


May 25, 2010

Serial No: 10-325
NLOS/GDM R1
Docket Nos.: 50-280, 281
License Nos.: DPR-32, 37

By letter dated October 16, 2009 (Serial No. 09-581), Dominion requested amendments, in the form of changes to the Technical Specifications (TS), to Facility Operating License Numbers DPR-32 and DPR-37 for Surry Power Station Units 1 and 2, respectively. The proposed changes included the relocation of specific core operating limits (e.g., reactor core safety limits, Overtemperature ΔT and Overpower ΔT setpoints, and Departure from Nucleate Boiling (DNB) parameter limits) currently contained in the TS to the Core Operating Limits Report (COLR), and added two COLR references.

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

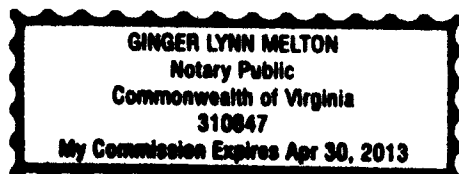

J. Alan Price
Vice President – Nuclear Engineering

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. Alan Price, who is Vice President – Nuclear Engineering of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 25th day of May, 2010.

My Commission Expires:



Winger L. Allgood
Notary Public
I was commissioned a notary public
as Winger L. Allgood.

Commitments made in this letter: None

Attachment:

- Response to NRC Request for Additional Information

cc: U.S. Nuclear Regulatory Commission
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ATTACHMENT

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

**VIRGINIA ELECTRIC AND POWER COMPANY
(DOMINION)
SURRY POWER STATION UNITS 1 AND 2**

Response to NRC Request for Additional Information
Surry Power Station Units 1 and 2

By letter dated October 16, 2009 (Serial No. 09-581), Dominion requested amendments, in the form of changes to the Technical Specifications (TS), to Facility Operating License Numbers DPR-32 and DPR-37 for Surry Power Station Units 1 and 2, respectively. The proposed changes included the relocation of specific core operating limits (e.g., reactor core safety limits, Overtemperature ΔT and Overpower ΔT setpoints, and Departure from Nucleate Boiling (DNB) parameter limits) currently contained in the TS to the Core Operating Limits Report (COLR) and added two COLR references.

Subsequent to this submittal, the NRC provided a request for additional information in an e-mail dated May 7, 2010. Dominion's response to the NRC request is provided below.

NRC Question No. 1

Review the methodology references listed on revised Technical Specification Page 6.2-2, and determine the currency of all references listed. Provide the results of this review. Should any references be determined to be inapplicable to current or recent fuel cycles, please revise the TS page to remove them and renumber the reference list.

Dominion Response

Dominion has reviewed the list of methodology references contained on Technical Specification Page 6.2-2 and verified all references to be current. The current references contained in Technical Specification 6.2.C are listed below:

1. VEP-FRD-42-A, "Reload Nuclear Design Methodology"
- 2a. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Westinghouse Proprietary)
- 2b. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," (W Proprietary)
- 2c. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," (W Proprietary)
- 2d. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Report," (Westinghouse Proprietary)
- 3a. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology"
- 3b. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code"

TS 6.2.C Reference 1, VEP-FRD-42-A, constitutes the reload methodology as discussed in Section 3.3.3.2 of Revision 41 of the Surry Updated Final Safety Analysis Report (UFSAR) [Reference 1]. TS 6.2.C Reference 2a, WCAP-16009-P-A, constitutes the methodology for the large break Loss of Coolant Accident (LOCA) as discussed in Section 14.5.1 of the Surry UFSAR. TS 6.2.C Reference 2b, WCAP-10054-P-A, and TS 6.2.C Reference 2c, WCAP-10079-P-A, document the methodologies that are used to analyze the small break LOCA as discussed in Section 14.5.2 of the Surry UFSAR. Sections 3.1, 3.5.2.1, 3.5.2.6, 14.1 and 14.3.3.2 of the Surry UFSAR reference TS 6.2.C Reference 2d, WCAP-12610-P-A, for the design bases and functional requirements for the fuel assembly components. TS 6.2.C Reference 3a, VEP-NE-2-A, constitutes Dominion's Statistical DNBR Evaluation Methodology as discussed in Sections 3.2.3.3 and 3.4.3.2 of the Surry UFSAR. The Statistical DNBR Evaluation Methodology is employed on a transient specific basis as indicated in the transient analyses in Chapter 14 of the Surry UFSAR. TS 6.2.C Reference 3b, VEP-NE-3-A, describes the qualification of the WRB-1 Critical Heat Flux (CHF) correlation in the COBRA code for the prediction of the Departure from Nucleate Boiling (DNB) analysis. The application of VEP-NE-3-A is discussed in Section 3.2.3.3 of the Surry UFSAR. This reference will continue to be used to perform the DNB analysis for the current Surry fuel product even after DOM-NAF-2-A [Reference 2] is added to the list of NRC approved methods for determining core operating limits. Therefore, the references listed under TS 6.2.C are current and applicable to current fuel cycles.

NRC Question No. 2

The NRC staff reviewed the SER approving the North Anna Improved Technical Specifications conversion and was unable to locate a sufficient technical basis for extending the completion time for a precision reactor coolant system flow measurement beyond the 24 hours contained in the Standard Technical Specifications. Please provide a technical basis for this deviation between NUREG-1431/TSTF Traveler 339-A, and the Surry proposed TS.

Dominion Response

The current requirement in the Surry TS (TS 3.12.F.1.b) is to determine that the Reactor Coolant System (RCS) total flow rate is within its limit by measurement at least once per refueling cycle. TS 3.12.F.1.b does not specify a time frame for completing this surveillance. Current plant practice is to perform this surveillance as soon as practical after the reactor reaches an equilibrium steady state condition (i.e., no xenon and no delta flux oscillations) at or near 100% RATED POWER after a refueling outage.

The request to deviate from the NUREG-1431 [Reference 3] requirement of 24 hours to allow a completion time of 7 days for performing the precision RCS flow rate measurement is based on the time required to escalate from 90% to 100% power (~10 hours), reach equilibrium conditions (~3 days), and perform and document this

measurement at the station (~5 hours). Performing the precision RCS flow rate measurement at higher power levels results in a more accurate measurement of the RCS flow rate due to the reduction in the uncertainty associated with the RCS flow rate at higher power levels. Allowing the reactor to reach an equilibrium condition results in a reduction of the uncertainty associated with the RCS flow measurement by reducing variations in the hot leg temperatures due to xenon oscillations in the core. Therefore, the 7 day period after reaching 90% of RATED POWER is reasonable to establish stable operating conditions, install the test equipment, perform the test and analyze the results.

The safety of the plant is not adversely affected by deviating from NUREG-1413, since the current Surry TS does not presently specify a time frame in which to complete the precision RCS flow rate measurement. Therefore, the proposed allowed completion time of 7 days is more restrictive than the current TS. Furthermore, the proposed 7 day completion time does not affect the safety of the plant, because RCS flow only shows slight variations from cycle to cycle. Any significant variations in RCS flow rate from one cycle to the next require a mechanistic cause, such as changes to steam generator tube plugging. These mechanistic changes will be known prior to startup. Also, other indications of RCS flow are available (e.g., elbow taps) prior to the precision RCS flow rate measurement.

Finally, the allowed time of 7 days to complete the precision RCS flow rate measurement has been approved for other plants [References 4, 5, 6, 7, 8 and 9].

Therefore, Dominion has concluded that the deviation from the 24 hours contained in the Standard Technical Specifications to 7 days is reasonable to perform the precision RCS flow rate measurement without adversely affecting the safety of the station.

NRC Question No. 3

The license amendment request (Attachment 1, Page 5) states, "DNB analyses for the Westinghouse 15x15 Upgrade product will use the NRC-approved VIPRE-D code and the W-3 or WRB-1 correlation, DOM-NAF-2-A, depending on the transient." The clause, "DOM-NAF-2-A," appears to be dangling, and it is not clear what it means in the context of this sentence. Please clarify.

Dominion Response

The sentence can be restated in the following manner for clarity: "Consistent with Appendix B of DOM-NAF-2-A, DNB analyses will use the NRC-approved VIPRE-D code and the W-3 or WRB-1 correlation depending on the transient, for the Westinghouse 15x15 Upgrade product. As stated in DOM-NAF-2-A, Appendix B, Section B.2, Dominion intends to use the VIPRE-D/W-3 code/correlation pair when the WRB-1 correlation is outside of its range of validity."

NRC Question No. 4

Does this license amendment request represent a first-time implementation of DOM-NAF-2-A at Surry? Please explain whether this method has been used previously without a requirement to appear as a COLR reference.

Dominion Response

Yes, this request represents a first-time implementation of DOM-NAF-2-A [Reference 2] at Surry. DOM-NAF-2-A has not been used to perform any licensing calculation for Surry. As part of the Safety Evaluation Report (SER) for DOM-NAF-2-A [Reference 2], Dominion is required to submit a site-specific application of DOM-NAF-2-A to the NRC for review and approval prior to performing any licensing calculations for the site. Upon approval of the inclusion of DOM-NAF-2-A to the list of NRC-approved methods for determining core operating limits, DOM-NAF-2-A will be used for performing the DNBR evaluation for Westinghouse 15x15 Upgrade fuel.

References

1. Surry Power Station Updated Final Safety Analysis Report, Revision 41, September 2009.
2. Fleet Report, DOM-NAF-2, Rev. 0.1-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," July 2009. [ADAMS Accession No. ML092190894]
3. NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Volume 1, Rev. 3.0, dated June 2004.
4. Braidwood Units 1 and 2 Technical Specifications. [ADAMS Accession No. ML053040362 and ML053040366]
5. Bryon Units 1 and 2 Technical Specifications. [ADAMS Accession No. ML052910365 and ML052910368]
6. Joseph M. Farley Units 1 and 2 Technical Specifications. [ADAMS Accession No. ML052780033 and ML052780039]
7. Vogtle, Units 1 and 2 Technical Specifications. [ADAMS Accession No. ML052840233 and ML052840236]
8. Wolf Creek Unit 1 Technical Specifications. [ADAMS Accession No. ML052720315]
9. Beaver Valley Unit 1 and 2 Technical Specifications. [ADAMS Accession No. ML052720291 and ML052720308]