ATTACHMENT

Supplement 1

VEP-FRD-42, Rev. 1-A

Reload Nuclear Design Methodology

November 1993

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Topical Report VEP-FRD-42, Revision 1-A, "Reload Nuclear Design Methodology," presents the methodology used by Virginia Electric and Power Company to perform a nuclear reload design analysis and safety evaluation. Since NRC approval of this methodology in 1986, several modifications have been made to the methodology outlined in this report. Additionally, enhancements have been incorporated into individual evaluation techniques and computer codes used for the reload safety evaluation. This supplement identifies the most significant changes and reflects the current reload methodology utilized by Virginia Electric and Power Company.

1. Topical Report VEP-FRD-42, Revision 1-A, "Reload Nuclear Design Methodology" (Ref. 1), indicates that Virginia Electric and Power Company uses the W-3 Critical Heat Flux (CHF) correlation in the COBRA Code to calculate the Departure from Nucleate Boiling Ratio (DNBR). By letter dated January 29, 1987, Topical Report VEP-NE-3, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code" (Ref. 2), was submitted for NRC review. The purpose of this submittal was to qualify the WRB-1 correlation to replace the older W-3 correlation for use in the COBRA DNBR analysis. The improved accuracy of the WRB-1 correlation results in a substantial gain in DNB margin over the use of the W-3 correlation.

   NRC acceptance of VEP-NE-3-A for application at the Surry and North Anna Power Stations and the use of the WRB-1 correlation for DNBR analysis was received by letter (Serial No. 89-571) dated July 25, 1989 (Ref. 3).

2. Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology" (Ref. 4), provides a description of our methodology for statistically treating several of the important uncertainties in DNBR analysis. Previously, these uncertainties were treated in a conservative deterministic fashion, with each parameter assumed to be simultaneously and continuously at the worst point in its uncertainty range with respect to the DNBR. The statistical methodology uses a statistical combination of some of these uncertainties maintaining the same uncertainty on each parameter, but permitting a more realistic combination of the independent variable errors and thus providing a more realistic evaluation of DNBR margin.

   NRC acceptance of the VEP-NE-2-A statistical evaluation methodology for application at the Surry and North Anna Power Stations was received by letter dated May 28, 1987 (Ref. 5).

3. Topical Report VEP-FRD-42, Revision 1-A, "Reload Nuclear Design Methodology" (Ref. 1), indicates that current limits for some key safety parameters are given in the Technical Specifications. By letter (Serial No. 90-030) dated March 29, 1990 (Ref. 6), and supplemented by letters dated May 8 (Serial No. 90-030A), August 8 (Serial No. 90-030B), and October 30, 1990 (Serial No. 90-030C), Technical Specification changes were submitted for North Anna Units 1 and 2 which would replace the cycle specific parameter limits with a reference to a Core Operating Limits Report (COLR).

   By letter (Serial No. 91-341) dated June 7, 1991 (Ref. 7), Amendment Nos. 146 and 130 to Facility Operating Licenses Nos. NPF-4 and NPF-7 for North Anna Power Station, Units 1 and 2 were issued which modified the Technical Specifications to allow reference to a COLR for the limits on the following cycle specific parameters:

   a. The moderator temperature coefficient (MTC) limits for Technical Specification 3.1.1.4 and Surveillance Requirement 4.1.1.4.

   b. The shutdown bank insertion limits for Technical Specification 3.1.3.5 and Surveillance Requirement 4.1.3.5.

   c. The control bank insertion limits for Technical Specification 3.1.3.6.

   d. The axial flux difference limits for Technical Specification 3.2.1 and Surveillance Requirement 4.2.1.

   e. The heat flux hot channel factor Fq limit at rated thermal power, the normalized Fq limit as a function of core height K(z), and the height dependent power factor N(z) for Technical Specification 3.2.2 and Surveillance Requirement 4.2.2.

   f. The nuclear enthalpy rise hot channel factor limit at rated thermal power and the power factor multiplier for Technical Specification 3.2.3 and Surveillance Requirement 4.2.3.

4. Topical Report VEP-FRD-42, Revision 1-A, "Reload Nuclear Design Methodology" (Ref. 1), presents a conservative dropped rod analysis methodology previously used by Virginia Electric and Power Company for North Anna and Surry.

   In 1986, a core uprate program was implemented for North Anna which required reanalysis of most of the UFSAR Chapter 15 accidents (Ref. 8) including the dropped rod event. Westinghouse performed these reanalyses, using the methodology of WCAP-10297-P-A, "Dropped Rod Methodology for Negative Flux Rate Trip Plants" (Ref. 9), to evaluate the dropped rod event.

Westinghouse has subsequently developed the methodology described in WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event" (Ref. 10), which was funded by the Westinghouse Owners' Group (WOG). This methodology, which is an extension of the methodology of WCAP-10297-P-A, takes no credit for any direct trip due to dropped rod(s) (current North Anna protection) or for automatic power reduction due to dropped rod(s) (current Surry protection).

By letter dated October 23, 1989 (Ref. 12), the NRC approved the methodology described in WCAP-11394-P-A for evaluation of the dropped rod event. In 1990, Virginia Electric and Power Company acquired the transient database and methodology information necessary to perform the dropped rod analyses of either WCAP-10297-P-A or WCAP-11394-P-A from Westinghouse. Subsequently, Virginia Electric and Power Company has performed evaluations which demonstrate the applicability of the methodology, the correlations, and the transient database for analysis of the dropped rod event for the North Anna and Surry Power Stations. The application of this methodology for the evaluation of the dropped rod(s) event has been implemented for both the North Anna and Surry Power Stations pursuant to the provisions of 10CFR50.59. This methodology is currently being used to provide assurance that DNBR limits are met on a reload basis for the dropped rod(s) event. The plant/cycle specific DNB limit lines required by this methodology are being generated by Virginia Electric and Power Company using our licensed thermal hydraulic models consistent with the Westinghouse methodology described in WCAP-11394-P-A.

5. Topical Report VEP-FRD-42, Revision 1-A, "Reload Nuclear Design Methodology" (Ref. 1), references and briefly describes two core physics analytical models which use the PDQ07 code.

A PDQ07 model has subsequently been developed in both 2-D and 3-D versions to upgrade and supplement those described in Reference 1. This model, designated the PDQ Two Zone model, has been validated by a process equivalent in scope and rigor to that used to validate existing PDQ07 models. Based on comparisons to core measurements, currently approved Nuclear Reliability Factors (Ref. 15) have been shown to be appropriate for Two Zone model calculations. The PDQ Two Zone model has been implemented for North Anna and Surry core physics calculations via the provisions of 10CFR50.59. This model is an equivalent replacement of existing models for calculations described in Reference 1.

References

1. Virginia Power Topical Report VEP-FRD-42 Revision 1-A, "Reload Nuclear Design Methodology," September 1986.

2. Virginia Power Topical Report VEP-NE-3-A, "Qualification of the WRB-1 CHF [Critical Heat Flux] Correlation in the Virginia Power COBRA Code," July 1990.

3. Letter (Serial No. 89-571) from G. S. Lainas (NRC) to W. R. Cartwright, entitled "Surry Units 1 and 2, and North Anna Units 1 and 2 - Use of Virginia Power Topical Report VEP-NE-3, 'Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code' (TAC Nos. 67363, 67364, 71071, and 71072)," dated July 25, 1989.

4. Virginia Power Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987.

5. Letter (Serial No. 87-335) from L. B. Engle (NRC) to W. L. Stewart, entitled "Statistical DNBR Evaluation Methodology, VEP-NE-2, Surry Power Station, Units No. 1 & No. 2 (Surry-1&2) and North Anna Power, Units No. 1 & No. 2 (NA-1&2)," dated May 28, 1987.

6. Letter (Serial No. 90-030) from W. L. Stewart to USNRC, entitled "North Anna Power Station Units 1 and 2, Implementation of Generic Letter 88-16," dated March 29, 1990, and supplemented by subsequent letters on May 8, 1990 (Serial No. 90-030A), August 8, 1990 (Serial No. 90-030B), and October 30, 1990 (Serial No. 90-030C).

7. Letter (Serial No. 91-341) from L. B. Engle (NRC) to W. L. Stewart, entitled "North Anna Units 1 and 2 - Issuance of Amendments RE: Core Operating Limits Report (TAC Nos. 76828 and 76829)," dated June 7, 1991. [Amendment Nos. 146 and 130 to Facility Operating License Nos. NPF-4 and NPF-7, respectively.]

8. Letter (Serial No. 85-077) from W. L. Stewart to H. R. Denton (NRC), entitled "Amendment to Operating Licenses NPF-4 and NPF-7, North Anna Power Station Units 1 and 2, Proposed Technical Specification Changes," dated May 2, 1985.

9. WCAP-10297-P-A, "Dropped Rod Methodology for Negative Flux Rate Trip Plants," June 1983.

10. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," January 1990.

11. WCAP-12282, "Implementation Guidelines for WCAP-11394 (Methodology for the Analysis of the Dropped Rod Event)," June 1989.

12. Letter from A. C. Thadani (NRC) to R. A. Newton (WOG), "Acceptance for Referencing of Licensing Topical Reports WCAP-11394(P) and WCAP-11395(NP), Methodology For The Analysis Of The Dropped Rod Event," October 23, 1989.

13. Virginia Power Topical Report VEP-NAF-1, "The PDQ Two Zone Model," July 1990.

14. Virginia Power Topical Report VEP-NAF-1, Supplement 1, "The PDQ Two Zone Model," November 1992.

15. Virginia Power Topical Report VEP-FRD-45A, "VEPCO Nuclear Design Reliability Factors," October 1982.