Dominion Resources Services, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard, 2SE, Glen Allen, VA 23060

Dominion®

August 24, 2011

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	11-486
NL&OS/GDM:	R0
Docket No.	50-281
License No.	DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
SURRY POWER STATION UNIT 2
CORE OPERATING LIMITS REPORT
SURRY 2 CYCLE 24 PATTERN UPG REVISION 1

Pursuant to Surry Technical Specification (TS) 6.2.C, enclosed is a copy of Dominion's Core Operating Limits Report (COLR) for Surry Unit 2 Cycle 24 Pattern UPG, Revision 1. The revision incorporates changes associated with the recently approved Surry Units 1 and 2 Technical Specifications (TS) Amendments 275/275 that relocate the Shutdown Margin (SDM) value in TS 3.12 and its basis from the TS to the COLR. Also, several editorial changes were made to improve readability.

If you have any questions or require additional information, please contact Mr. Gary Miller at (804) 273-2771.

Sincerely,

C. L. Funderburk, Director
Nuclear Licensing and Operations Support
Dominion Resources Services, Inc. for
Virginia Electric and Power Company

Enclosure

Commitment Summary: There are no new commitments as a result of this letter.

cc: U.S. Nuclear Regulatory Commission
Region II
Marquis One Tower
245 Peachtree Center Avenue, NE
Suite 1200
Atlanta, Georgia 30303-1257

K. R. Cotton
NRC Project Manager
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 08 G-9A
11555 Rockville Pike
Rockville, Maryland 20852-2738

R. E. Martin
NRC Project Manager
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 08 G-9A
11555 Rockville Pike
Rockville, Maryland 20852-2738

NRC Senior Resident Inspector
Surry Power Station

ENCLOSURE

CORE OPERATING LIMITS REPORT
Surry 2 Cycle 24 Pattern UPG
Revision 1

August 2011

1.0 INTRODUCTION

This Core Operating Limits Report (COLR) for Surry Unit 2 Cycle 24 has been prepared in accordance with the requirements of Surry Technical Specification 6.2.C.

The Technical Specifications affected by this report are:

TS 2.1 – Safety Limit, Reactor Core
TS 2.3.A.2.d – Overtemperature ΔT
TS 2.3.A.2.e – Overpower ΔT
TS 3.1.E - Moderator Temperature Coefficient
TS 3.12.A.1, TS 3.12.A.2, TS 3.12.A.3 and TS 3.12.C.3.b.1(b) - Control Bank Insertion Limits
TS 3.12.A.1.a, TS 3.12.A.2.a, TS 3.12.A.3.c and TS 3.12.G – Shutdown Margin
TS 3.12.B.1 and TS 3.12.B.2 - Power Distribution Limits
TS 3.12.F – DNB Parameters
TS Table 4.1-2A – Minimum Frequency for Equipment Tests: Item 22 – RCS Flow

2.0 REFERENCES

1. VEP-FRD-42, Rev. 2.1-A, "Reload Nuclear Design Methodology," August 2003.

   Methodology for:
   TS 3.1.E - Moderator Temperature Coefficient
   TS 3.12.A.1, TS 3.12.A.2, TS 3.12.A.3 and TS 3.12.C.3.b.1(b) - Control Bank Insertion Limit
   TS 3.12.A.1.a, TS 3.12.A.2.a, TS 3.12.A.3.c and TS 3.12.G – Shutdown Margin
   TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor and Nuclear Enthalpy Rise Hot Channel Factor
   TS 3.12.F – DNB Parameters
   TS Table 4.1-2A – Minimum Frequency for Equipment Tests: Item 22 – RCS Flow

2. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Westinghouse Proprietary), January 2005.

   Methodology for:
   TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor

3. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," (Westinghouse Proprietary), August 1985.

   Methodology for:
   TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor

4. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," (Westinghouse Proprietary), August 1985.

   Methodology for:
   TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor

5. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Report," (Westinghouse Proprietary), April 1995.

   Methodology for:
   TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor

6. WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO," (Westinghouse Proprietary), July 2006.

   Methodology for:
   TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor

7. VEP-NE-2-A, Rev. 0, "Statistical DNBR Evaluation Methodology," June 1987.

   Methodology for:
   TS 3.12.B.1 and TS 3.12.B.2 - Nuclear Enthalpy Rise Hot Channel Factor

8. VEP-NE-3-A, Rev. 0, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code," July 1990.

   Methodology for:
   TS 3.12.B.1 and TS 3.12.B.2 - Nuclear Enthalpy Rise Hot Channel Factor

9. DOM-NAF-2-A, Rev. 0.2-P-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code,"including Appendix B, "Qualification of the Westinghouse WRB-1 CHF Correlation in the Dominion VIPRE-D Computer Code," August 2010.

   Methodology for:
   TS 3.12.B.1 and TS 3.12.B.2 - Nuclear Enthalpy Rise Hot Channel Factor

10. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function," September 1986.

    Methodology for:
    TS 2.3.A.2.d – Overtemperature ΔT
    TS 2.3.A.2.e – Overpower ΔT

3.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 6.2.C.

3.1 Safety Limit, Reactor Core (TS 2.1)

The Reactor Core Safety Limits are presented in **Figure A-1**.

3.2 Overtemperature ΔT (TS 2.3.A.2.d)

$$\Delta T \leq \Delta T_0 \left[K_1 - K_2 \left(\frac{1 + t_1 s}{1 + t_2 s}\right)(T - T') + K_3(P - P') - f(\Delta I)\right]$$

Where:

ΔT is measured RCS ΔT, °F.

ΔT_0 is the indicated ΔT at RATED POWER, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature (T_{avg}), °F.

T' is the nominal T_{avg} at RATED POWER, ≤ **573.0°F**.

P is the measured pressurizer pressure, psig.

P' is the nominal RCS operating pressure ≥ **2235 psig**.

$K_1 \leq$ **1.1425** $K_2 \geq$ **0.01059** /°F $K_3 \geq$ **0.000765** /psig

$t_1 \geq$ **29.7** seconds $t_2 \leq$ **4.4** seconds

$f(\Delta I) \geq$ **0.0268** $\{-24 - (q_t - q_b)\}$, when $(q_t - q_b) <$ **-24.0%** RATED POWER

0, when **-24.0%** RATED POWER $\leq (q_t - q_b) \leq$ **8.0%** RATED POWER

0.0188 $\{(q_t - q_b) -$ **8.0**$\}$, when $(q_t - q_b) >$ **+8.0%** RATED POWER

Where q_t and q_b are percent RATED POWER in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RATED POWER.

3.3 <u>Overpower ΔT</u> (TS 2.3.A.2.e)

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5\left(\frac{t_3 s}{1 + t_3 s}\right)T - K_6(T - T') - f(\Delta I)\right]$$

Where:

ΔT is measured RCS ΔT, °F.

ΔT_0 is the indicated ΔT at RATED POWER, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature (T_{avg}), °F.

T' is the nominal T_{avg} at RATED POWER, ≤ **573.0°F**.

$K_4 \leq$ **1.0965**

$K_5 \geq$ **0.0198** /°F for increasing T_{avg}
≥ **0** /°F for decreasing T_{avg}

$K_6 \geq$ **0.001074** /°F for T > T'
≥ **0** for T ≤ T'

$t_3 \geq$ **9.0** seconds

f(ΔI) = as defined above for OTΔT

3.4 <u>Moderator Temperature Coefficient</u> (TS 3.1.E)

The Moderator Temperature Coefficient (MTC) limits are:
+6.0 pcm/°F at less than 50 percent of RATED POWER, and
+6.0 pcm/°F at 50 percent of RATED POWER and linearly decreasing to 0 pcm/°F at RATED POWER

3.5 <u>Control Bank Insertion Limits</u> (TS 3.12.A.1, TS 3.12.A.2, TS 3.12.A.3 and TS 3.12.C.3.b.1(b))

3.5.1 The control rod banks shall be limited in physical insertion as shown in **Figure A-2**.

3.5.2 The rod insertion limit for the A and B control banks is the fully withdrawn position as shown on **Figure A-2**.

3.5.3 The rod insertion limit for the A and B shutdown banks is the fully withdrawn position as shown on **Figure A-2**.

3.6 <u>Shutdown Margin</u> (TS 3.12.A.1.a, TS 3.12.A.2.a, TS 3.12.A.3.c and TS 3.12.G)

Whenever the reactor is subcritical the shutdown margin (SDM) shall be ≥ **1.77** %Δk/k.

3.7 <u>Power Distribution Limits</u> (TS 3.12.B.1 and TS 3.12.B.2)

3.7.1 Heat Flux Hot Channel Factor - FQ(z)

$$FQ(z) \leq \frac{CFQ}{P} K(z)\ for\ P > 0.5$$

$$FQ(z) \leq \frac{CFQ}{0.5} K(z)\ for\ P \leq 0.5$$

$$where: P = \frac{THERMAL\ POWER}{RATED\ POWER}$$

CFQ = 2.5

K(*z*) = 1.0 for all core heights, z

3.7.2 Nuclear Enthalpy Rise Hot Channel Factor - FΔH(N)

$$F\Delta H(N) \leq CFDH * \{1 + PFDH(1 - P)\}$$

$$where: P = \frac{THERMAL\ POWER}{RATED\ POWER}$$

CFDH = 1.56

PFDH = 0.3

3.8 <u>DNB Parameters</u> (TS 3.12.F and TS Table 4.1-2A)

Departure from Nucleate Boiling (DNB) Parameters shall be maintained within their limits during POWER OPERATION:

- Reactor Coolant System $T_{avg} \leq$ **577.0 °F**
- Pressurizer Pressure ≥ **2205 psig**
- Reactor Coolant System Total Flow ≥ 273,000 gpm (Tech Spec Limit) and ≥ **276,000 gpm** (COLR Limit)

Figure A-1

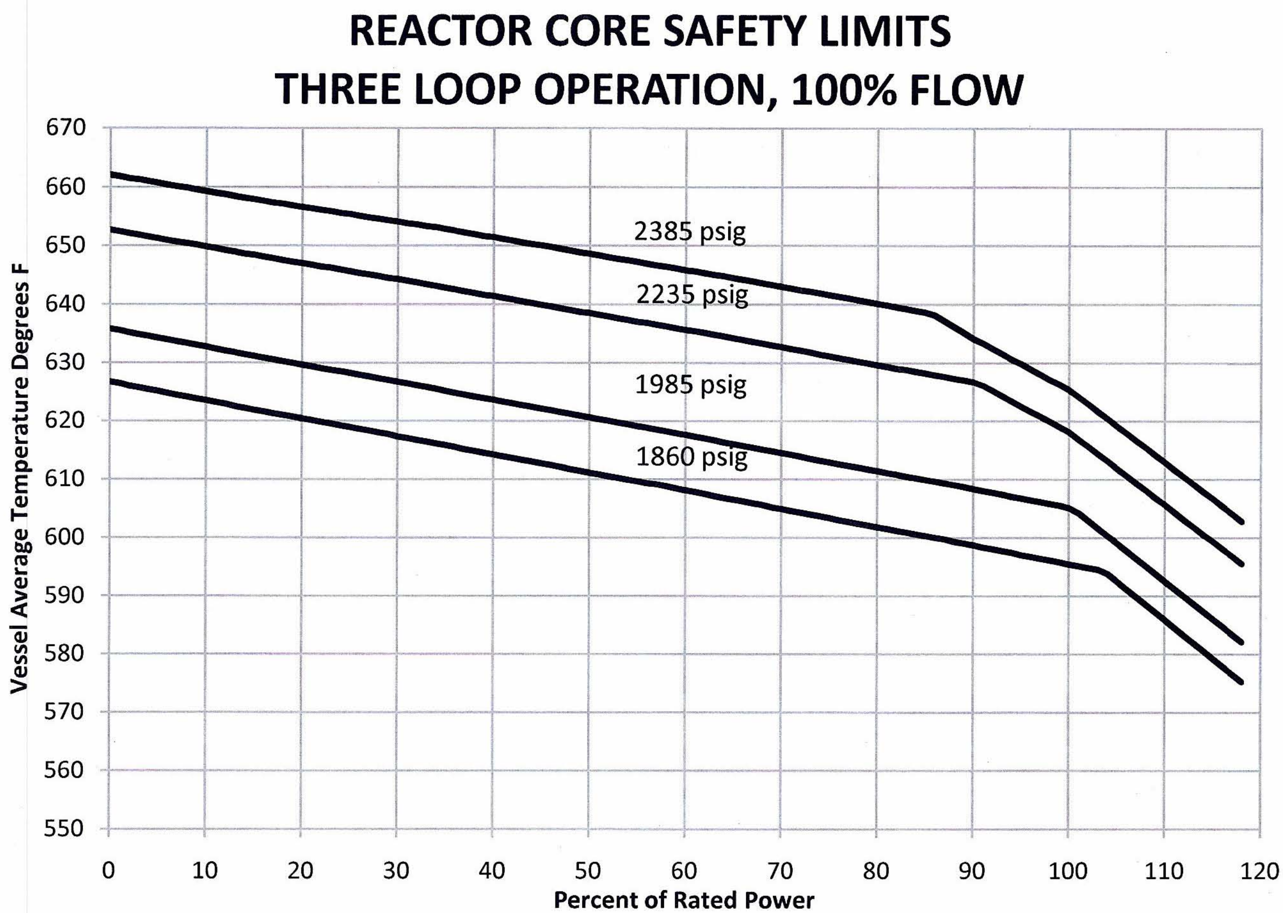
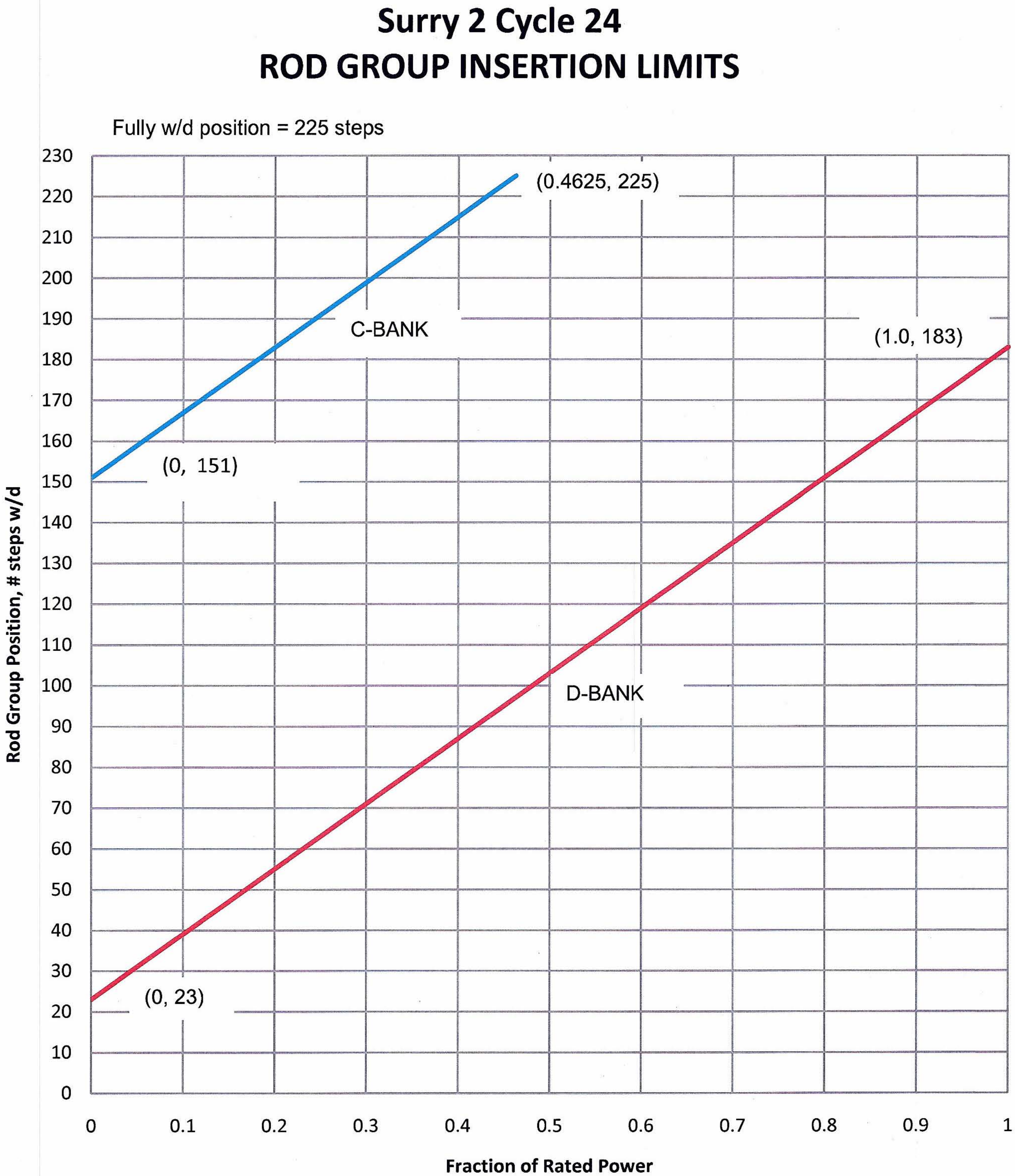

REACTOR CORE SAFETY LIMITS
THREE LOOP OPERATION, 100% FLOW
2385 psig
2235 psig
1985 psig
1860 psig
Vessel Average Temperature Degrees F
670
660
650
640
630
620
610
600
590
580
570
560
550
0
10
20
30
40
50
60
70
80
90
100
110
120
Percent of Rated Power

Figure A-2


Surry 2 Cycle 24
ROD GROUP INSERTION LIMITS
Fully w/d position = 225 steps
(0.4625, 225)
C-BANK
(1.0, 183)
(0, 151)
D-BANK
(0, 23)
Rod Group Position, # steps w/d
Fraction of Rated Power
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
150
160
170
180
190
200
210
220
230
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1