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

March 9, 2010

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: CORRECTION TO AMENDMENT NOS. 258 AND 259 FOR NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS TO ADOPT TSTF-490, REVISION 0, REGARDING DELETION OF E BAR DEFINITION AND REVISION TO REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS (TAC NOS. ME0264 AND ME0265)

Dear Mr. Heacock:

On March 3, 2010, the U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 258 to Renewed Facility Operating License No. NPF-4 and Amendment No. 259 to Renewed Facility Operating License No. NPF-7 for the North Anna Power Station, Unit Nos. 1 and 2, respectively. These amendments were in response to your application dated December 17, 2008, as supplemented by letters dated January 26, May 26, and November 23, 2009.

These amendments revised the license and Technical Specifications (TSs) to reflect changes in the adoption of Technical Specification Task Force (TSTF)-490, Revision 0, "Deletion of E Bar Definition and Revision to RCS [reactor coolant system] Specific Activity Technical Specification," for pressurized water reactor Standard Technical Specifications (STS). The NRC staff has corrected the Unit 2 Amendment to reflect the correct Amendment No. which should be 239.

Enclosed are the corrected amendment page, license page, errata sheet, TS pages 1.1-2, 1.1-3, 1.1-4, 1.1-5, 1.1-6, 3.4.16-1, 3.4.16-2, and page 1 of the Safety Evaluation (SE). The NRC regrets any inconvenience this may have caused. The revisions are identified by lines in the margins.

Sincerely,

Karen Cotton for

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. 239 to NPF-7
2. License page
3. Errata Sheet
4. TS pages
5. Page 1 of SE

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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. 239
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 December 17, 2008, as supplemented by letters dated January 26, May 26, and November 23, 2009, 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.

Corrected by letter dated March 9, 2010

2. Accordingly, the license is amended by changes to paragraph 2.C.(1) 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. 239, 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 within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Gloria Kulesa, 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: March 3, 2010

Corrected by letter dated March 9, 2010

ATTACHMENT

TO LICENSE AMENDMENT NO. 258

RENEWED FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

AND

TO LICENSE AMENDMENT NO. 239

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
1.1-2	1.1-2
1.1-3	1.1-3
1.1-4	1.1-4
1.1-5	1.1-5
1.1-6	1.1-6
3.4.16-1	3.4.16-1
3.4.16-2	3.4.16-2
3.4.16-3	—

Corrected by letter dated March 9, 2010

(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 No239 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

Corrected by letter dated March 9, 2010

1.1 Definitions

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using the thyroid dose conversion factors listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or in Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

1.1 Definitions

DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

(continued)

1.1 Definitions

LEAKAGE (continued)

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE–OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train,

(continued)

1.1 Definitions

OPERABLE–OPERABILITY (continued)	component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Tests and Operation, of the UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2940 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

(continued)

Corrected by letter dated March 9, 2010

1.1 Definitions

SHUTDOWN MARGIN (SDM) (continued)	a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing all slave relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.

Corrected by letter dated March 9, 2010

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	------------NOTE------------ LCO 3.0.4.c is applicable. ----------------------------	
	A.1 Verify DOSE EQUIVALENT I-131 $\leq$ 60 μCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	------------NOTE------------ LCO 3.0.4.c is applicable. ----------------------------	
	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	6 hours
OR	AND	
DOSE EQUIVALENT I-131 > 60 μCi/gm.	C.2 Be in MODE 5.	36 hours

Corrected by letter dated March 9, 2010

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 197 μCi/gm.	7 days
SR 3.4.16.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

Corrected by letter dated March 9, 2010

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 258

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-4

AND

AMENDMENT NO. 239

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By letter dated December 17, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML083530982), Virginia Electric and Power Company (the licensee) submitted a request for changes to the North Anna Power Station, Unit Nos. 1 and 2 (NAPS 1 and 2), Technical Specifications (TSs). The requested changes are the adoption of Technical Specification Task Force (TSTF)-490, Revision 0, "Deletion of E-Bar Definition and Revision to Reactor Coolant System (RCS) Specific Activity Technical Specification," to the pressurized water reactor (PWR) Standard Technical Specifications (STSs). Subsequently, the licensee's original license amendment request (LAR) was supplemented by letters dated January 26 (ADAMS Accession No. ML090260528), May 26 (ADAMS Accession No. ML091460589), and November 23, 2009 (ADAMS Accession No. ML093280238). The supplements dated January 26, May 26, and November 23, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the original proposed no significant hazards consideration determination.

By letter dated September 13, 2005 (ADAMS Accession No. ML052630462), the TSTF submitted TSTF-490 for Nuclear Regulatory Commission (NRC) staff review. The TSTF with proposed changes for incorporation into the STS as TSTF-490, Revision 0, was referenced in the *Federal Register* Notice (FRN) (71 FR 67170) of November 20, 2006, for public comments. By FRN (72 FR 12217) dated March 15, 2007 (ADAMS Accession No. ML070250176), the NRC published a "Notice of Availability Model Application Concerning Technical Specification Improvement Regarding Deletion of E Bar Definition and Revision to Reactor Coolant System Specific Activity Technical Specification Using the Consolidated Line Item Improvement Process." This TSTF involves changes to NUREG-1430, NUREG-1431, and NUREG-1432 STS Section 3.4.16.

Corrected by letter dated March 9, 2010

Enclosed are the corrected amendment page, license page, errata sheet, TS pages 1.1-2, 1.1-3, 1.1-4, 1.1-5, 1.1-6, 3.4.16-1, 3.4.16-2, and page 1 of the Safety Evaluation (SE). The NRC regrets any inconvenience this may have caused. The revisions are identified by lines in the margins.

Sincerely,

/RA/

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. 239 to NPF-7
2. License page
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4. TS pages
5. Page 1 of SE

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Amendment No.: ML100670110

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