5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

5.6.1 Not Used.

5.6.2 Annual Radiological Environmental Monitoring Report

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

The Annual Radiological Environmental Monitoring Report covering the operation of the plant 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 Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Monitoring Report shall include summarized and tabulated results, in the format of Regulatory Guide 4.8, December 1975, of all radiological environmental samples taken during the report period. 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.

The report shall also include the following: a summary description of the radiological environmental monitoring program; a map of sampling locations keyed to a table giving distances and directions from the reactor site; and the results of licensees participation in the Interlaboratory Comparison Program defined in the ODCM.

5.6 Reporting Requirements (continued)

5.6.3 Radioactive Effluent Report

---NOTE---
A single submittal may be made for the plant. The submittal shall combine sections common to both units.

The Radioactive Effluent Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Not Used.

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:

TS 2.1.1, "Reactor Core SLs";
LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
LCO 3.1.3, "Isothermal Temperature Coefficient (ITC)";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits";
LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2";

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation" Overtemperature ΔT and Overpower ΔT Parameter Values for Table 3.3.1-1;
LCO 3.4.1, "RCS Pressure, Temperature, and Flow - Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1, "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. NSPNAD-8101-PA, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version);

2. NSPNAD-8102-PA, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units" (latest approved version);

3. NSPNAD-97002-PA, "Northern States Power Company's "Steam Line Break Methodology", (latest approved version);

4. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology";

5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code";

6. Deleted;

7. WCAP-10924-P-A, "Westinghouse Large Break LOCA Best Estimate Methodology";

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. XN-NF-77-57 (A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II";

9. WCAP-13677, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support $ZIRLO_{TM}$ Cladding Options";

10. NSPNAD-93003-A, "Transient Power Distribution Methodology", (latest approved version);

11. NAD-PI-003, "Prairie Island Nuclear Power Plant Required Shutdown Margin During Physics Tests";

12. NAD-PI-004, "Prairie Island Nuclear Power Plant $F_Q^W(Z)$ Penalty With Increasing $\left[F_Q^C(Z) / K(Z)\right]$ Trend";

13. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification";

14. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions";

15. WCAP-11397-P-A, "Revised Thermal Design Procedure";

16. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report";

17. WCAP-7588 Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods";

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

18. WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod";

19. WCAP-7907-P-A, "LOFTRAN Code Description";

20. WCAP-7979-P-A, "TWINKLE – A Multidimensional Neutron Kinetics Computer Code";

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

22. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event";

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

24. WCAP-12910 Rev. 1-A, "Pressurizer Safety Valve Set Pressure Shift";

25. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis"; and

26. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses".

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 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat-up, cooldown, low temperature operation, criticality, and hydrostatic testing, OPPS arming, PORV lift settings and Safety Injection Pump Disable Temperature as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.10, "Pressurizer Safety Valves";
LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) > Safety Injection (SI) Pump Disable Temperature";
LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) ≤ Safety Injection (SI) Pump Disable Temperature"; and
LCO 3.5.3, "ECCS - Shutdown".

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

b. The analytical methods used to determine the RCS pressure and temperature limits and Cold Overpressure Mitigation System setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (includes any exemption granted by NRC to ASME Code Case N-514).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties shall be submitted to the NRC prior to issuance of an updated PTLR.

5.6.7 Steam Generator Tube Inspection Report

1. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.

2. The results of steam generator tube in-service inspections shall be included with the summary reports of ASME Code Section XI inspections submitted within 90 days of the end of each refueling outage. Results of steam generator tube in-service inspections not associated with a refueling outage shall be submitted within 90 days of the completion of the inspection. These reports 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 sleeved.

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

3. Results of steam generator tube inspections which fall into Category C-3 require notification to the Commission prior to resumption of plant operation, and reporting as a special report to the Commission within 30 days. This special report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

4. The results of inspections performed under Specification 5.5.8.b for all tubes that have defects below the F* or EF* distance, and were not plugged, shall be reported to the Commission within 15 days following the inspection. The report shall include:

   a. Identification of F* and EF* tubes, and

   b. Location and extent of degradation.

5. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

   a. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

   b. If circumferential crack-like indications are detected at the tube support plate intersections.

   c. If indications are identified that extend beyond the confines of the tube support plate.

   d. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

e. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1E-02, notify the NRC and provide an assessment of the safety significance of the occurrence.

5.6.8 EM Report

When a report is required by Condition C or I of LCO 3.3.3, "Event Monitoring (EM) 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.