L-PI-03-042
10 CFR 50.90

June 16, 2003

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKETS 50-282 AND 50-306
LICENSE Nos. DPR-42 AND DPR-60

SUPPLEMENT TO LICENSE AMENDMENT REQUEST (LAR) DATED
MARCH 25, 2003, SAFETY ANALYSES TRANSITION

By letter dated March 25, 2003, the Nuclear Management Company, LLC (NMC) submitted an LAR titled, "Safety Analyses Transition" which proposes Technical Specification (TS) changes associated with transition to Westinghouse performance of safety analyses for the Prairie Island Nuclear Generating Plant (PINGP). This letter supplements the subject LAR. NMC submits this supplement in accordance with the provisions of 10 CFR 50.90.

NMC has identified additional TS and Bases pages associated with the subject LAR that require revision. The Safety Analyses Transition LAR, as submitted on March 25, 2003, proposes to set the axial flux difference function value to zero for the overpower delta-T reactor trip function, TS Table 3.3.1-1, Function 7, and thus delete reference to Surveillance Requirements (SRs) 3.3.1.3 and 3.3.1.6. Changes to TS page 3.1.8-1 and Bases pages B 3.1.8-6 and B 3.3.1-21 are required to support the proposed change to the overpower delta-T reactor trip function and associated deletion of SRs. Additional Bases pages, B 3.1.1-2, B 3.2.2-2 and B 3.2.4-1, require changes to support the 200 calories per gram fuel energy deposition limit, during an ejected rod accident, proposed in the subject LAR.

NMC has determined that a more recent version of WCAP-10924 than referenced in TS 5.6.5.b Item 7 is used in PINGP loss of coolant accident analyses. In lieu of updating this reference, NMC proposes to implement the guidance of NUREG-1431, Revision 2, "Standard Technical Specifications, Westinghouse Plants" for Core Operating Limits Report (COLR) references. With respect to COLR references, NUREG-1431 states,

> Identify the Topical Report(s) by number and title or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date. The

A001

COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

In accordance with the guidance of NUREG-1431, NMC proposes to make changes to TS pages 5.0-35 and 5.0-36, in addition to those proposed in the March 25, 2003 submittal, to delete topical report dates, references to NRC safety evaluations, volume and addendum in TS 5.6.5.b Items 4, 5, 7, 8 and 9. The COLR will contain the complete identification for each of these topical reports as required by NUREG-1431. When references to date, volume and addendum are removed, TS 5.6.5.b Items 6 and 7 would be identical; therefore NMC proposes to replace TS 5.6.5.b Item 6 with "Deleted". An additional page, 5.0-34, was revised to include reference to TS 2.1.1 which was removed from page 5.0-35. Other changes shown on these pages were previously proposed in the March 25, 2003 submittal of the subject LAR.

Exhibit A for the original submittal is not affected by the changes proposed in this supplement and is not included. Exhibit B presents the original submittal Exhibit B cover page marked up to show the seven additional pages submitted with this supplement and the marked up TS and Bases pages to show the proposed changes. Exhibit C presents the original submittal Exhibit C cover page marked up to show the seven additional pages submitted with this supplement and the revised TS and Bases pages incorporating the proposed changes.

This supplement does not change the schedule requirements presented in the March 25, 2003 letter for this LAR. The proposed changes in this supplement do not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Assessment presented in the original March 25, 2003 submittal. The original submittal discussed deletion of SRs 3.3.1.3 and 3.3.1.6, 200 calories per gram limit and changes to TS 5.6.5.b COLR references based on the guidance of NUREG-1431.

This letter contains one new commitment, "The COLR will contain the complete identification for each of these topical reports as required by NUREG-1431." This letter contains no revisions to existing commitments.

In accordance with 10 CFR 50.91, NMC is notifying the State of Minnesota of this LAR supplement by transmitting a copy of this letter and attachments to the designated State Official.

Please address any comments or questions regarding this LAR supplement to Mr. Dale Vincent at 1-651-388-1121.

I declare under penalty of perjury that the foregoing is true and accurate. Executed on June 16, 2003.

Joseph M. Solymossy
Site Vice-President, Prairie Island Nuclear Generating Plant

CC Regional Administrator, USNRC, Region III
Project Manager, Prairie Island Nuclear Generating Plant, USNRC, NRR
NRC Resident Inspector – Prairie Island Nuclear Generating Plant
Glenn Wilson, State of Minnesota

Exhibits:
B. Marked Up Pages
C. Revised Pages

EXHIBIT B

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Letter L-PI-03-042, Supplement to
License Amendment Request dated March 25, 2003

<u>Marked Up Pages</u>
(shaded material to be added, strikethrough material to be removed)

Technical Specification Pages

2.0-1	3.3.1-18
2.0-2	3.3.1-23
3.1.8-1	3.3.1-24
3.2.1-2	5.0-34
3.2.1-3	5.0-35
3.2.3-1	5.0-36
3.2.3-2	5.0-37
3.2.3-3	5.0-38
3.2.3-4	5.0-39
Insert A 3.2.3-1	5.0-40

Bases Pages

B 2.1.1-2	B 3.2.1-14
B 2.1.1-3	B 3.2.2-2
B 2.1.1-4	B 3.2.3-1
B 2.1.1-5	B 3.2.3-2
B 2.1.1-6	B 3.2.3-3
B 3.1.1-2	B 3.2.3-4
B 3.1.8-6	B 3.2.3-5
B 3.2.1-2	B 3.2.3-6
B 3.2.1-3	B 3.2.3-7
B 3.2.1-4	B 3.2.3-8
B 3.2.1-5	B 3.2.3-9
B 3.2.1-6	B 3.2.3-10
B 3.2.1-7	B 3.2.3-11
B 3.2.1-8	B 3.2.4-1
B 3.2.1-9	B 3.3.1-20
B 3.2.1-10	B 3.3.1-21
B 3.2.1-11	B 3.3.1-56
B 3.2.1-12	B 3.3.1-58
B 3.2.1-13	

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions - MODE 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Isothermal Temperature Coefficient (ITC)";
LCO 3.1.4, "Rod Group Alignment Limits";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, ~~7,~~ and 16.e may be reduced to "3" required channels, provided:

a. RCS lowest loop average temperature is ≥ 535°F;

b. SDM is within the limits provided in the COLR; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes
	AND	
	A.2 Suspend PHYSICS TESTS exceptions.	1 hour

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 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

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 $\left(F^N_{\Delta H}\right)$”;
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-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”~~, July, 1985~~;

5. WCAP-10054-P-A, “Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code”~~, August, 1985~~;

6. Deleted~~WCAP-10924-P-A, “Westinghouse Large Break LOCA Best-Estimate Methodology”, December, 1988~~;

7. WCAP-10924-P-A, ~~Volume 1, Addendum 4,~~ “Westinghouse Large Break LOCA Best Estimate Methodology”~~, August, 1990~~;

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"~~, May, 1981~~;

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"~~, April 1993 (approved by NRC SE dated November 26, 1993)~~;

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,"~~(approved by NRC SE dated July 30, 2002)~~; ~~and~~

12. NAD-PI-004, "Prairie Island Nuclear Power Plant $F_Q^W(Z)$ Penalty With Increasing $[F_Q^C(Z)/K(Z)]$ Trend,"~~approved by NRC SE dated July 30, 2002).~~;

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"; and

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

BASES

BACKGROUND (continued)

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. The primary safety analyses that rely on the SDM limits are the boron dilution and main steam line break (MSLB) analyses.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 200~~280~~ cal/gm energy deposition for the rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements, at end of cycle (EOC), is based on a MSLB, as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently an RCS cooldown. This results in a reduction of the reactor coolant

BASES

LCO (continued)	The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, ~~7,~~ and 16.e, may be reduced to "3" required channels during the performance of PHYSICS TESTS provided: a. RCS lowest loop average temperature is ≥ 535°F; b. SDM is within the limits provided in the COLR; and c. THERMAL POWER is ≤ 5% RTP.
APPLICABILITY	This LCO is applicable when performing low power PHYSICS TESTS. The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.
ACTIONS	<u>A.1 and A.2</u> If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification within 1 hour.

BASES

BACKGROUND (continued)

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling ratio (DNBR) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency (referred to as Condition II events). The departure from nucleate boiling (DNB) design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to a value greater than the criterion listed in Reference 1. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Controlling $F^N_{\Delta H}$ precludes core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition during Condition II transients (Ref. 1);

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 200~~280~~ cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During transient conditions arising from events of moderate frequency (Condition II events), there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 200~~280~~ cal/gm (Ref. 1); ~~and~~

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Allowable Value. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE. Note that the Overpower ΔT trip Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions. ~~While performing PHYSICS TESTS in accordance with LCO 3.1.8, the number of required channels may be reduced to three.~~

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure-High and-Low trips and the Overtemperature ΔT trip.

a. Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

EXHIBIT C

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Letter L-PI-03-042, Supplement to License Amendment Request dated March 25, 2003

<u>Revised Pages</u>

Technical Specification Pages

2.0-1	5.0-34
3.1.8-1	5.0-35
3.2.1-2	5.0-36
3.2.1-3	5.0-37
3.2.3-1	5.0-38
3.3.1-18	5.0-39
3.3.1-23	5.0-40
3.3.1-24	

Bases Pages

B 2.1.1-2	B 3.2.1-12
B 2.1.1-3	B 3.2.1-13
B 2.1.1-4	B 3.2.1-14
B 2.1.1-5	B 3.2.1-15
B 3.1.1-2	B 3.2.1-16
B 3.1.8-6	B 3.2.2-2
B 3.2.1-2	B 3.2.3-1
B 3.2.1-3	B 3.2.3-2
B 3.2.1-4	B 3.2.3-3
B 3.2.1-5	B 3.2.3-4
B 3.2.1-6	B 3.2.4-1
B 3.2.1-7	B 3.3.1-20
B 3.2.1-8	B 3.3.1-21
B 3.2.1-9	B 3.3.1-56
B 3.2.1-10	B 3.3.1-58
B 3.2.1-11	

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions - MODE 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of

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LCO 3.1.4, "Rod Group Alignment Limits";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, and 16.e may be reduced to "3" required channels, provided:

a. RCS lowest loop average temperature is ≥ 535°F;

b. SDM is within the limits provided in the COLR; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit. AND A.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour

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.

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   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
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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:

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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";

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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"; and

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

BASES

BACKGROUND (continued)

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. The primary safety analyses that rely on the SDM limits are the boron dilution and main steam line break (MSLB) analyses.

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a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and $\leq$ 200 cal/gm energy deposition for the rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements, at end of cycle (EOC), is based on a MSLB, as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently an RCS cooldown. This results in a reduction of the reactor coolant

BASES

LCO (continued)

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, and 16.e, may be reduced to "3" required channels during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 535°F;

b. SDM is within the limits provided in the COLR; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY

This LCO is applicable when performing low power PHYSICS TESTS. The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification within 1 hour.

BASES

BACKGROUND (continued)

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling ratio (DNBR) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency (referred to as Condition II events). The departure from nucleate boiling (DNB) design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to a value greater than the criterion listed in Reference 1. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Controlling $F^N_{\Delta H}$ precludes core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition during Condition II transients (Ref. 1);

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 200 cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During transient conditions arising from events of moderate frequency (Condition II events), there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 200 cal/gm (Ref. 1); and

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Allowable Value. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE. Note that the Overpower ΔT trip Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure-High and-Low trips and the Overtemperature ΔT trip.

a. Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.