(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NMC to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NMC to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NMC to transfer byproduct materials from other job sites owned by Northern States Power Company for the purpose of volume reduction and decontamination.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

NMC is authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 176, are hereby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NMC shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," Revision 0, submitted by letter dated October 18, 2004.

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NMC to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NMC to transfer byproduct materials from other job sites owned by Northern States Power Company for the purposes of volume reduction and decontamination.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

NMC is authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 166, are hereby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NMC shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," Revision 0, submitted by letter dated October 18, 2004.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met, except for SRs with a specified Frequency of 24 months, if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

The specified Frequency is met for each SR with a specified Frequency of 24 months if the Surveillance is performed within 24 months, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension (1.25 times the interval specified) does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the interval extension (1.25 times the interval specified) applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

3.0 SR APPLICABILITY (continued)

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

SR 3.0.4

Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM. Determination of projected dose contributions for radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days from the liquid effluent releases would exceed 0.12 mrem to the total body or 0.4 mrem to any organ; or from the gaseous effluent releases would exceed 0.4 mrad for gamma air dose, 0.8 mrad for beta air dose, or 0.6 mrem organ dose;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

  1. for noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

  2. for iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

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 $\left(F^N_{\Delta H}\right)$";
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-A, "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-P-A, "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".