR and RAI response. In summary, the isolation provided between the protection system and the new MSS is equivalent to the isolation previously provided for the single channel that was used to supply data to the feedwater control system.

The staff also examined the precedent safety evaluations for Beaver Valley 2 (ADAMS Accession Nos. ML003772209/ML003772207) and Beaver Valley 1 (ADAMS Accession No. ML011410204). Both credited qualified isolation devices for providing compliance with Clause 4.7.2. The Beaver Valley 2 safety evaluation specifically credits the Westinghouse 7300 for providing the isolation.

Because the amendment proposes the use of the same qualified isolation devices for routing steam generator level indication signals from the protection system to the control system currently in use, the staff finds that the installation of the MSS will not impact plant compliance with Clause 4.7.2 of IEEE 279-1971.

10 CFR 50.36(c)(2)(ii)

The staff reviewed Chapter 15 of the North Anna UFSAR for any instances of the Steam Generator Water Level Low Coincident with Steam Flow/Feedwater Flow Mismatch Reactor Trip Function being credited.

The following analyses mention the steam flow/feedwater flow mismatch conditions:

- Section 15.2.8 describes loss of normal feedwater in which the Steam Generator Water Level Low Coincident with Steam Flow/Feedwater Flow Mismatch Reactor Trip Function could be actuated. However, the reactor trip on low-low steam generator water level (function 14 of TS Table 3.3.1-1) is given as the primary means of tripping the reactor in this instance.
- Section 15.2.10 addresses excessive heat removal due to feedwater system malfunctions. A mismatch in steam and feedwater flow is mentioned; however, only the reactor trip on low-low steam generator water level (function 14 of TS Table 3.3.1-1) is credited.
- Section 15.4.2.2, which describes the major rupture of a main feedwater pipe, specifically mentions the Steam Generator Water Level Low Coincident with Steam Flow/Feedwater Flow Mismatch Reactor Trip Function; however, it is mentioned as one of five potential conditions that would trip the reactor.
- Section 15.4.3 describes a steam generator tube rupture event. A steam flow/feedwater flow mismatch condition is noted, but a low pressurizer pressure trip is credited with reactor shutdown.

Based on the foregoing, the staff concludes that the accident analyses, in Chapter 15 of the North Anna UFSAR do not credit Steam Generator Water Level Low Coincident with Steam Flow/Feedwater Flow Mismatch Reactor Trip Function, and that this trip function therefore need not be included in the TS based upon the requirements of 10 CFR 50.36(c)(2)(ii).

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 comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components 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 the amendments involve no significant hazards considerations, and there has been no public comment on such finding (77 FR 35076). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). 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

Since the installation of the MSS obviates the need for a Steam Generator Water Level Low Coincident with Steam Flow/Feedwater Flow Mismatch Reactor Trip Function and no other accident analyses credit that function, the staff concludes that North Anna may remove function 15 from Table 3.3.1-1 of its TSs.

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

Principal Contributors: Timothy Mossman, NRR/DE
Sheiba Tafazzoli, NRR/DE

Date: November 20, 2012

November 20, 2012

Mr. David A. Heacock
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS TO ELIMINATE THE STEAM GENERATOR WATER LEVEL LOW COINCIDENT WITH STEAM FLOW/FEEDWATER FLOW MISMATCH REACTOR TRIP (TAC NOS. ME8566 AND ME8567)

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A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Dr. V. Sreenivas, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
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3. Safety Evaluation

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