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

May 14, 2015

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, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING ALTERNATE CONTROL ROD POSITION MONITORING (TAC NOS. MF5683 AND MF5684)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 273 and 255 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station (NAPS), Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TSs) in response to your application dated February 4, 2015.

These amendments revise Technical Specification 3.1.7, "Rod Position Indication," to include an additional monitoring option for an inoperable control rod position indicator. Specifically, the proposed changes would allow monitoring of control rod drive mechanism stationary gripper coil voltage every eight hours as an alternative to using the movable incore detectors every eight hours to verify control rod position.

A copy of the Safety Evaluation is also enclosed. A 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. 273 to NPF-4
2. Amendment No. 255 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. 273
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 (the licensee) dated February 4, 2015, 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 hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-4 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 273 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 30 days of issuance.

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: May 14, 2015

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. 255
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 (the licensee) dated February 4, 2015, 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

   C. 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.

Enclosure 2

2. Accordingly, the license is hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-7 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 255 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 30 days of issuance.

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: May 14, 2015

ATTACHMENT TO

LICENSE AMENDMENT NO. 273

RENEWED FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

AND

TO LICENSE AMENDMENT NO. 255

RENEWED FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the Renewed Facility Operating Licenses and the Appendix "A" Technical Specifications (TSs) with the enclosed pages as indicated. The revised pages are identified by amendment number and contain marginal 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
Remove	Insert
3.1.7-1	3.1.7-1
3.1.7-2	3.1.7-2
3.1.7-3	3.1.7-3
3.1.7-4	3.1.7-4

(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. 273 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. 255 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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Rod Position Indication (RPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RPI per group inoperable for one or more groups.	A.1 Verify the position indirectly of the rods with inoperable position indicators by using movable incore detectors.	Once per 8 hours
	<u>OR</u>	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
	<u>OR</u> ---------NOTE--------- Rod position monitoring by Action A.3.1 and A.3.2 may be applied to only one inoperable rod position indicator and shall be allowed until an entry into MODE 5. -----------------------------	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3.1 Verify the position of the rod with the inoperable rod position indication indirectly by using the movable incore detectors.	Within 8 hours of condition entry (or rod control system indication of potential rod movement) AND Once per 31 days thereafter
	AND	
	A.3.2 Review the parameters of the rod control system for indications of rod movement for the rod with an inoperable position indicator.	16 hours AND Once per 8 hours thereafter
B. More than one RPI per group inoperable.	B.1 Place the control rods under manual control.	Immediately
	AND	
	B.2 Monitor and record RCS T_{avg}.	Once per 1 hour
	AND	
	B.3 Verify the position of the rods with inoperable position indicators indirectly by using the movable incore detectors.	Once per 8 hours
	AND	
	B.4 Restore inoperable position indicator to OPERABLE status such that a maximum of one RPI per group is inoperable.	24 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	C.1 Verify the position indirectly of the rods with inoperable position indicators by using movable incore detectors.	4 hours
	<u>OR</u>	
	C.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
D. One demand position indicator per bank inoperable for one or more banks.	D.1.1 Verify by administrative means all RPIs for the affected banks are OPERABLE.	Once per 8 hours
	<u>AND</u>	
	D.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart.	Once per 8 hours
	<u>OR</u>	
	D.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Perform CHANNEL CALIBRATION of each RPI.	In accordance with the Surveillance Frequency Control Program

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 273 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-4

AND

AMENDMENT NO. 255 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 application dated February 4, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15041A667), the Virginia Electric and Power Company (the licensee) requested changes to the Technical Specifications (TSs) for Facility Operating License Nos. NPF-4 and NPF-7 for North Anna Power Station (NAPS), Units 1 and 2 (NAPS1 and NAPS2), respectively.

The requested amendment would revise TS 3.1.7, "Rod Position Indication," to allow for the use of an alternate method, other than the movable incore detectors, to confirm rod position when the rod position indication is inoperable. Monitoring of the control rod drive mechanism (CRDM) stationary gripper coil voltage will confirm rod position has not changed, once verified by the movable incore detectors. The use of this alternate method will reduce the required frequency of flux mapping using the movable incore detectors to determine the position of the non-indicating rod, thus reducing the wear on the movable incore detector system that is also used to complete other required TS surveillances. In addition, the licensee requested that an asterisk associated with a cycle specific requirement in Condition D of TS 3.1.7 and the associated note be deleted since NAPS2 is no longer in Cycle 22.

2.0 REGULATORY EVALUATION

In Title 10 of the *Code* of *Federal Regulations* (10 CFR) Section 50.36, the Commission established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TS.

As stated in 10 CFR 50.36(c)(2)(i), the "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met." Criterion 2 of 10 CFR 50.36(c)(2)(ii) requires an LCO to be established for 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.

General Design Criterion (GDC) 13 in 10 CFR 50, Appendix A, specifies that instrumentation shall be provided to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions. NAPS TS 3.1.7 requires operability of the Rod Position Indication (RPI) system and the Demand Position Indication System, and thereby ensures compliance with the control rod alignment and insertion limits. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the licensee's license amendment request (LAR) to verify that the licensing basis criteria stated in the Updated Final Safety Analysis Report (UFSAR) continues to meet the instrumentation monitoring requirements in GDC 13 with these proposed changes.

3.0 TECHNICAL EVALUATION

3.1 Background

The objectives of the rod control system and rod position indication system are to ensure that control rod alignment and insertion limits are maintained. Operators utilize the RPI system to monitor the positions of the rods to establish that the plant is operating within the bounds of the accident analysis assumptions. Operability, including position indication, of the control rods and shutdown rods is an initial condition assumption in all safety analyses that assume rod insertion upon a reactor trip. Maximum rod misalignment is an initial condition assumption in the safety analysis that directly affects core power distributions and assumptions of available shutdown margin (SDM). Control rod inoperability or misalignment may cause increased power peaking due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown.

The axial position of control or shutdown rods is indicated by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the RPI System. The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch).

The RPI system provides an accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. The analog signal is produced for each rod by a linear variable differential transformer. Direct continuous readout of each rod is provided by individual meter indications on the Control Room benchboard. The RPI system is capable of monitoring rod position within at least ± 12 steps.

Operators use the RPI system to monitor the position of the rods to establish that the plant is operating within the bounds of the accident analysis. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident with control or shutdown rods operating outside their limits undetected.

3.2 Description of Proposed TS Change

The proposed change to TS 3.1.7 will add new actions that allow the use of an alternative rod position monitoring method. New Actions A.3.1 and A.3.2 will be added to provide for the alternative monitoring. The new requirements are stated in the licensee's LAR as follows:

> Action A.3.1
> Verify the position of the non-indicating rod indirectly by using the movable incore detectors.
>
> A.3.1 Completion Time:
> Once within 8 hours of condition entry (or rod control system indication of potential rod movement) and every 31 days thereafter.
>
> Action A.3.2
> Review the parameters of the rod control system for indications of rod movement for the rod with an inoperable position indicator.
>
> A.3.2 Completion Time:
> Once within 16 hours and every 8 hours thereafter.
>
> A Note is added that will apply to the new Actions A.3.1 and A.3.2. This note describes the limitations for use of these new provisions. The note states:
>
> > Rod position monitoring by Actions A.3.1 and A.3.2 may be applied to only one inoperable rod position indicator and shall only be allowed until an entry into MODE 5.

The licensee also proposes to delete an asterisk associated with a cycle specific requirement in Condition D and the associated note because NAPS2 is no longer in Cycle 22.

3.3 Evaluation of Proposed TS Change

The proposed change will provide an alternative to the use of the movable incore detectors every 8 hours for an extended period of time until repairs can be completed. The parameter monitored for a control rod or shutdown rod with an inoperable position indicator will be the stationary gripper coil voltage. The control rods are held in place by energized stationary gripper coils. The control rod cannot be moved without de-energizing the stationary gripper coil.

The gripper coil voltage will be monitored on a temporary digital recorder. There will be alarm capability to inform the control room operators that the rod moved based on a change in CRDM stationary gripper coil voltage. Operators will verify and record the CRDM stationary gripper coil voltage in accordance with the Completion Time of Required Action. This will allow for trending and historical data retrieval.

NAPS proposes to continue monitoring to determine if the coils have changed position on a once every 8 hour basis. Should the parameters of the coils of the monitored rod indicate movement, a determination of the position of the rod will be made using the movable incore detectors within 8 hours. This timeframe is consistent with existing TS 3.1.7 Action A.1. Since verification with incore detectors is performed within 8 hours, continued monitoring of rod control system parameters is not required until 16 hours after rod movement is identified as stated in proposed Action A.3.2. Continued implementation of TS 3.1.7 Action A.1 would result in at least 90 operations of the incore detector system per month and may result in excessive wear on the system. Although wear of the incore detector system does not pose a significant reduction in the margin of safety, excessive wear could result in a loss of system functionality. This could lead to the inability to complete required surveillances. In practice, this would occur 8 hours after verification is performed with movable incore detectors. Compliance with either Action A.1 or the proposed Action A.3 will result in the verification of the position of the affected rod within 8 hours by use of the movable incore detectors.

To provide a verification of the reliability of the alternate monitoring system, rod position is verified by moveable incore monitoring every 31 days. This frequency minimizes use of the movable incore monitoring system and can be performed concurrently with existing surveillance requirements for Hot Channel Factors. The Hot Channel Factors are safety limits measured by the incore detector system under TS 3.2.1, "Heat Flux Hot Channel Factor," and TS 3.2.2, "Nuclear Enthalpy Ride Hot Channel Factor." SRs for these TSs require measurement of the Hot Channel Factors every 31 effective full power days.

Part of the RPI system is located inside containment. Repair of the RPI coils would require removal of the reactor head missile shield assembly that provides a duct system for the control rod drive mechanism cooling air flow. Inspection access doors are provided, however repair of the rod position indicator coil cannot be performed with the head assembly package in place. Repairs cannot be performed until the reactor head assembly is removed and the unit is in the cold shutdown condition (Mode 5). The note added to TS 3.1.7 is to allow for monitoring until the plant enters Mode 5 and repair of the RPI indication can be safety performed. This ensures that the alternate monitoring may be in place for one inoperable indicator for at most one refueling cycle (18 months) or when the plant enters Mode 5 for a forced outage, whichever comes first.

The following operational events were reviewed by the NRC staff when considering the proposed TS change:

1. Dropped Rod or Rod Misalignment During Power Operations

A rod drop is described in Section 15.2.3 of the UFSAR for NAPS as dropping of a full-length rod cluster control assembly (RCCA) when the drive mechanism is de-energized. A dropped RCCA would cause a power reduction and an increase in the hot channel factor. It is classified as a Condition II event of moderate frequency.

If a RCCA drops into the core during power operation, it would be detected by a rod bottom signal, by an ex-core nuclear instrument, or by both. The RPI system senses each RCCA's position and provides a rod bottom signal for any dropped RCCA. If the RPI is inoperable, the UFSAR describes that an independent indication of a dropped RCCA is derived from the excore power range nuclear instruments. This rod drop detection circuit is actuated when a rapid decrease in the local neutron flux is sensed in any of the four channels. The circuitry is designed to accommodate normal load variations in order to avoid spurious actuation. A rod drop signal from the rod position indication channel or from one or more of the four power range channels blocks further automatic rod withdrawal by the rod control system. Core parameters such as core power and average temperature would also have a noticeable change in the case of a full rod drop and would initiate operator action.

Rod misalignment is detectable by a change in axial flux, channel deviation, and control rod detection alarms. After identifying a RCCA group misalignment condition, the operator must take action as required by the plant TSs and operating instructions. The initiation of operator action in the event of a misaligned control rod is independent of the status of the RPI system.

Based on the available indications and alarms, the likelihood of not detecting a rod drop or misalignment during power operation while alternate monitoring is in place is negligible. Therefore, the NRC staff finds that the design basis analyses of a rod drop or misalignment during power operation remain acceptable.

2. Dropped Rod or Rod Misalignment During Reactor Startup

If an unplanned outage that does not result in an entry into Mode 5 and repairs of the inoperable RPI are not possible in another operating Mode, the licensee plans to use alternate monitoring methods to determine control rod position. Since the movable incore detectors cannot be used to determine rod position until sometime after entry into Mode 2 when neutron flux becomes adequate, CRDM traces will be used as the alternate method to verify that the rod is fully withdrawn. Rod position verification using this method will permit startup and entry into Mode 2. The movable incore detectors will be used to verify rod position when neutron flux becomes adequate.

Following verification that the rod is withdrawn, a rod misalignment would be detectable by means other than the RPI system by using CRDM trace monitoring, axial flux difference, channel deviation, and/or rod control stationary regulation failure alarms. The required operator actions would be independent of the status of the individual rod position indication from RPI. Based on the available indications and alarms, as well as diverse verification methods, the increase in the likelihood of an undetected rod drop or misalignment during reactor startup is negligible. Therefore, the NRC staff finds that the design basis analyses of a rod drop or misalignment during power operation remain acceptable.

3. Reactor Trip

Following a reactor trip, the position indication system is used to verify that all rods have fully inserted. Boration is required if one or more rods fails to fully insert. If it cannot be verified with RPI indication that all rods are on the bottom, then the licensee use TS 5.4 procedures. The operators require to respond to the reactor trip by entering Emergency Operating Procedure (EOP) Emergency Subprocedure (ES)-0.1, "Reactor Trip Response," which requires boration to account for each rod not fully inserted. Also, if there are one or more untrippable rods, TS 3.1.4, "Rod Group Alignment Limits," requires boron initiation to restore SDM to within limits specified in the core operating limit report (COLR) and to be in MODE 3 within six hours. Therefore, the NRC staff finds that there are adequate controls to provide reasonable assurance that the plant will achieve subcriticality following a reactor trip.

4. SDM

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operation occurences.

SDM in Modes 1 and 2 with $k_{eff} \geq 1.0$ is ensured by verifying that control bank and shutdown bank rods are within limits specified in the COLR as described is TS 3.1.5, "Shutdown Bank Insertion Limit" and TS 3.1.6, "Control Rod Bank Insertion Limit." This is done by monitoring the rod insertion limit. A description of the rod insertion limit circuit was described in detail in the February 4, 2015, LAR (Accession No. ML15041A667).

> The rod insertion limit (RIL) circuit is designed to provide operators with a continuously calculated insertion limit for each of the control banks that is variable with coolant loop average differential temperature (delta-T) power. The RIL circuitry provides alarms to assist operators in ensuring operation above the RIL. The RIL circuit performs its function by receiving control bank position pulse data from the rod control system. It compares this data to the calculated limit that is determined by reactor power as measured from delta-T. The rod insertion limits ensure that adequate shutdown margin exists to shutdown the reactor at any time and condition in the life of the operating cycle. The RIL monitoring circuit is independent of the rod position indication circuit. Inoperability of the rod position indicator has no impact on the RIL monitoring system and therefore the inoperable rod position indicator has no impact on the ability of operators to verify Shutdown Margin via the RIL monitoring circuitry. The proposed alternate method to monitor CRDM stationary gripper coil voltage for a control rod with an inoperable rod position indicator will provide assurance that the rod position has not changed and remains within the allowed misalignment with the group step counter demand position for the affected control rod and the control rod bank insertion limits of TS 3.1.5 and 3.1.6.

The rod insertion limit monitoring circuit is independent of the rod position indication circuit. Inoperability of the rod position indication has no impact on the rod insertion limit monitoring and therefore has no impact on the ability of operators to verify SDM. The alternate monitoring method, which will monitor the stationary gripper coil for the control rod with an inoperable RPI, will provide assurance that the position has not changed and remains within the allowed misalignment for the affected control rod and the control rod bank insertion limits of TS 3.1.5 and TS 3.1.6. Therefore, based on as stated above, the NRC staff concludes that there are adequate controls, while operating in Modes 1 and 2 with $k_{eff} \geq 1.0$, to provide reasonable assurance that the plant will continue to achieve subcriticality during a reactor trip.

The licensee's LAR also states that in accordance with TS 3.1.1, the SDM in Mode 2 with $K_{eff} < 1.0$ is determined by comparing the RCS boron concentration to a shutdown margin requirement curve. While in Mode 2, the RPI system is relied upon to determine rod position. While a single rod position indicator remains out of service, rod position cannot be easily determined. Accordingly, the RCS boron concentration requirements will be increased to consider an allowance for the withdrawn worth of a control rod with an inoperable position indicator. In Modes 3, 4, and 5, the RPI system group step counters are relied upon to determine rod position in accordance with Technical Requirement 3.1.3. In Modes 3, 4, and 5, the individual rod position indicators are not used to determine shutdown margin.

In summary, under the proposed TS change, the non-indicating rod would be treated as if it has not fully inserted on a reactor trip and operators will take actions as currently driven by procedures to safely shut down the reactor. Therefore, while operating in Mode 2 with $k_{eff} < 1.0$ and Modes 3, 4, and 5, the NRC staff concludes that there are adequate controls to provide reasonable assurance that the plant will continue to achieve subcriticality upon a reactor shutdown.

As a result, the NRC staff concludes that the use of an alternate method for monitoring the non-indicating rod position provides an acceptable process for knowing the non-indicating rod position and therefore continues to meet the instrumentation monitoring requirements in GDC 13.

In addition, NAPS requested that an asterisk associated with a cycle specific requirement in Condition D of TS 3.1.7 and the associated note be deleted. Because NAPS2 is no longer in Cycle 22, the NRC staff finds that deletion of an asterisk associated with a cycle specific requirement in Condition D of TS 3.1.7 and the associated note on the bottom of the TS 3.1.7 page is acceptable.

3.4 Conclusion

The NRC staff concludes that the alternate monitoring method, which provides continuous monitoring, and the movable incore detector method, which provides monitoring on an intermittent basis, in conjunction with each other, provides an adequate monitoring process associated with an inoperable RPI. Therefore, based on the evaluation described above, the NRC staff has determined that the proposed LAR is acceptable because: (1) the TS changes provide adequate controls to ensure that the rod position is known, (2) any rod misalignment is detectable for the one rod with an inoperable RPI, and (3) the TS requires operators to take appropriate action to ensure that the rod stays within its alignment limit and that SDM is maintained.

In addition, the NRC staff determines that the completion times of Action A.3.1 and Action A.3.2 discussed in Section 3.2 above are acceptable because they are consistent with the TS discussed in Section 3.3.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified on March 3, 2015, of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements 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 the amendments involve no significant hazards consideration, and there has been no public comment on such finding (80 FR 11488, March 3, 2015). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

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) there is reasonable assurance that 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: V. Sreenivas
D. Palmrose

Date: May 14, 2015

May 14, 2015

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, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING ALTERNATE CONTROL ROD POSITION MONITORING (TAC NOS. MF5683 AND MF5684)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 273 and 255 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station (NAPS), Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TSs) in response to your application dated February 4, 2015.

These amendments revise Technical Specification 3.1.7, "Rod Position Indication," to include an additional monitoring option for an inoperable control rod position indicator. Specifically, the proposed changes would allow monitoring of control rod drive mechanism stationary gripper coil voltage every eight hours as an alternative to using the movable incore detectors every eight hours to verify control rod position.

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,
/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. 273 to NPF-4
2. Amendment No. 255 to NPF-7
3. Safety Evaluation

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