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5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Deleted

5.6.2 Annual Radiological Environmental Operating Report

------------------------------NOTE------------------------------
A single submittal may be made for both units. The submittal should combine sections common to both units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the ODCM, and in 10 CFR Part 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6 Reporting Requirements

5.6.3 Radioactive Effluent Release Report

-----NOTE-----
A single submittal may be made for both units. The submittal should combine sections common to both units at the station.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a, as modified by approved exemptions. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the units. The material provided shall be consistent with the objectives outlined in the ODCM, Process Control Program, and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Deleted

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

3.1.1	SHUTDOWN MARGIN
3.1.3	Moderator Temperature Coefficient
3.1.4	CEA Alignment
3.1.6	Regulating Control Element Assembly Insertion Limit
3.2.1	Linear Heat Rate
3.2.2	Total Planar Radial Peaking Factor
3.2.3	Total Integrated Radial Peaking Factor
3.2.5	AXIAL SHAPE INDEX
3.3.1	RPS Instrumentation - Operating
3.9.1	Boron Concentration

5.6 Reporting Requirements

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. CENPD-199-P, "C-E Setpoint Methodology: C-E Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems"

2. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 1: C-E Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for Calvert Cliffs Units I and II"

3. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Calvert Cliffs Units 1 and 2"

4. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 3: C-E Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for Calvert Cliffs Units 1 and 2"

5. CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2"

6. Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated June 24, 1982, Unit 1 Cycle 6 License Approval (Amendment No. 71 to DPR-53 and SER)

7. CEN-348(B)-P, "Extended Statistical Combination of Uncertainties"

8. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated October 21, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-348(B)-P, Extended Statistical Combination of Uncertainties"

5.6 Reporting Requirements

9. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core"

10. CENPD-162-P-A, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution"

11. CENPD-207-P-A, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 2, Non-Uniform Axial Power Distribution"

12. CENPD-206-P-A, "TORC Code, Verification and Simplified Modeling Methods"

13. CENPD-225-P-A, "Fuel and Poison Rod Bowing"

14. CENPD-266-P-A, "The ROCS and DIT Computer Code for Nuclear Design"

15. CENPD-275-P-A, "C-E Methodology for Core Designs Containing Gadolinia - Urania Burnable Absorbers"

16. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers"

17. CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report"

18. CEN-161-(B)-P-A, "Improvements to Fuel Evaluation Model"

19. CEN-161-(B)-P, Supplement 1-P, "Improvements to Fuel Evaluation Model"

20. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated February 4, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-161-(B)-P, Supplement 1-P, Improvements to Fuel Evaluation Model"

21. CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure"

5.6 Reporting Requirements

22. Letter from Mr. A. E. Scherer (CE) to Mr. J. R. Miller (NRC), dated December 15, 1981, LD-81-095, Enclosure 1-P, "C-E ECCS Evaluation Model Flow Blockage Analysis"

23. CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS"

24. CENPD-133, Supplement 5, "CEFLASH-4A, a FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis"

25. CENPD-134, Supplement 2, "COMPERC-II, a Program for Emergency Refill-Reflood of the Core"

26. Letter from Mr. D. M. Crutchfield (NRC) to Mr. A. E. Scherer (CE), dated July 31, 1986, "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Licensing Topical Reports"

27. CENPD-135, Supplement 5-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program"

28. Letter from Mr. R. L. Baer (NRC) to Mr. A. E. Scherer (CE), dated September 6, 1978, "Evaluation of Topical Report CENPD-135, Supplement 5"

29. CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model"

30. CENPD-133, Supplement 3-P, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident"

31. Letter from Mr. K. Kniel (NRC) to Mr. A. E. Scherer (CE), dated September 27, 1977, "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P"

5.6 Reporting Requirements

32. CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup"

33. Letter from Mr. C. Aniel (NRC) to Mr. A. E. Scherer, dated April 10, 1978, "Evaluation of Topical Report CENPD-138, Supplement 2-P"

34. Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. J. R. Miller (NRC) dated February 22, 1985, "Calvert Cliffs Nuclear Power Plant Unit 1; Docket No. 50-317, Amendment to Operating License DPR-53, Eighth Cycle License Application"

35. Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated May 20, 1985, "Safety Evaluation Report Approving Unit 1 Cycle 8 License Application"

36. Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. R. A. Clark (NRC), dated September 22, 1980, "Amendment to Operating License No. 50-317, Fifth Cycle License Application"

37. Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated December 12, 1980, "Safety Evaluation Report Approving Unit 1, Cycle 5 License Application"

38. Letter from Mr. J. A. Tiernan (BG&E) to Mr. A. C. Thadani (NRC), dated October 1, 1986, "Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2, Docket Nos. 50-317 & 50-318, Request for Amendment"

39. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated July 7, 1987, Docket Nos. 50-317 and 50-318, Approval of Amendments 127 (Unit 1) and 109 (Unit 2)

40. CENPD-188-A, "HERMITE: A Multi-Dimensional Space-Time Kinetics Code for PWR Transients"

5.6 Reporting Requirements

41. The power distribution monitoring system referenced in various specifications and the BASES, is described in the following documents:

    i. CENPD-153-P, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered, Fixed Incore Detector System"

    ii. CEN-119(B)-P, "BASSS, Use of the Incore Detector System to Monitor the DNB-LCO on Calvert Cliffs Unit 1 and Unit 2"

    iii. Letter from Mr. G. C. Creel (BG&E) to NRC Document Control Desk, dated February 7, 1989, "Calvert Cliffs Nuclear Power Plant Unit No. 2; Docket 50-318, Request for Amendment, Unit 2 Ninth Cycle License Application"

    iv. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. G. C. Creel (BG&E), dated January 10, 1990, "Safety Evaluation Report Approving Unit 2 Cycle 9 License Application"

42. Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. R. E. Denton (BGE), dated May 11, 1995, "Approval to Use Convolution Technique in Main Steam Line Break Analysis - Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M90897 and M90898)

43. CENPD-387-P-A, "ABB Critical Heat Flux Correlations for PWR Fuel"

44. CENPD-199-P, Supplement 2-P-A, Appendix A, "CE Setpoint Methodology"

45. CENPD-404-P-A, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs"

5.6 Reporting Requirements

46. CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model"

47. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model"

48. WCAP-11596-P-A, "Qualification of the PHOENIX-P, ANC Nuclear Design System for Pressurized Water Reactor Cores"

49. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code"

50. WCAP-10965-P-A Addendum 1, "ANC: A Westinghouse Advanced Nodal Computer Code; Enhancements to ANC Rod Power Recovery"

51. WCAP-16072-P-A, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs"

52. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON"

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, 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 mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Not Used

5.6 Reporting Requirements

5.6.7 Post-Accident Monitoring Report

When a report is required by Condition B or G of LCO 3.3.10, "Post Accident Monitoring Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.9 Steam Generator Tube Inspection Report

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 15 days.

b. The complete results of the steam generator tube inservice inspection during the report period shall be submitted to the NRC prior to March 1 of each 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; and

   3. Identification of tubes plugged or repaired.

5.6 Reporting Requirements

c. Results of steam generator tube inspections which fall into Category C-3 require verbal notification of the NRC Regional Administrator by telephone within 24 hours prior to resumption of plant operation. The written follow-up of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence and shall be submitted within the next 30 days.