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

June 25, 2009

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: SURRY POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING ROD POSITION AND BANK DEMAND POSITION INDICATION SYSTEMS (TAC NOS. MD8925 AND MD8926)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 265 to Renewed Facility Operating License No. DPR-32 and Amendment No. 264 to Renewed Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TSs) in response to your application dated June 9, 2008.

These amendments revise action statements in TS 3.12 for insertion limit and shutdown margin requirements, revise the applicability for the operability of the rod position indication and bank demand position indication systems, revise/add action statements for rod position indication, and add action statements for group step demand counters.

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,

for Robert Martin

~~John~~ Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosures:
1. Amendment No. 265 to DPR-32
2. Amendment No. 264 to DPR-37
3. Safety Evaluation

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 265
Renewed License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated June 9, 2008, 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.

Enclosure

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-32 is hereby amended to read as follows:

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

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 265 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Lois M James for

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. DPR-32
and the Technical Specifications

Date of Issuance: June 25, 2009.

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 264
Renewed License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated June 9, 2008, 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.

Enclosure

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-37 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 264 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Lois James for

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes License No. DPR-37
and the Technical Specifications

Date of Issuance: June 25, 2009.

ATTACHMENT

TO LICENSE AMENDMENT NO. 265

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

AND

TO LICENSE AMENDMENT NO. 264

RENEWED FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

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

Remove Pages	Insert Pages
License	License
License No. DPR-32, page 3	License No. DPR-32, page 3
License No. DPR-37, page 3	License No. DPR-37, page 3
TSs	TSs
TS 3.12-1	TS 3.12-1
------	TS 3.12-1A
TS 3.12-7	TS 3.12-7
TS 3.12-8	TS 3.12-8
TS 3.12-9	TS 3.12-9
TS 3.12-11	TS 3.12-11
TS 3.12-12	TS 3.12-12
-------	TS 3.12-12A
TS 3.12-13	TS 3.12-13
TS 3.12-14	TS 3.12-14
TS 4.10-1	TS 4.10-1

Enclosure

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

   A. <u>Maximum Power Level</u>

      The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2546 megawatts (thermal).

   B. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 265 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

   C. <u>Reports</u>

      The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. <u>Records</u>

      The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

   E. Deleted by Amendment 65

   F. Deleted by Amendment 71

   G. Deleted by Amendment 227

   H. Deleted by Amendment 227

   I. <u>Fire Protection</u>

      The licensee shall implement and maintain in effect the provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report and as approved in the SER dated September 19, 1979, (and Supplements dated May 29, 1980, October 9, 1980, December 18, 1980, February 13, 1981, December 4, 1981, April 27, 1982, November 18, 1982, January 17, 1984, February 25, 1988, and

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

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

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2546 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 264 , are hereby incorporated in this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. Deleted by Amendment 54

F. Deleted by Amendment 59 and Amendment 65

G. Deleted by Amendment 227

H. Deleted by Amendment 227

3.12 CONTROL ROD ASSEMBLIES AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rod assemblies and power distribution limits.

Objective

To ensure core subcriticality after a reactor trip, a limit on potential reactivity insertions from hypothetical control rod assembly ejection, and an acceptable core power distribution during power operation.

Specification

A. Control Bank Insertion Limits

1. Whenever the reactor is critical, except for physics tests and control rod assembly surveillance testing, each shutdown bank shall be within the insertion limits specified in the CORE OPERATING LIMITS REPORT. With one or more shutdown banks not within limits:

   a. Within 1 hour, verify shutdown margin is within the limits specified in the CORE OPERATING LIMITS REPORT or initiate boration to restore shutdown margin to within limit and

   b. Within 2 hours, restore shutdown banks to within limits.

   If the above requirements are not met, be in HOT SHUTDOWN within 6 hours.

2. Whenever the reactor is critical, except for physics tests and control rod assembly surveillance testing, the full length control banks shall be within the insertion limits specified in the CORE OPERATING LIMITS REPORT. With control bank insertion limits not met:

   a. Within 1 hour, verify shutdown margin is within the limits specified in the CORE OPERATING LIMITS REPORT or initiate boration to restore shutdown margin to within limit and

b. Within 2 hours, restore control banks to within limits.

If the above requirements are not met, be in HOT SHUTDOWN within 6 hours.

3. The Control Bank Insertion Limits shown in the CORE OPERATING LIMITS REPORT may be revised on the basis of physics calculations and physics data obtained during unit startup and subsequent operation, in accordance with the following:

5. The allowable QUADRANT POWER TILT is 2.0% and is only applicable while operating at THERMAL POWER > 50%.

6. If, except for operation at THERMAL POWER < 50% or for physics and control rod assembly surveillance testing, the QUADRANT POWER TILT exceeds 2%, then:

   a. Within 2 hours, either the hot channel factors shall be determined and the power level adjusted to meet the requirement of Specification 3.12.B.1, or

   b. The power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours.

   c. If the QUADRANT POWER TILT exceeds 10%, the power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT within the next 30 minutes. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours.

7. If, except for operation at THERMAL POWER < 50% or for physics and control rod assembly surveillance testing, after a further period of 24 hours, the QUADRANT POWER TILT in Specification 3.12.B.5 above is not corrected to less than 2%:

   a. If the design hot channel factors for RATED POWER are not exceeded, an evaluation as to the cause of the discrepancy shall be made and a special report issued to the Nuclear Regulatory Commission.

   b. If the design hot channel factors for RATED POWER are exceeded and the power is greater than 10%, then the high neutron flux, Overpower ΔT and Overtemperature ΔT trip setpoints shall be reduced 1% for each percent the hot channel factor exceeds the RATED POWER design values within the next 4 hours, and the Nuclear Regulatory Commission shall be notified.

c. If the hot channel factors are not determined, then the Overpower DT and Overtemperature ΔT trip setpoints shall be reduced by the equivalent of 2% power for every 1% QUADRANT POWER TILT within the next 4 hours, and the Nuclear Regulatory Commission shall be notified.

C. Control Rod Assemblies

1. To be considered OPERABLE during startup and POWER OPERATION each control rod assembly shall:

    1) be trippable,

    2) aligned within ± 12 steps or ± 24 steps of its group step demand position, as defined in Section 3.12.E.1.b, and

    3) have a drop time of less than or equal to 2.4 seconds to dashpot entry.

2. To be considered OPERABLE during shutdown modes, each control rod assembly shall:

    1) be trippable, and

    2) have a drop time of less than or equal to 2.4 seconds to dashpot entry.

3. Startup and POWER OPERATION may continue with one control rod assembly inoperable provided that within one hour either:

    a. The control rod assembly is restored to OPERABLE status, as defined in Specification 3.12.C.1 and 2, or

    b. the shutdown margin requirement of Specification 3.12.A.3.c is satisfied. POWER OPERATION may then continue provided that:

        1) either:

(a) power shall be reduced to less than 75% of RATED POWER within one (1) hour, and the High Neutron Flux trip setpoint shall be reduced to less than or equal to 85% of RATED POWER within the next four (4) hours, or

(b) the remainder of the control rod assemblies in the group with the inoperable control rod assembly are aligned to within 12 steps of the inoperable rod within one (1) hour while maintaining the control rod assembly sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT; the THERMAL POWER level shall be restricted pursuant to Specification 3.12.A during subsequent operation.

2) the shutdown margin requirement of Specification 3.12.A.3.c is determined to be met within one hour and at least once per 12 hours thereafter.

3) the hot channel factors are shown to be within the design limits of Specification 3.12.B.1 within 72 hours. Further, it shall be demonstrated that the value of $F_{xy}(Z)$ used in the Constant Axial Offset Control analysis is still valid.

4) a reevaluation of each accident analysis of Table 3.12-1 is performed within 5 days. This reevaluation shall confirm that the previous analyzed results of these accidents remain valid for the duration of operation under these conditions.

E. <u>Rod Position Indication System and Bank Demand Position Indication System</u>

1. From movement of control banks to achieve criticality and with the REACTOR CRITICAL, rod position indication shall be provided as follows:

    a. Above 50% power, the Rod Position Indication System shall be OPERABLE and capable of determining the control rod assembly positions to within ± 12 steps of their respective group step demand counter indications.

    b. From movement of control banks to achieve criticality up to 50% power, the Rod Position Indication System shall be OPERABLE and capable of determining the control rod assembly positions to within ± 24 steps of their respective group step demand counter indications for a maximum of one hour out of twenty-four, and to within ± 12 steps otherwise.

    c. From movement of control banks to achieve criticality and with the REACTOR CRITICAL, the Bank Demand Position Indication System shall be OPERABLE and capable of determining the group demand positions to within ± 2 steps.

2. If one rod position indicator per group for one or more groups is inoperable, the position of the control rod assembly shall be verified indirectly using the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating control rod assembly exceeding 24 steps. Alternatively, reduce power to less than 50% of RATED POWER within 8 hours. During operations below 50% of RATED POWER, no special monitoring is required.

3. If more than one rod position indicator per group is inoperable, place the control rods under manual control immediately, monitor and record RCS T_{avg} once per hour, verify the position of the control rod assemblies indirectly using the movable incore detectors at least once per 8 hours, and restore inoperable position indicators to OPERABLE status such that a maximum of one position indicator per group is inoperable within 24 hours.

4. If 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, verify the position of the control rod assemblies indirectly using the movable incore detectors within 4 hours or reduce power to less than 50% of RATED POWER within 8 hours.

5. If one group step demand counter per bank for more than one or more banks is inoperable, verify that all rod position indicators for the affected bank(s) are OPERABLE once per 8 hours and verify that the most withdrawn rod and the least withdrawn rod of the affected bank(s) are less than or equal to 12 steps apart once per 8 hours. Alternatively, reduce power to less than 50% of RATED POWER within 8 hours.

6. If the requirements of Specification 3.12.E.2, 3.12.E.3, 3.12.E.4, or 3.12.E.5 are not satisfied, then the unit shall be placed in HOT SHUTDOWN within 6 hours.

F. <u>DNB Parameters</u>

1. The following DNB related parameters shall be maintained within their limits during POWER OPERATION:

- Reactor Coolant System $T_{avg} \leq 577.0°F$
- Pressurizer Pressure $\geq$ 2205 psig
- Reactor Coolant System Total Flow Rate $\geq$ 273,000 gpm

a. The Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once every 12 hours.

b. The Reactor Coolant System Total Flow Rate shall be determined to be within its limit by measurement at least once per refueling cycle.

2. When any of the parameters in Specification 3.12.F.1 has been determined to exceed its limit, either restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED POWER within the next 4 hours.

3. The limit for Pressurizer Pressure in Specification 3.12.F.1 is not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED POWER per minute or a THERMAL POWER step increase in excess of 10% of RATED POWER.

G. Shutdown Margin

1. Whenever the reactor is subcritical, the shutdown margin shall be within the limits specified in the CORE OPERATING LIMITS REPORT. If the shutdown margin is not within limits, within 15 minutes, initiate boration to restore shutdown margin to within limits.

Basis

The reactivity control concept assumed for operation is that reactivity changes accompanying changes in reactor power are compensated by control rod assembly motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to COLD SHUTDOWN) are compensated for by changes in the soluble boron concentration. During POWER OPERATION, the shutdown control rod assemblies are fully withdrawn and control of power is by the control banks. A reactor trip occurring during POWER OPERATION will place the reactor into HOT SHUTDOWN. The control rod assembly insertion limits provide for achieving HOT SHUTDOWN by reactor trip at any time, assuming the highest worth control rod assembly remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted control rod assembly worth in the unlikely event of a hypothetical assembly ejection and provide for acceptable nuclear peaking factors. The limit may be determined on the basis of unit startup and operating data to provide a more realistic limit which will allow for more flexibility in unit operation and still assure compliance with the shutdown requirement.

The maximum shutdown margin requirement occurs at end of core life and is based on the value used in the analyses of the hypothetical steam break accident. The control rod assembly insertion limits are based on end of core life conditions. The shutdown margin for the entire cycle length is established at 1.77% reactivity. Other accident analyses with the exception of the Chemical and Volume Control System malfunction analyses are based on 1% reactivity shutdown margin. Relative positions of control banks are determined by a specified control bank overlap. This overlap is based on the consideration of axial power shape control. The specified control rod assembly insertion limits have been established to limit the potential ejected control rod assembly worth in order to account for the effects of fuel densification. The various control rod assemblies (shutdown banks, control banks A, B, C, and D) are each to be moved as a bank; that is, with each assembly in the bank within one step (5/8 inch) of the bank position.

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called the group step demand counters) and the Rod Position Indication System.

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one group step demand counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step demand counter for that group. The Bank Demand Position Indication System is considered highly precise (± 2 steps).

The Rod Position Indication System provides an accurate indication of actual rod position, but at a lower precision than the group step demand counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. The Rod Position Indication System is capable of monitoring rod position within at least ± 12 steps during steady state temperature conditions and within ± 24 steps during transient temperature conditions. Below 50% RATED POWER, a wider tolerance on indicated rod position for a maximum of one hour in every 24 hours is permitted to allow the system to reach thermal equilibrium. This thermal soak time is available both for a continuous one hour period or several discrete intervals as long as the total time does not exceed 1 hour in any 24 hour period and the indicated rod position does not exceed 24 steps from the group step demand counter position.

The requirements on the rod position indicators and the group step demand counters are only applicable from the movement of control banks to achieve criticality and with the REACTOR CRITICAL, because these are the only conditions in which the rods can affect core power distribution and in which the rods are relied upon to provide required shutdown margin. The various action statement time requirements are based on operating experience and reflect the significance of the circumstances with respect to verification of rod position and potential rod misalignment. Reduction of RATED POWER to less than or equal to 50% puts the core into a condition where rod position is not significantly affecting core peaking factors. Therefore, during operation below 50% RATED POWER, no special monitoring is required. In the shutdown conditions, the operability of the shutdown banks and control banks has the potential to affect the required shutdown margin, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

The specified control rod assembly drop time is consistent with safety analyses that have been performed.

An inoperable control rod assembly imposes additional demands on the operators. The permissible number of inoperable control rod assemblies is limited to one in order to limit the magnitude of the operating burden, but such a failure would not prevent dropping of the OPERABLE control rod assemblies upon reactor trip.

4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of applicable reactivity anomalies within the reactor.

Specification

A. Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be compared monthly with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made. The provisions of Specification 4.0.4 are not applicable.

B. During periods of POWER OPERATION at greater than 10% of RATED POWER, the hot channel factors identified in Section 3.12 shall be determined during each effective full power month of operation using data from limited core maps. If these factors exceed their limits, an evaluation as to the cause of the anomaly shall be made. The provisions of Specification 4.0.4 are not applicable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 265 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AND

AMENDMENT NO. 264 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated June 9, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081630307), Virginia Electric and Power Company (the licensee) submitted a request for changes to the Surry Power Station, Unit Nos. 1 and 2 (Surry 1 and 2), Technical Specifications (TSs).

The proposed changes would revise action statements in TS 3.12 for insertion limit and shutdown margin requirements, revise the applicability for the operability of the rod position indication and bank demand position indication systems, revise/add action statements for rod position indication, and add action statements for group step demand counters.

2.0 REGULATORY EVALUATION

Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR) specifies the content of theTSs. TSs are required to include items in the following five specific categories: (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 regulation does not specify the particular requirements to be included in a plant's TSs.

On July 22, 1993 (58 FR 39132), the Commission published a "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (Final Policy Statement) which discussed the criteria to determine which items are required to be included in the TSs as LCOs. The criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36 (60 FR 36953).

Section 50.36(c)(2)(ii) requires that a TS LCO limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

Enclosure

(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.

TSs may be changed to reflect modifications to the plant design basis (since TSs are derived from the design basis), and/or to apply policy or guidance with regard to the required content or preferred format of TSs. In determining the acceptability of such changes, the NRC staff interprets the requirements of 10 CFR 50.36, and uses, as guidance, the applicable Standard Technical Specifications (STSs). For this review, the NRC staff used NUREG-1431, Revision 3, "Standard Technical Specifications, Westinghouse Plants."

The following general design criteria (GDC) also pertain to the NRC staff's review of Surry's application:

(1) GDC 10, insofar as it requires that the reactor coolant system (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations, including anticipated operational occurrences (AOOs);

(2) GDC-11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity;

(3) GDC-12, insofar as it requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed;

(4) GDC-13, insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges;

(5) GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to automatically initiate operation of systems and components important to safety under accident conditions;

(6) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems;

(7) GDC 26, insofar as it requires that a reactivity control system be provided and be capable of reliably controlling the rate of reactivity changes to ensure that, under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

(8) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and

(9) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary (RCPB) greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.

3.0 TECHNICAL EVALUATION

The following specific changes to the Surry Units 1 and 2 TSs are proposed:

(1) TS 3.12.A Control Bank Insertion Limits

Revision of TS 3.12.A.1 requirements and action statements for the shutdown banks:

TS 3.12.A.1 currently requires that the shutdown control rod assemblies be fully withdrawn whenever the reactor is critical. The applicant proposes to change TS 3.12.A.1 to specify that shutdown banks be within the insertion limits provided in the CORE OPERATING LIMITS REPORT (COLR), and to adopt the action statements of NUREG-1431, Section 3.1.5.

The Surry Units 1 and 2 reactors normally operate with shutdown control rod assemblies fully withdrawn, and it is expected that the insertion limits in the COLR would specify full withdrawal. Whatever the insertion limits, the shutdown margin requirement would still apply, and this will provide the equivalent protection.

Revision of TS 3.12.A.2 requirements and action statements for the control banks:

TS 3.12.A.2 currently requires that the control rod assemblies not be inserted any deeper than specified by the control rod insertion limits in the COLR. The applicant proposes to change TS 3.12.A.2 to specify that control banks continue to be within the insertion limits provided in the COLR, and to adopt the action statements of NUREG-1431, Section 3.1.6.

The proposed change adopts the action statements of NUREG-1431, Section 3.1.6. These action statements are consistent with the action statements for shutdown banks that are not within the specified insertion limits (NUREG-1431, Section 3.1.5). Satisfaction of the shutdown margin requirement, as specified in the action statement, will provide the equivalent protection.

(2) TS 3.12.B Power Distribution Limits

Revision of TS 3.12.B.6.c to delete the ± associated with a reference to 10% QUADRANT POWER TILT:

QUADRANT POWER TILT is expressed as an absolute value. A negative value would have no meaning, and could become a source of questions and confusion. The proposed change could be construed as a safety improvement, since it clarifies the TS requirements.

(3) TS 3.12.C Control Rod Assemblies

Revision of TS 3.12.C.3.b.1(b) to replace the reference to Figure 3.12-1A and Figure 3.12-1 B with a reference to the CORE OPERATING LIMITS REPORT.

This is an administrative change with no safety significance.

(4) TS 3.12.E Rod Position Indication System

Revision of the title of TS 3.12.E from "Rod Position Indication System" to "Rod Position Indication System and Bank Demand Position Indication System."

This is an administrative change with no safety significance.

Revision of the applicability of TS 3.12.E.1 for rod position indication and group step demand counter operability to be "from movement of control banks to achieve criticality and with the REACTOR CRITICAL," and revision of corresponding sections of TS 3.12.C.

The proposed change, which refers to rod motion that causes the reactor to become critical and to operation when the reactor is critical, is consistent with NUREG-1431, Section 3.1.7, which applies to Modes 1 and 2 (power operation and hot standby).

Revision of existing TS 3.12.E.2.a, TS 3.12.E.2.b, and TS 3.12.E.3 action Statements, and addition of actions in TS 3.12.E.4, and TS 3.12.E.6, for rod position indication inoperability. Additions to TS 3.12.E.5, and TS 3.12.E.6, to include action statements for group step demand counter inoperability.

Rod position indication does not have a direct input to the automatic reactor protection system logic or reactor trip setpoints. However, rod position indication is important because it is used by the operators to verify that rod positions remain in compliance with applicable TS requirements, such as control rod and shutdown bank insertion limits. This assures that the core power shape remains within the body of analyzed power shapes, and that the licensing basis accident analyses, which are performed to demonstrate that SAFDLs are not exceeded during normal operations or AOOs. Initial conditions for these accident analyses are based upon the assumption that the plant is operating within TS limits (i.e., the plant is not in an action statement when the postulated accident or AOO happens).

The proposed changes would provide protection that is equivalent to the protection currently provided. The proposed changes in action statements also serve to make the Surry 1 and 2 TSs consistent with the action statements in NUREG-1431, Section 3.1.7.

(5) TS 3.12.G Shutdown Margin.

Addition of TS 3.12.G to include shutdown margin requirements and action statement.

The proposed addition requires that, when the reactor is subcritical, the COLR shutdown margin requirements must be satisfied. If not, then boration must be begun within 15 minutes, in order to regain the required shutdown margin. The added requirements are consistent with Section 3.1.1 of NUREG 1431.

(6) TS 3.12 Basis

Revision to the Basis for TS 3.12 to reflect the revisions to TS 3.12.E. The TS 3.12 Basis change is included for the NRC's information.

(7) TS 4.10 Reactivity Anomalies

Revision to TS 4.10.A to delete the requirement to report the evaluation of the specified boron concentration discrepancy to the Nuclear Regulatory Commission per Section 6.6 of the TSs.

The proposed change deletes the phrase "and reported to the Nuclear Regulatory Commission per Section 6.6 of the Specifications" in TS 4.10.A. NUREG-1431, Section 5.6, "Reporting Requirements," which is based upon 10 CFR 50.4, does not require the reporting of any discrepancy that may exist between measured and predicted boron concentrations.

The proposed change deletes an action that is not required.

As addition background information, to the reasons stated above, the NRC staff finds that the requested changes to Surry, Units 1 and 2 TSs conform to the NRC staff precedents for LCO 3.1, "Reactivity Control Systems," Standard Technical Specifications (STS) – Westinghouse plants, NUREG-1431, Revision 3 dated June 2004.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified 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 (73 FR 43957). 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) 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 Contributors: Samuel Miranda, SRXB
Ravinder Grove, IT

Date: June 25, 2009.

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

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 265 to Renewed Facility Operating License No. DPR-32 and Amendment No. 264 to Renewed Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TSs) in response to your application dated June 9, 2008.

These amendments revise action statements in TS 3.12 for insertion limit and shutdown margin requirements, revise the applicability for the operability of the rod position indication and bank demand position indication systems, revise/add action statements for rod position indication, and add action statements for group step demand counters.

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 RMartin for)

John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosures:
1. Amendment No. 265 to DPR-32
2. Amendment No. 264 to DPR-37
3. Safety Evaluation

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