2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded.

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to the 95/95 DNBR criterion for the DNB correlations and methodologies specified in Section 5.6.5.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup, for Westinghouse fuel and < 5173°F, decreasing by 65°F per 10,000 MWD/MTU of burnup, for Framatome fuel.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

4.0 DESIGN FEATURES

4.1 Site Location

The North Anna Power Station is located in the north-central portion of Virginia in Louisa County and is approximately 40 miles north-northwest of Richmond, 36 miles east of Charlottesville; 22 miles southwest of Fredericksburg; and 70 miles southwest of Washington, D.C. The site is on a peninsula on the southern shore of Lake Anna at the end of State Route 700.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 157 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO or M5 fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core locations.

4.2.2 Control Rod Assemblies

The reactor core shall contain 48 control rod assemblies. The control material shall be silver indium cadmium, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.6 weight percent;

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. (continued)

3. Moderator Temperature Coefficient,
4. Shutdown Bank Insertion Limits,
5. Control Bank Insertion Limits,
6. AXIAL FLUX DIFFERENCE limits,
7. Heat Flux Hot Channel Factor,
8. Nuclear Enthalpy Rise Hot Channel Factor,
9. Power Factor Multiplier,
10. Reactor Trip System Instrumentation - OTΔT and OPΔT Trip Parameters,
11. RCS Pressure, Temperature, and Flow DNB Limits, and
12. Boron Concentration.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. VEP-FRD-42-A, "Reload Nuclear Design Methodology."
2. WCAP-9220-P-A, "WESTINGHOUSE ECCS EVALUATION MODEL—1981 VERSION."
3. WCAP-9561-P-A, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS—SPECIAL REPORT: THIMBLE MODELING IN W ECCS EVALUATION MODEL."
4. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."
5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

b. (continued)

6. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code."
7. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY-REFERENCE CORE REPORT."
8. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology."
9. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code."
10. VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."
11. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function."
12. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report."
13. BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel."
14. BAW-10199P-A, "The BWU Critical Heat Flux Correlations."
15. BAW-10170P-A, "Statistical Core Design for Mixing Vane Cores."
16. EMF-2103 (P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors."
17. EMF-96-029 (P)(A), "Reactor Analysis System for PWRs."
18. BAW-10168P-A, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Volume II only (SBLOCA models).

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Steam Generator Tube Inspection Report

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Nuclear Regulatory Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported to the NRC by March 1 of each year for the previous calender year. This report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Identification of tubes plugged.

c. Results of steam generator tube inspections that fall into Category C-3 require prompt notification of the Commission pursuant to Section 50.72 to 10 CFR Part 50. A Licensee Event Report shall be submitted pursuant to Section 50.73 to 10 CFR Part 50 and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.