Docket Nos. 50-282
and 50-306

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| DDiIanni | DHagan |
| OGC | |

Mr. T. M. Parker, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Parker:

SUBJECT: AMENDMENT NOS. 92 AND 85 TO FACILITY OPERATING LICENSES NOS. DPR-42 AND DPR-60: RELOCATION OF CYCLE SPECIFIC OPERATING PARAMETERS FROM THE TECHNICAL SPECIFICATION TO CORE REPORTING LIMITS REPORTS (TAC NOS. 75314 AND 75315)

The Commission has issued the enclosed Amendment No. 92 to Facility Operating License No. DPR-42 and Amendment No. 85 to the Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated November 17, 1989.

The amendments delete the cycle-specific parameter limits from the technical specifications and locate them in the core operating limits report. These changes are in accordance with the guidelines given in our Generic Letter (GL) 88-16. Technical Specifications 1.0, 3.1, 3.10, and 6.7 and the Basis section are affected by these amendments.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice. The issuance of these amendments completes our work effort under TAC Nos. 75314 and 75315.

Sincerely,

Dominic C. DiIanni

Dominic C. DiIanni, Project Manager
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 92 to License No. DPR-42
2. Amendment No. 85 to License No. DPR-60
3. Safety Evaluation

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See next page

| | | | |
|---|---|---|---|
| LA/PD31:DRSP | PM/PD31:DRSP | (A)D/PD31:DRSP | OGC |
| MRShuttleworth | DDiIanni | JThoma | |
| 02/28/90 | 03/1/90 | 03/7/90 | 02/26/90 |

DF01
111

Docket file



dealing with the reactor trip initiated by low steam generator water level coincidence with the steam/feedwater mismatch has been deleted from the technical specification.

Sincerely,

original signed by

Dominic C. DiIanni, Project Manager
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

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*See previous concurrence

| | | | |
|---|---|---|---|
| *LA/PD31:DRSP | PM/PD31:DRSP | (A)D/PD31:DRSP | *OGC |
| MRShuttleworth | DDiIanni | JThoma | |
| 02/22/90 | 03/7/90 | 03/7/90 | 02/28/90 |

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 13, 1990

Docket Nos. 50-282
and 50-306

Mr. T. M. Parker, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Parker:

SUBJECT: AMENDMENT NOS.92AND 85 TO FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60: RELOCATION OF CYCLE SPECIFIC OPERATING PARAMETERS FROM THE TECHNICAL SPECIFICATION TO CORE REPORTING LIMITS REPORTS (TAC NOS. 75314 AND 75315)

The Commission has issued the enclosed Amendment No. 92 to Facility Operating License No. DPR-42 and Amendment No. 85 to the Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated November 17, 1989.

The amendments delete the cycle-specific parameter limits from the technical specifications and locate them in the core operating limits report. These changes are in accordance with the guidelines given in our Generic Letter (GL) 88-16. Technical Specifications 1.0, 3.1, 3.10, and 6.7 and the Basis section are affected by these amendments.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice. The issuance of these amendments completes our work effort under TAC Nos. 75314 and 75315.

Amendment Nos. 87 and 80 issued by letter dated April 3, 1989 found the technical specification changes associated with the elimination of the reactor trip initiated by low steam generator water level coincidence with steam/feedwater mismatch acceptable. However, this acceptance was conditional upon, the staff finding an adequate verification and validation program in place during the staff audit at the vendor's site. The reactor trip was not removed from the TS because of this condition. The staff performed this verification and validation audit on July 13 and 14, 1989 at the Westinghouse site and the findings are given in the safety evaluation enclosure 1. The staff has concluded that the licensee has demonstrated an acceptable verification and validation program. On this basis, the technical specification TS 2.3.A.3(c)

dealing with the reactor trip initiated by low steam generator water level coincidence with the steam/feedwater mismatch has been deleted from the technical specification.

Sincerely,

Dominic C. DiIanni

Dominic C. DiIanni, Project Manager
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 92 to License No. DPR-42
2. Amendment No. 85 to License No. DPR-60
3. Safety Evaluation

cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 92
License No. DPR-42

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated November 17, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act, and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-42 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.92, are hereby incorporated in the license The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John O. Thoma

John O. Thoma, Acting Director
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 13, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 92

FACILITY OPERATING LICENSE NO. DPR-42

DOCKET NO. 50-282

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change

| REMOVE | INSERT |
|---|---|
| TS-ix | TS-ix |
| TS-xiii | TS-xiii |
| TS.1-2 | TS.1-2 |
| TS 2.3-3 | TS 2.3-3 |
| TS.3.1-12 | TS.3.1-12 |
| Table TS 3.5-2 | Table TS 3.5-2 |
| TS.3.10-1 | TS.3.10-1 |
| TS.3.10-2 | TS.3.10-2 |
| TS.3.10-3 | TS.3.10-3 |
| TS.3.10-4 | TS.3.10-4 |
| TS.3.10-5 | TS.3.10-5 |
| TS.3.10-7 | TS.3.10-7 |
| TS.3.10-8 | TS.3.10-8 |
| TS.6.7-4 | TS.6.7-4 |
| TS.6.7-5 | TS.6.7-5 |
| - | TS.6.7-6 |
| B.2.1-2 | B.2.1-2 |
| B.3.10-1 | B.3.10-1 |
| B.3.10-3 | B.3.10-3 |
| B.3.10-4 | B.3.10-4 |
| B.3.10-6 | B.3.10-4 |
| B.3.10-10 | B.3.10-10 |

TABLE OF CONTENTS (Continued)

# APPENDIX A TECHNICAL SPECIFICATIONS

## LIST OF FIGURES

CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY shall exist when:

1. Penetrations required to be isolated during accident conditions are either:

   a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Specifications 3.6.C and 3.6.D.

2. Blind flanges required by Table TS.4.4-1 are installed.

3. The equipment hatch is closed and sealed.

4. Each air lock is in compliance with the requirements of Specification 3.6.M.

5. The containment leakage rates are within their required limits.

COLD SHUTDOWN

A reactor is in the COLD SHUTDOWN condition when the reactor is subcritical by at least 1% $\Delta k/k$ and the reactor coolant average temperature is less than 200°F.

CORE ALTERATION

CORE ALTERATION is the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel, which may affect core reactivity. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.7.A.6. Plant operation within these operating limits is addressed in individual specifications.

2.3.A.2.g. Open reactor coolant pump motor breaker.

1. Reactor coolant pump bus undervoltage - ≥75% of normal voltage.

2. Reactor coolant pump bus underfrequency - ≥58.2 Hz

h. Power range neutron flux rate.

1. Positive rate - ≤15% of RATED THERMAL POWER with a time constant ≥2 seconds

2. Negative rate - ≤7% of RATED THERMAL POWER with a time constant ≥2 seconds

3. Other reactor trips

a. High pressurizer water level - ≤90% of narrow range instrument span.

b. Low-low steam generator water level - ≥5% of narrow range instrument span.

c. Turbine Generator trip

1. Turbine stop valve indicators - closed

2. Low auto stop oil pressure - ≥45 psig

d. Safety injection - See Specification 3.5

3.1.F. ISOTHERMAL TEMPERATURE COEFFICIENT (ITC)

1. When the reactor is critical, the isothermal temperature coefficient shall be less than 5 pcm/°F with all rods withdrawn, except during low power PHYSICS TESTS and as specified in 3.1.F.2 and 3.

2. When the reactor is above 70 percent RATED THERMAL POWER with all rods withdrawn, the isothermal temperature coefficient shall be negative, except as specified in 3.1.F.3.

3. If the limits of 3.1.F.1 or 2 cannot be met, POWER OPERATION may continue provided the following actions are taken:

   a. Establish and maintain control rod withdrawal limits sufficient to restore the ITC to less than the limits specified in Specification 3.1.F.1 and 2 above within 24 hours or be in HOT SHUTDOWN within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits specified in the CORE OPERATING LIMITS REPORT.

   b. Maintain the control rods within the withdrawal limits established above until a subsequent calculation verifies that the ITC has been restored to within its limit for the all rods withdrawn condition.

   c. Submit a special report to the Commission within 30 days, describing the value of the measured ITC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the ITC to within its limit for the all rods withdrawn condition.

TABLE TS.3.5-2 (Page 2 of 2)

INSTRUMENT OPERATING CONDITIONS FOR REACTOR TRIP

| FUNCTIONAL UNIT | 1<br>MINIMUM OPERABLE CHANNELS | 2<br>MINIMUM DEGREE OF REDUNDANCY | 3<br>PERMISSIBLE BYPASS CONDITIONS(1) | 4<br>OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET |
|---|---|---|---|---|
| 13. Undervoltage 4KV RCP Bus | 1/bus | 1/bus | | Maintain hot shutdown |
| 14. Underfrequency 4KV Bus | 1/bus | 1/bus | | Maintain hot shutdown |
| 15. Control Rod Misalignment Monitor | | | | |
| a. Rod position deviation | 1 | - | | Log data required by |
| b. Quadrant power tilt | 1 | - | | TS 3.10 I and TS 3.10 J |
| 16. RCP Breaker Open | 2 | 1 | | Maintain hot shutdown |
| 17. Safety Injection Actuation Signal | 2 | 1 | | Maintain hot shutdown |
| 18. Automatic Trip Logic including Reactor Trip Breakers** | 2 | 1 | | Notes 3, 4 |

Note 1: Automatic permissives not listed

Note 2: When bypass condition exists, maintain normal operation

Note 3: With the number of operable channels one less than the minimum operable channels requirement, be in at least hot shutdown within 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.1, provided the other channel is operable.

Note 4: When in the hot shutdown condition with the number of operable channels one less than the minimum operable channels requirement, restore the inoperable channel to operable status within 48 hours or open the reactor trip breakers within the next hour.

F.P. - Full Power

* - One additional channel may be taken out of service for low power physics testing

** - Includes both undervoltage and shunt trip circuits and if either circuit becomes inoperable the respective channel shall be considered inoperable.

## 3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

### Applicability

Applies to the limits on core fission power distribution and to the limits on control rod operations.

### Objective

To assure 1) core subcriticality after reactor trip, 2) acceptable core power distributions during power operation, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

### Specification

A. Shutdown Margin

The shutdown margin with allowance for a stuck control rod assembly shall exceed the applicable value shown in Figure TS.3.10-1 under all steady-state operating conditions, except for PHYSICS TESTS, from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at HOT SHUTDOWN conditions if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon or boron concentration.

B. Power Distribution Limits

1. At all times, except during low power PHYSICS TESTING, measured hot channel factors, $F_Q^N$ and $F_{\Delta H}^N$, as defined below and in the bases, shall meet the following limits:

$$F_Q^N \times 1.03 \times 1.05 \leq (F_Q^{RTP} / P) \times K(Z)$$

$$F_{\Delta H}^N \times 1.04 \leq F_{\Delta H}^{RTP} \times [1 + PFDH(1-P)]$$

where the following definitions apply:

- $F_Q^{RTP}$ is the $F_Q$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.
- $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.
- PFDH is the Power Factor Multiplier for $F_{\Delta H}^N$ specified in the CORE OPERATING LIMITS REPORT.
- K(Z) is a normalized function that limits $F_Q(z)$ axially as specified in the CORE OPERATING LIMITS REPORT.
- Z is the core height location.
- P is the fraction of RATED THERMAL POWER at which the core is operating. In the $F_Q^N$ limit determination when $P \leq 0.50$, set $P = 0.50$.

3.10.B.1. - $F_Q^N$ or $F_{\Delta H}^N$ is defined as the measured $F_Q$ or $F_{\Delta H}$ respectively, with the smallest margin or greatest excess of limit.

- 1.03 is the engineering hot channel factor, $F_Q^E$, applied to the measured $F_Q^N$ to account for manufacturing tolerance.

- 1.05 is applied to the measured $F_Q^N$ to account for measurement uncertainty.

- 1.04 is applied to the measured $F_{\Delta H}^N$ to account for measurement uncertainty.

2. Hot channel factors, $F_Q^N$ and $F_{\Delta H}^N$, shall be measured and the target flux difference determined, at equilibrium conditions according to the following conditions, whichever occurs first:

   (a) At least once per 31 effective full-power days in conjunction with the target flux difference determination, or

   (b) Upon reaching equilibrium conditions after exceeding the reactor power at which target flux difference was last determined, by 10% or more of RATED THERMAL POWER.

   $F_Q^N$ (equil) shall meet the following limit for the middle axial 80% of the core:

   $$F_Q^N \text{ (equil)} \times V(Z) \times 1.03 \times 1.05 \leq (F_Q^{RTP} / P) \times K(Z)$$

   where V(Z) is specified in the CORE OPERATING LIMITS REPORT and other terms are defined in 3.10.B.1 above.

3. (a) If either measured hot channel factor exceeds its limit specified in 3.10.B.1, reduce reactor power and the high neutron flux trip set-point by 1% for each percent that the measured $F_Q^N$ or by the factor specified in the CORE OPERATING LIMITS REPORT for each percent that the measured $F_{\Delta H}^N$ exceeds the 3.10.B.1 limit. Then follow 3.10.B.3(c).

   (b) If the measured $F_Q^N$ (equil) exceeds the 3.10.B.2 limits but not the 3.10.B.1 limit, take one of the following actions:

      1. Within 48 hours place the reactor in an equilibrium configuration for which Specification 3.10.B.2 is satisfied, or

      2. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured $F_Q^N$ (equil) x 1.03 x 1.05 x V(Z) exceeds the limit.

3.10.B.3. (c) If subsequent in-core mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a HOT SHUTDOWN condition with return to power authorized up to 50% of RATED THERMAL POWER for the purpose of PHYSICS TESTING. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above 50% of RATED THERMAL POWER. THERMAL POWER may then be increased provided $F_Q^N$ or $F_{\Delta H}^N$ is demonstrated through in-core mapping to be within its limits.

(d) If two successive measurements indicate an increase in the peak rod power $F_{\Delta H}^N$ with exposure, either of the following actions shall be taken:

1. $F_Q^N$ (equil) shall be multiplied by 1.02 x V(Z) x 1.03 x 1.05 for comparison to the limit specified in 3.10.B.2, or

2. $F_Q^N$ (equil) shall be measured at least once per seven effective full power days until two successive maps indicate that the peak pin power, $F_{\Delta H}^N$, is not increasing.

4. Except during PHYSICS TESTS, and except as provided by specifications 5 through 8 below, the indicated axial flux difference for at least three operable excore channels shall be maintained within the target band about the target flux difference. The target band is specified in the CORE OPERATING LIMITS REPORT.

5. <u>Above 90 percent of RATED THERMAL POWER:</u>

   If the indicated axial flux difference of two OPERABLE excore channels deviates from the target band, within 15 minutes either eliminate such deviation, or reduce THERMAL POWER to less than 90 percent of RATED THERMAL POWER.

6. <u>Between 50 and 90 percent of RATED THERMAL POWER:</u>

   a. The indicated axial flux difference may deviate from the target band for a maximum of one* hour (cumulative) in any 24-hour period provided that the difference between the indicated axial flux difference about the target flux difference does not exceed the envelope specified in the CORE OPERATING LIMITS REPORT.

   b. If 6.a is violated for two OPERABLE excore channels then the THERMAL POWER shall be reduced to less than 50% of RATED THERMAL POWER and the high neutron flux setpoint reduced to less than 55% of RATED THERMAL POWER.

*May be extended to 16 hours during incore/excore calibration.

3.10.B.6. c. A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference of at least three OPERABLE excore channels being within the target band.

7. Less than 50 percent of RATED THERMAL POWER:

a. The indicated axial flux difference may deviate from the target band.

b. A power increase to a level greater than 50 percent of RATED THERMAL POWER is contingent upon the indicated axial flux difference of at least three OPERABLE excore channels not being outside the target band for more than one hour (cumulative) out of the preceding 24 hour period.

8. In applying 6a and 7b above, penalty deviations outside the target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of power operation outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of power operation outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

9. If alarms associated with monitoring the indicated axial flux difference deviations from the target band are not operable, the indicated axial flux difference value for each OPERABLE excore channel shall be logged at least once per hour for the first 24 hours and half-hourly thereafter until the alarms are returned to an OPERABLE status. For the purpose of applying this specification, logged values of indicated axial flux difference must be assumed to apply during the previous interval between loggings.

C. QUADRANT POWER TILT RATIO

1. Except for PHYSICS TESTS, if the QUADRANT POWER TILT RATIO exceeds 1.02 but is less than 1.07, the rod position indication shall be monitored and logged once each shift to verify rod position within each bank assignment and, within two hours, one of the following steps shall be taken:

a. Correct the QUADRANT POWER TILT RATIO to less than 1.02.

b. Restrict core power level so as not to exceed RATED THERMAL POWER less 2% for every 0.01 that the QUADRANT POWER TILT RATIO exceeds 1.0.

3.10.C.2. If the QUADRANT POWER TILT RATIO exceeds 1.02 but is less than 1.07 for a sustained period of more than 24 hours, or if such a tilt recurs intermittently, the reactor shall be brought to the HOT SHUTDOWN condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

3. Except for PHYSICS TESTS if the QUADRANT POWER TILT RATIO exceeds 1.07, the reactor shall be brought to the HOT SHUTDOWN condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

4. If the core is operating above 85% power with one excore nuclear channel inoperable, then the core quadrant power balance shall be determined daily and after a 10% power change using either 2 movable detectors or 4 core thermocouples per quadrant, per Specification 3.11.

D. Rod Insertion Limits

1. The shutdown rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT when the reactor is critical or approaching criticality.

2. When the reactor is critical or approaching criticality, the control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT.

3. Insertion limits do not apply during PHYSICS TESTS or during periodic exercise of individual rods. The shutdown margin shown in Figure TS.3.10-1 must be maintained except for low power PHYSICS TESTING. For this test the reactor may be critical with all but one high worth full-length control rod inserted for a period not to exceed 2 hours per year provided a rod drop test is run on the high worth full-length rod prior to this particular low power PHYSICS TEST.

3.10.G. Inoperable Rod Limitations

1. An inoperable rod is a rod which (a) does not trip, (b) is declared inoperable under specification 3.10.E. or 3.10.H. or (c) cannot be moved by its drive mechanism and cannot be corrected within 8 hours.

2. The reactor shall be brought to the HOT SHUTDOWN condition within 6 hours should more than one inoperable rod be discovered during POWER OPERATION.

3. If the inoperable rod is located below the 200 step level and is capable of being tripped, or if the rod is located below the 30 step level whether or not it is capable of being tripped, then the insertion limits specified in the CORE OPERATING LIMITS REPORT apply.

4. If the inoperable rod cannot be located, or if the inoperable rod is located above the 30 step level and cannot be tripped, then the insertion limits specified in the CORE OPERATING LIMITS REPORT apply.

5. If POWER OPERATION is continued with one inoperable rod, the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days unless the rod is earlier made OPERABLE. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, THERMAL POWER shall be reduced to a level consistent with the safety analysis.

H. Rod Drop Time

At operating temperature and full flow, the drop time of each RCCA shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry. If the time is greater than 1.8 seconds, the rod shall be declared inoperable.

3.10.I. Monitor Inoperability Requirements

1. If the rod bank insertion limit monitor is inoperable, or if the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift, after a load change greater than 10 percent of RATED THERMAL POWER, and after 30 inches or more of rod motion.

2. If both the rod position deviation monitor and one or both of the quadrant power tilt monitors are inoperable for 2 hours or more, the nuclear overpower trip shall be reset to 93% of RATED THERMAL POWER in addition to the increased surveillance requirements.

3. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the calculated power tilt shall be logged every two hours after a load change greater than 10% of RATED THERMAL POWER

J. DNB Parameters

The following DNB related parameters limits shall be maintained during POWER OPERATION:

| | | |
|---|---|---|
| a. | Reactor Coolant System Tavg | ≤564°F |
| b. | Pressurizer Pressure | ≥2220 psia* |
| c. | Reactor Coolant Flow | ≥the value specified in the CORE OPERATING LIMITS REPORT |

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours. Compliance with c. is demonstrated by verifying that the parameter is within its limit after each refueling cycle.

*Limit not applicable during either a THERMAL POWER ramp increase in excess of (5%) RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of (10%) RATED THERMAL POWER

6.7.A.5. Annual Summaries of Meteorological Data

An annual summary of meteorological data shall be submitted for the previous calendar year in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability at the request of the Commission.

6.7.A.6. Core Operating Limits Report

a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Heat Flux Hot Channel Factor Limit ($F_Q^{RPT}$), Nuclear Enthalpy Rise Hot Channel Factor Limit ($F_{\Delta H}^{RTP}$), PFDH, K(Z) and V(Z) (Specifications 3.10.B.1, 3.10.B.2 and 3.10.B.3)

2. Axial Flux Difference Limits and Target Band (Specifications 3.10.B.4 through 3.10.B.9)

3. Shutdown and Control Bank Insertion Limits (Specification 3.10.D)

4. Reactor Coolant System Flow Limit (Specification 3.10.J)

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:

NSPNAD-8101-A, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version)

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

WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology", July, 1985

WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", August, 1985

WCAP-10924-P-A, "Westinghouse Large-Break LOCA Best-Estimate Methodology", December, 1988

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

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 shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be supplied upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

B. REPORTABLE EVENTS

The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified by a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Operations Committee and the results of this review shall be submitted to the Safety Audit Committee and the Vice President Nuclear Generation.

C. Environmental Reports

The reports listed below shall be submitted to the Administrator of the appropriate Regional NRC Office or his designate:

1. Annual Radiation Environmental Monitoring Report

(a) Annual Radiation Environmental Monitoring Reports covering the operation of the program during the previous calendar year shall be submitted prior to May 1 of each year.

(b) The Annual Radiation Environmental Monitoring Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 4.10.B.1. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

(c) The Annual Radiation Environmental Monitoring Reports 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 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 as soon as possible in a supplementary report.

(d) The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensees participation in the Interlaboratory Comparison Program, required by Specification 4.10.C.1.

2. Environmental Special Reports

(a) When radioactivity levels in samples exceed limits specified in Table 4.10-3, an Environmental Special Report shall be submitted within 30 days from the end of the affected calendar quarter. For certain cases involving long analysis time, determination of quarterly averages may extend beyond the 30 day period. In these cases the potential for exceeding the quarterly limits will be reported within the 30 day period to be followed by the Environmental Special Report as soon as practicable.

3. Other Environmental Reports (non-radiological, non-aquatic)

Written reports for the following items shall be submitted to the appropriate NRC Regional Administrator:

a. Environmental events that indicate or could result in a significant environmental impact casually related to plant operation. The following are examples: excessive bird impaction; onsite plant or animal disease outbreaks; unusual mortality of any species protected by the Endangered Species Act of 1973; or increase in nuisance organisms or conditions. This report shall be submitted within 30 days of the event and shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

b. Proposed changes, test or experiments which may result in a significant increase in any adverse environmental impact which was not previously reviewed or evaluated in the Final Environmental Statement or supplements thereto. This report shall include an evaluation of the environmental impact of the proposed activity and shall be submitted 30 days prior to implementing the proposed change, test or experiment.

D. Special Reports

Unless otherwise indicated, special reports required by the Technical Specifications shall be submitted to the appropriate NRC Regional Administrator within the time period specified for each report.

## 2.1 SAFETY LIMIT, REACTOR CORE

Bases continued

power levels of 91% and 74% respectively. For the 2235 psig and 2385 psig curves, the coolant average temperature at the core exit is equal to 650°F below power levels of 64% and 73% respectively.

The third and fourth criteria are evaluated using standard DNB methodology. For all four curves the DNBR is limiting at higher power levels. The area of safe operation is below these curves.

The plant conditions required to violate the limits in the lower power range are precluded by the self-actuated safety valves on the steam generators. The highest nominal setting of the steam generator safety valves is 1129 psig (saturation temperature 560°F). At zero power the difference between primary coolant and secondary coolant is zero and at full power it is 50°F. The reactor conditions at which steam generator safety valves open is shown as a dashed line on Figure TS.2.1-1.

Except for special tests, POWER OPERATION with only one loop or with natural circulation is not allowed. Safety limits for such special tests will be determined as a part of the test procedure.

The curves are conservative for the following nuclear hot channel factors:

$$F^N_{\Delta H} = F^{RTP}_{\Delta H}\,[1 + PFDH(1-P)]\ ;\ \text{and}\ F^N_Q = F^{RTP}_Q$$

where:

- $F^{RTP}_Q$ is the $F_Q$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.
- $F^{RTP}_{\Delta H}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.
- PFDH is the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the CORE OPERATING LIMITS REPORT.

Use of these factors results in more conservative safety limits than would result from power distribution limits in Specification TS.3.10.

This combination of hot channel factors is higher than that calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are covered by Specification 3.10. Adverse power distribution factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits specified in the CORE OPERATING LIMITS REPORT assure that the DNB ratio is always greater at part power than at full power.

The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than 1.30 for Exxon Nuclear fuel and less than 1.17 for Westinghouse fuel.

## 3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

### Bases

Throughout the 3.10 Technical Specifications, the terms "rod(s)" and "RCCA(s)" are synonymous.

A. Shutdown Margin

Trip shutdown reactivity is provided consistent with plant safety analyses assumptions. One percent shutdown margin is adequate except for the steam break analysis, which requires more shutdown reactivity due to the more negative moderator temperature coefficient at end of life (when boron concentration is low). Figure TS.3.10-1 is drawn accordingly.

B. Power Distribution Control

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operations) and II (Incidents of Moderate frequency) events by: (a) maintaining the minimum DNBR in the core of greater than or equal to 1.30 for Exxon fuel and 1.17 for Westinghouse fuel during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. The ECCS analysis was performed in accordance with SECY 83-472. One calculation at the 95% probability level was performed as well as one calculation with all the required features of 10 CFR Part 50, Appendix K. The 95% probability level calculation used the peak linear heat generation rate specified in the CORE OPERATING LIMITS REPORT. The Appendix K calculation used the peak linear heat generation rate specified in the CORE OPERATING LIMITS REPORT for the $F_Q$ limit specified in the CORE OPERATING LIMITS REPORT. Maintaining 1) peaking factors below the $F_Q$ limit specified in the CORE OPERATING LIMITS REPORT during all Condition I events and 2) the peak linear heat generation rate below the value specified in the CORE OPERATING LIMITS REPORT at the 95% probability level assures compliance with the ECCS analysis.

During operation, the plant staff compares the measured hot channel factors, $F_Q^N$ and $F_{\Delta H}^N$, (described later) to the limits determined in the transient and LOCA analyses. The terms on the right side of the equations in Section 3.10.B.1 represent the analytical limits. Those terms on the left side represent the measured hot channel factors corrected for engineering, calculational, and measurement uncertainties.

$F_Q^N$ is the measured Nuclear Hot Channel Factor, defined as the maximum local heat flux on the surface of a fuel rod divided by the average heat flux in the core. Heat fluxes are derived from measured neutron fluxes and fuel enrichment.

The K(Z) function specified in the CORE OPERATING LIMITS REPORT is a normalized function that limits $F_Q$ axially. The K(Z) value is based on large and small break LOCA analyses.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

When a measurement of $F^N_{\Delta H}$ is taken, measurement error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup PHYSICS TESTS, at least once each effective full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would otherwise affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An accidental misalignment limit of 13 steps precludes a rod misalignment greater than 15 inches with consideration of maximum instrumentation error.

2. Control rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.

3. The control bank insertion limits specified in the CORE OPERATING LIMITS REPORT are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

B. Power Distribution Control (continued)

The permitted relaxation in $F^N_{\Delta H}$ and $F^N_Q$ allows for radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factor limits are met. In specification 3.10, $F^N_Q$ is arbitrarily limited for P less than or equal to 0.5 (except for low power PHYSICS TESTS).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically control of flux difference is required to limit the difference between the current value of Flux Difference ($\Delta I$) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = $\Delta I$/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that the $F_Q$ limit is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows: At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the full length rod control rod bank more than 190 steps withdrawn (i.e., normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium was noted, no allowances for excore detector error are necessary and indicated deviation from the indicated reference value but within the target band is permitted. The allowed deviation from the target flux difference as a function of THERMAL POWER is specified in the CORE OPERATING LIMITS REPORT.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

B. Power Distribution Control (continued)

In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The consequences of being outside the target band but within the limits specified in the CORE OPERATING LIMITS REPORT for power levels between 50% and 90% has been evaluated and determined to result in acceptable peaking factors. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated axial flux difference. In all cases the target band is the Limiting Condition for Operation. Only when the target band is violated do the limits specified in the CORE OPERATING LIMITS REPORT apply.

If, for any reason, the indicated axial flux difference is not controlled within the target band for as long a period as one hour, then xenon distributions may be significantly changed and operation at or below 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Condition II events the core is protected from overpower and a minimum DNBR of 1.30 for Exxon fuel and 1.17 for Westinghouse fuel by an automatic protection system. Compliance with operating procedures is assumed as a precondition for Condition II transients, however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

C. QUADRANT POWER TILT RATIO

QUADRANT POWER TILT RATIO limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation, e.g. rod misalignment, x-y xenon transient, or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication System or core instrumentation per Specification 3.10.F, and

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

H. Rod Drop Time

The required drop time to dashpot entry is consistent with the safety analysis.

I. Monitor Inoperability Requirements

If either the rod bank insertion limit monitor or rod position deviation monitor are inoperable, additional surveillance is required to ensure adequate shutdown margin is maintained.

If the rod position deviation monitor and quadrant power tilt monitor(s) are inoperable, the overpower reactor trip setpoint is reduced (and also power) to ensure that adequate core protection is provided in the event that unsatisfactory conditions arise that could affect radial power distribution.

Increased surveillance is required, if the quadrant power tilt monitors are inoperable and a load change occurs, in order to confirm satisfactory power distribution behavior. The automatic alarm functions related to QUADRANT POWER TILT must be considered incapable of alerting the operator to unsatisfactory power distribution conditions.

J. DNB Parameters

The RCS flow rate, $T_{avg}$, and Pressurizer Pressure requirements are based on transient analyses assumptions. The flow rate shall be verified by calorimetric flow data and/or elbow taps. Elbow taps are used in the reactor coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. If a reduction in flow rate is indicated below the value specified in the CORE OPERATING LIMITS REPORT, shutdown is required to investigate adequacy of core cooling during operation.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 85
License No. DPR-60

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated November 17, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.85, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John O. Thoma

John O. Thoma, Acting Director
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 13, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 85

FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NO. 50-306

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change

| REMOVE | INSERT |
|---|---|
| TS-ix | TS-ix |
| TS-xiii | TS-xiii |
| TS.1-2 | TS.1-2 |
| TS 2.3-3 | TS 2.3-3 |
| TS.3.1-12 | TS.3.1-12 |
| Table TS 3.5-2 | Table TS 3.5-2 |
| TS.3.10-1 | TS.3.10-1 |
| TS.3.10-2 | TS.3.10-2 |
| TS.3.10-3 | TS.3.10-3 |
| TS.3.10-4 | TS.3.10-4 |
| TS.3.10-5 | TS.3.10-5 |
| TS.3.10-7 | TS.3.10-7 |
| TS.3.10-8 | TS.3.10-8 |
| TS.6.7-4 | TS.6.7-4 |
| TS.6.7-5 | TS.6.7-5 |
| - | TS.6.7-6 |
| B.2.1-2 | B.2.1-2 |
| B.3.10-1 | B.3.10-1 |
| B.3.10-3 | B.3.10-3 |
| B.3.10-4 | B.3.10-4 |
| B.3.10-6 | B.3.10-4 |
| B.3.10-10 | B.3.10-10 |

TABLE OF CONTENTS (Continued)

## APPENDIX A TECHNICAL SPECIFICATIONS

## LIST OF FIGURES

CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY shall exist when:

1. Penetrations required to be isolated during accident conditions are either:

   a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Specifications 3.6.C and 3.6.D.

2. Blind flanges required by Table TS.4.4-1 are installed.

3. The equipment hatch is closed and sealed.

4. Each air lock is in compliance with the requirements of Specification 3.6.M.

5. The containment leakage rates are within their required limits.

COLD SHUTDOWN

A reactor is in the COLD SHUTDOWN condition when the reactor is subcritical by at least 1% $\Delta k/k$ and the reactor coolant average temperature is less than 200°F.

CORE ALTERATION

CORE ALTERATION is the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel, which may affect core reactivity. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.7.A.6. Plant operation within these operating limits is addressed in individual specifications.

2.3.A.2.g. Open reactor coolant pump motor breaker.

1. Reactor coolant pump bus undervoltage - ≥75% of normal voltage.

2. Reactor coolant pump bus underfrequency - ≥58.2 Hz

h. Power range neutron flux rate.

1. Positive rate - ≤15% of RATED THERMAL POWER with a time constant ≥2 seconds

2. Negative rate - ≤7% of RATED THERMAL POWER with a time constant ≥2 seconds

3. Other reactor trips

a. High pressurizer water level - ≤90% of narrow range instrument span.

b. Low-low steam generator water level - ≥5% of narrow range instrument span.

c. Turbine Generator trip

1. Turbine stop valve indicators - closed

2. Low auto stop oil pressure - ≥45 psig

d. Safety injection - See Specification 3.5

3.1.F. ISOTHERMAL TEMPERATURE COEFFICIENT (ITC)

1. When the reactor is critical, the isothermal temperature coefficient shall be less than 5 pcm/°F with all rods withdrawn, except during low power PHYSICS TESTS and as specified in 3.1.F.2 and 3.

2. When the reactor is above 70 percent RATED THERMAL POWER with all rods withdrawn, the isothermal temperature coefficient shall be negative, except as specified in 3.1.F.3.

3. If the limits of 3.1.F.1 or 2 cannot be met, POWER OPERATION may continue provided the following actions are taken:

   a. Establish and maintain control rod withdrawal limits sufficient to restore the ITC to less than the limits specified in Specification 3.1.F.1 and 2 above within 24 hours or be in HOT SHUTDOWN within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits specified in the CORE OPERATING LIMITS REPORT.

   b. Maintain the control rods within the withdrawal limits established above until a subsequent calculation verifies that the ITC has been restored to within its limit for the all rods withdrawn condition.

   c. Submit a special report to the Commission within 30 days, describing the value of the measured ITC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the ITC to within its limit for the all rods withdrawn condition.

TABLE TS.3.5-2 (Page 2 of 2)

INSTRUMENT OPERATING CONDITIONS FOR REACTOR TRIP

| FUNCTIONAL UNIT | 1<br>MINIMUM OPERABLE CHANNELS | 2<br>MINIMUM DEGREE OF REDUNDANCY | 3<br>PERMISSIBLE BYPASS CONDITIONS(1) | 4<br>OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET |
|---|---|---|---|---|
| 13. Undervoltage 4KV RCP Bus | 1/bus | 1/bus | | Maintain hot shutdown |
| 14. Underfrequency 4KV Bus | 1/bus | 1/bus | | Maintain hot shutdown |
| 15. Control Rod Misalignment Monitor | | | | |
| a. Rod position deviation | 1 | - | | Log data required by |
| b. Quadrant power tilt | 1 | - | | TS 3.10 I and TS 3.10 J |
| 16. RCP Breaker Open | 2 | 1 | | Maintain hot shutdown |
| 17. Safety Injection Actuation Signal | 2 | 1 | | Maintain hot shutdown |
| 18. Automatic Trip Logic including Reactor Trip Breakers** | 2 | 1 | | Notes 3, 4 |

Note 1: Automatic permissives not listed

Note 2: When bypass condition exists, maintain normal operation

Note 3: With the number of operable channels one less than the minimum operable channels requirement, be in at least hot shutdown within 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.1, provided the other channel is operable.

Note 4: When in the hot shutdown condition with the number of operable channels one less than the minimum operable channels requirement, restore the inoperable channel to operable status within 48 hours or open the reactor trip breakers within the next hour.

F.P. - Full Power

* - One additional channel may be taken out of service for low power physics testing

** - Includes both undervoltage and shunt trip circuits and if either circuit becomes inoperable the respective channel shall be considered inoperable.

## 3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

### Applicability

Applies to the limits on core fission power distribution and to the limits on control rod operations.

### Objective

To assure 1) core subcriticality after reactor trip, 2) acceptable core power distributions during power operation, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

### Specification

A. Shutdown Margin

The shutdown margin with allowance for a stuck control rod assembly shall exceed the applicable value shown in Figure TS.3.10-1 under all steady-state operating conditions, except for PHYSICS TESTS, from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at HOT SHUTDOWN conditions if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon or boron concentration.

B. Power Distribution Limits

1. At all times, except during low power PHYSICS TESTING, measured hot channel factors, $F_Q^N$ and $F_{\Delta H}^N$, as defined below and in the bases, shall meet the following limits:

$$F_Q^N \times 1.03 \times 1.05 \leq (F_Q^{RTP} / P) \times K(Z)$$

$$F_{\Delta H}^N \times 1.04 \leq F_{\Delta H}^{RTP} \times [1 + PFDH(1-P)]$$

where the following definitions apply:

- $F_Q^{RTP}$ is the $F_Q$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.
- $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.
- PFDH is the Power Factor Multiplier for $F_{\Delta H}^N$ specified in the CORE OPERATING LIMITS REPORT.
- K(Z) is a normalized function that limits $F_Q(z)$ axially as specified in the CORE OPERATING LIMITS REPORT.
- Z is the core height location.
- P is the fraction of RATED THERMAL POWER at which the core is operating. In the $F_Q^N$ limit determination when $P \leq 0.50$, set $P = 0.50$.

3.10.B.1. - $F_Q^N$ or $F_{\Delta H}^N$ is defined as the measured $F_Q$ or $F_{\Delta H}$ respectively, with the smallest margin or greatest excess of limit.

- 1.03 is the engineering hot channel factor, $F_Q^E$, applied to the measured $F_Q^N$ to account for manufacturing tolerance.

- 1.05 is applied to the measured $F_Q^N$ to account for measurement uncertainty.

- 1.04 is applied to the measured $F_{\Delta H}^N$ to account for measurement uncertainty.

2. Hot channel factors, $F_Q^N$ and $F_{\Delta H}^N$, shall be measured and the target flux difference determined, at equilibrium conditions according to the following conditions, whichever occurs first:

(a) At least once per 31 effective full-power days in conjunction with the target flux difference determination, or

(b) Upon reaching equilibrium conditions after exceeding the reactor power at which target flux difference was last determined, by 10% or more of RATED THERMAL POWER.

$F_Q^N$ (equil) shall meet the following limit for the middle axial 80% of the core:

$$F_Q^N \text{ (equil)} \times V(Z) \times 1.03 \times 1.05 \leq (F_Q^{RTP} / P) \times K(Z)$$

where V(Z) is specified in the CORE OPERATING LIMITS REPORT and other terms are defined in 3.10.B.1 above.

3. (a) If either measured hot channel factor exceeds its limit specified in 3.10.B.1, reduce reactor power and the high neutron flux trip set-point by 1% for each percent that the measured $F_Q^N$ or by the factor specified in the CORE OPERATING LIMITS REPORT for each percent that the measured $F_{\Delta H}^N$ exceeds the 3.10.B.1 limit. Then follow 3.10.B.3(c).

(b) If the measured $F_Q^N$ (equil) exceeds the 3.10.B.2 limits but not the 3.10.B.1 limit, take one of the following actions:

1. Within 48 hours place the reactor in an equilibrium configuration for which Specification 3.10.B.2 is satisfied, or

2. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured $F_Q^N$ (equil) x 1.03 x 1.05 x V(Z) exceeds the limit.

3.10.B.3. (c) If subsequent in-core mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a HOT SHUTDOWN condition with return to power authorized up to 50% of RATED THERMAL POWER for the purpose of PHYSICS TESTING. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above 50% of RATED THERMAL POWER. THERMAL POWER may then be increased provided $F_Q^N$ or $F_{\Delta H}^N$ is demonstrated through in-core mapping to be within its limits.

(d) If two successive measurements indicate an increase in the peak rod power $F_{\Delta H}^N$ with exposure, either of the following actions shall be taken:

1. $F_Q^N$ (equil) shall be multiplied by 1.02 x V(Z) x 1.03 x 1.05 for comparison to the limit specified in 3.10.B.2, or

2. $F_Q^N$ (equil) shall be measured at least once per seven effective full power days until two successive maps indicate that the peak pin power, $F_{\Delta H}^N$, is not increasing.

4. Except during PHYSICS TESTS, and except as provided by specifications 5 through 8 below, the indicated axial flux difference for at least three operable excore channels shall be maintained within the target band about the target flux difference. The target band is specified in the CORE OPERATING LIMITS REPORT.

5. <u>Above 90 percent of RATED THERMAL POWER:</u>

If the indicated axial flux difference of two OPERABLE excore channels deviates from the target band, within 15 minutes either eliminate such deviation, or reduce THERMAL POWER to less than 90 percent of RATED THERMAL POWER.

6. <u>Between 50 and 90 percent of RATED THERMAL POWER:</u>

a. The indicated axial flux difference may deviate from the target band for a maximum of one* hour (cumulative) in any 24-hour period provided that the difference between the indicated axial flux difference about the target flux difference does not exceed the envelope specified in the CORE OPERATING LIMITS REPORT.

b. If 6.a is violated for two OPERABLE excore channels then the THERMAL POWER shall be reduced to less than 50% of RATED THERMAL POWER and the high neutron flux setpoint reduced to less than 55% of RATED THERMAL POWER.

*May be extended to 16 hours during incore/excore calibration.

3.10.B.6. c. A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference of at least three OPERABLE excore channels being within the target band.

7. Less than 50 percent of RATED THERMAL POWER:

a. The indicated axial flux difference may deviate from the target band.

b. A power increase to a level greater than 50 percent of RATED THERMAL POWER is contingent upon the indicated axial flux difference of at least three OPERABLE excore channels not being outside the target band for more than one hour (cumulative) out of the preceding 24 hour period.

8. In applying 6a and 7b above, penalty deviations outside the target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of power operation outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of power operation outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

9. If alarms associated with monitoring the indicated axial flux difference deviations from the target band are not operable, the indicated axial flux difference value for each OPERABLE excore channel shall be logged at least once per hour for the first 24 hours and half-hourly thereafter until the alarms are returned to an OPERABLE status. For the purpose of applying this specification, logged values of indicated axial flux difference must be assumed to apply during the previous interval between loggings.

C. QUADRANT POWER TILT RATIO

1. Except for PHYSICS TESTS, if the QUADRANT POWER TILT RATIO exceeds 1.02 but is less than 1.07, the rod position indication shall be monitored and logged once each shift to verify rod position within each bank assignment and, within two hours, one of the following steps shall be taken:

a. Correct the QUADRANT POWER TILT RATIO to less than 1.02.

b. Restrict core power level so as not to exceed RATED THERMAL POWER less 2% for every 0.01 that the QUADRANT POWER TILT RATIO exceeds 1.0.

3.10.C.2. If the QUADRANT POWER TILT RATIO exceeds 1.02 but is less than 1.07 for a sustained period of more than 24 hours, or if such a tilt recurs intermittently, the reactor shall be brought to the HOT SHUTDOWN condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

3. Except for PHYSICS TESTS if the QUADRANT POWER TILT RATIO exceeds 1.07, the reactor shall be brought to the HOT SHUTDOWN condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

4. If the core is operating above 85% power with one excore nuclear channel inoperable, then the core quadrant power balance shall be determined daily and after a 10% power change using either 2 movable detectors or 4 core thermocouples per quadrant, per Specification 3.11.

D. Rod Insertion Limits

1. The shutdown rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT when the reactor is critical or approaching criticality.

2. When the reactor is critical or approaching criticality, the control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT.

3. Insertion limits do not apply during PHYSICS TESTS or during periodic exercise of individual rods. The shutdown margin shown in Figure TS.3.10-1 must be maintained except for low power PHYSICS TESTING. For this test the reactor may be critical with all but one high worth full-length control rod inserted for a period not to exceed 2 hours per year provided a rod drop test is run on the high worth full-length rod prior to this particular low power PHYSICS TEST.

3.10.G. Inoperable Rod Limitations

1. An inoperable rod is a rod which (a) does not trip, (b) is declared inoperable under specification 3.10.E. or 3.10.H. or (c) cannot be moved by its drive mechanism and cannot be corrected within 8 hours.

2. The reactor shall be brought to the HOT SHUTDOWN condition within 6 hours should more than one inoperable rod be discovered during POWER OPERATION.

3. If the inoperable rod is located below the 200 step level and is capable of being tripped, or if the rod is located below the 30 step level whether or not it is capable of being tripped, then the insertion limits specified in the CORE OPERATING LIMITS REPORT apply.

4. If the inoperable rod cannot be located, or if the inoperable rod is located above the 30 step level and cannot be tripped, then the insertion limits specified in the CORE OPERATING LIMITS REPORT apply.

5. If POWER OPERATION is continued with one inoperable rod, the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days unless the rod is earlier made OPERABLE. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, THERMAL POWER shall be reduced to a level consistent with the safety analysis.

H. Rod Drop Time

At operating temperature and full flow, the drop time of each RCCA shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry. If the time is greater than 1.8 seconds, the rod shall be declared inoperable.

3.10.I. Monitor Inoperability Requirements

1. If the rod bank insertion limit monitor is inoperable, or if the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift, after a load change greater than 10 percent of RATED THERMAL POWER, and after 30 inches or more of rod motion.

2. If both the rod position deviation monitor and one or both of the quadrant power tilt monitors are inoperable for 2 hours or more, the nuclear overpower trip shall be reset to 93% of RATED THERMAL POWER in addition to the increased surveillance requirements.

3. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the calculated power tilt shall be logged every two hours after a load change greater than 10% of RATED THERMAL POWER

J. DNB Parameters

The following DNB related parameters limits shall be maintained during POWER OPERATION:

| | | |
|---|---|---|
| a. | Reactor Coolant System Tavg | $\leq$564°F |
| b. | Pressurizer Pressure | $\geq$2220 psia* |
| c. | Reactor Coolant Flow | $\geq$the value specified in the CORE OPERATING LIMITS REPORT |

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours. Compliance with c. is demonstrated by verifying that the parameter is within its limit after each refueling cycle.

*Limit not applicable during either a THERMAL POWER ramp increase in excess of (5%) RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of (10%) RATED THERMAL POWER

6.7.A.5. Annual Summaries of Meteorological Data

An annual summary of meteorological data shall be submitted for the previous calendar year in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability at the request of the Commission.

6.7.A.6. Core Operating Limits Report

a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Heat Flux Hot Channel Factor Limit ($F_Q^{RPT}$), Nuclear Enthalpy Rise Hot Channel Factor Limit ($F_{\Delta H}^{RTP}$), PFDH, K(Z) and V(Z) (Specifications 3.10.B.1, 3.10.B.2 and 3.10.B.3)

2. Axial Flux Difference Limits and Target Band (Specifications 3.10.B.4 through 3.10.B.9)

3. Shutdown and Control Bank Insertion Limits (Specification 3.10.D)

4. Reactor Coolant System Flow Limit (Specification 3.10.J)

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:

NSPNAD-8101-A, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version)

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

WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology", July, 1985

WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", August, 1985

WCAP-10924-P-A, "Westinghouse Large-Break LOCA Best-Estimate Methodology", December, 1988

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

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 shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be supplied upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

B. REPORTABLE EVENTS

The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified by a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Operations Committee and the results of this review shall be submitted to the Safety Audit Committee and the Vice President Nuclear Generation.

C. Environmental Reports

The reports listed below shall be submitted to the Administrator of the appropriate Regional NRC Office or his designate:

1. Annual Radiation Environmental Monitoring Report

(a) Annual Radiation Environmental Monitoring Reports covering the operation of the program during the previous calendar year shall be submitted prior to May 1 of each year.

(b) The Annual Radiation Environmental Monitoring Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 4.10.B.1. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

(c) The Annual Radiation Environmental Monitoring Reports 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 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 as soon as possible in a supplementary report.

(d) The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensees participation in the Interlaboratory Comparison Program, required by Specification 4.10.C.1.

2. Environmental Special Reports

(a) When radioactivity levels in samples exceed limits specified in Table 4.10-3, an Environmental Special Report shall be submitted within 30 days from the end of the affected calendar quarter. For certain cases involving long analysis time, determination of quarterly averages may extend beyond the 30 day period. In these cases the potential for exceeding the quarterly limits will be reported within the 30 day period to be followed by the Environmental Special Report as soon as practicable.

3. Other Environmental Reports (non-radiological, non-aquatic)

Written reports for the following items shall be submitted to the appropriate NRC Regional Administrator:

a. Environmental events that indicate or could result in a significant environmental impact casually related to plant operation. The following are examples: excessive bird impaction; onsite plant or animal disease outbreaks; unusual mortality of any species protected by the Endangered Species Act of 1973; or increase in nuisance organisms or conditions. This report shall be submitted within 30 days of the event and shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

b. Proposed changes, test or experiments which may result in a significant increase in any adverse environmental impact which was not previously reviewed or evaluated in the Final Environmental Statement or supplements thereto. This report shall include an evaluation of the environmental impact of the proposed activity and shall be submitted 30 days prior to implementing the proposed change, test or experiment.

D. Special Reports

Unless otherwise indicated, special reports required by the Technical Specifications shall be submitted to the appropriate NRC Regional Administrator within the time period specified for each report.

2.1 SAFETY LIMIT, REACTOR CORE

Bases continued

power levels of 91% and 74% respectively. For the 2235 psig and 2385 psig curves, the coolant average temperature at the core exit is equal to 650°F below power levels of 64% and 73% respectively.

The third and fourth criteria are evaluated using standard DNB methodology. For all four curves the DNBR is limiting at higher power levels. The area of safe operation is below these curves.

The plant conditions required to violate the limits in the lower power range are precluded by the self-actuated safety valves on the steam generators. The highest nominal setting of the steam generator safety valves is 1129 psig (saturation temperature 560°F). At zero power the difference between primary coolant and secondary coolant is zero and at full power it is 50°F. The reactor conditions at which steam generator safety valves open is shown as a dashed line on Figure TS.2.1-1.

Except for special tests, POWER OPERATION with only one loop or with natural circulation is not allowed. Safety limits for such special tests will be determined as a part of the test procedure.

The curves are conservative for the following nuclear hot channel factors:

$$F^N_{\Delta H} = F^{RTP}_{\Delta H}\,[1 + PFDH(1-P)] \;;\; \text{and } F^N_Q = F^{RTP}_Q$$

where:

- $F^{RTP}_Q$ is the $F_Q$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.
- $F^{RTP}_{\Delta H}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.
- PFDH is the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the CORE OPERATING LIMITS REPORT.

Use of these factors results in more conservative safety limits than would result from power distribution limits in Specification TS.3.10.

This combination of hot channel factors is higher than that calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are covered by Specification 3.10. Adverse power distribution factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits specified in the CORE OPERATING LIMITS REPORT assure that the DNB ratio is always greater at part power than at full power.

The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than 1.30 for Exxon Nuclear fuel and less than 1.17 for Westinghouse fuel.

## 3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases

Throughout the 3.10 Technical Specifications, the terms "rod(s)" and "RCCA(s)" are synonymous.

A. Shutdown Margin

Trip shutdown reactivity is provided consistent with plant safety analyses assumptions. One percent shutdown margin is adequate except for the steam break analysis, which requires more shutdown reactivity due to the more negative moderator temperature coefficient at end of life (when boron concentration is low). Figure TS.3.10-1 is drawn accordingly.

B. Power Distribution Control

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operations) and II (Incidents of Moderate frequency) events by: (a) maintaining the minimum DNBR in the core of greater than or equal to 1.30 for Exxon fuel and 1.17 for Westinghouse fuel during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. The ECCS analysis was performed in accordance with SECY 83-472. One calculation at the 95% probability level was performed as well as one calculation with all the required features of 10 CFR Part 50, Appendix K. The 95% probability level calculation used the peak linear heat generation rate specified in the CORE OPERATING LIMITS REPORT. The Appendix K calculation used the peak linear heat generation rate specified in the CORE OPERATING LIMITS REPORT for the $F_Q$ limit specified in the CORE OPERATING LIMITS REPORT. Maintaining 1) peaking factors below the $F_Q$ limit specified in the CORE OPERATING LIMITS REPORT during all Condition I events and 2) the peak linear heat generation rate below the value specified in the CORE OPERATING LIMITS REPORT at the 95% probability level assures compliance with the ECCS analysis.

During operation, the plant staff compares the measured hot channel factors, $F_Q^N$ and $F_{\Delta H}^N$, (described later) to the limits determined in the transient and LOCA analyses. The terms on the right side of the equations in Section 3.10.B.1 represent the analytical limits. Those terms on the left side represent the measured hot channel factors corrected for engineering, calculational, and measurement uncertainties.

$F_Q^N$ is the measured Nuclear Hot Channel Factor, defined as the maximum local heat flux on the surface of a fuel rod divided by the average heat flux in the core. Heat fluxes are derived from measured neutron fluxes and fuel enrichment.

The K(Z) function specified in the CORE OPERATING LIMITS REPORT is a normalized function that limits $F_Q$ axially. The K(Z) value is based on large and small break LOCA analyses.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

When a measurement of $F^N_{\Delta H}$ is taken, measurement error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup PHYSICS TESTS, at least once each effective full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would otherwise affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An accidental misalignment limit of 13 steps precludes a rod misalignment greater than 15 inches with consideration of maximum instrumentation error.

2. Control rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.

3. The control bank insertion limits specified in the CORE OPERATING LIMITS REPORT are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

B. Power Distribution Control (continued)

The permitted relaxation in $F^N_{\Delta H}$ and $F^N_Q$ allows for radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factor limits are met. In specification 3.10, $F^N_Q$ is arbitrarily limited for P less than or equal to 0.5 (except for low power PHYSICS TESTS).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically control of flux difference is required to limit the difference between the current value of Flux Difference ($\Delta I$) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = $\Delta I$/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that the $F_Q$ limit is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows: At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the full length rod control rod bank more than 190 steps withdrawn (i.e., normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium was noted, no allowances for excore detector error are necessary and indicated deviation from the indicated reference value but within the target band is permitted. The allowed deviation from the target flux difference as a function of THERMAL POWER is specified in the CORE OPERATING LIMITS REPORT.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

B. Power Distribution Control (continued)

In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The consequences of being outside the target band but within the limits specified in the CORE OPERATING LIMITS REPORT for power levels between 50% and 90% has been evaluated and determined to result in acceptable peaking factors. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated axial flux difference. In all cases the target band is the Limiting Condition for Operation. Only when the target band is violated do the limits specified in the CORE OPERATING LIMITS REPORT apply.

If, for any reason, the indicated axial flux difference is not controlled within the target band for as long a period as one hour, then xenon distributions may be significantly changed and operation at or below 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Condition II events the core is protected from overpower and a minimum DNBR of 1.30 for Exxon fuel and 1.17 for Westinghouse fuel by an automatic protection system. Compliance with operating procedures is assumed as a precondition for Condition II transients, however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

C. QUADRANT POWER TILT RATIO

QUADRANT POWER TILT RATIO limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation, e.g. rod misalignment, x-y xenon transient, or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication System or core instrumentation per Specification 3.10.F, and

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

H. Rod Drop Time

The required drop time to dashpot entry is consistent with the safety analysis.

I. Monitor Inoperability Requirements

If either the rod bank insertion limit monitor or rod position deviation monitor are inoperable, additional surveillance is required to ensure adequate shutdown margin is maintained.

If the rod position deviation monitor and quadrant power tilt monitor(s) are inoperable, the overpower reactor trip setpoint is reduced (and also power) to ensure that adequate core protection is provided in the event that unsatisfactory conditions arise that could affect radial power distribution.

Increased surveillance is required, if the quadrant power tilt monitors are inoperable and a load change occurs, in order to confirm satisfactory power distribution behavior. The automatic alarm functions related to QUADRANT POWER TILT must be considered incapable of alerting the operator to unsatisfactory power distribution conditions.

J. DNB Parameters

The RCS flow rate, $T_{avg}$, and Pressurizer Pressure requirements are based on transient analyses assumptions. The flow rate shall be verified by calorimetric flow data and/or elbow taps. Elbow taps are used in the reactor coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. If a reduction in flow rate is indicated below the value specified in the CORE OPERATING LIMITS REPORT, shutdown is required to investigate adequacy of core cooling during operation.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 92 AND 85 TO

FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-282 AND 50-306

## 1.0 INTRODUCTION

By letter dated November 17, 1989, (Ref. 1), Northern States Power Company (the licensee) proposed changes to the Technical Specifications (TS) for Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to the Core Operating Limits Report (COLR). The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of the TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988 (Ref. 2).

## 2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

2.1 The Definitions section of the TS was modified to include a definition of the Core Operating Limits Report that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with an NRC approved methodology that maintains the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

2.2 The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(a) Specification 3.10.B.1, 3.10.B.2 and 3.10.B.3

Power distribution parameters (the heat flux hot channel factor limit ($F_Q^{RTP}$), the nuclear enthalpy rise hot channel factor limit ($F_{delta\ H}^{RTP}$), PFDH, the normalized function that limits the heat flux hot channel factor axially K(Z), and the transient axial xenon factor V(Z) for these specifications are specified in the COLR. The

high neutron flux trip set point reduction factor, which is used when the nuclear enthalpy rise hot channel factor exceeds its limit, for Specification 3.10.B.3 is also specified in the COLR.

(b) Specification 3.10.B.4 through 3.10.B.9

The axial flux difference limits and target band for these specifications are specified in the COLR.

(c) Specification 3.10.D

Shutdown and control bank insertion limits for this specification are specified in the COLR.

(d) Specification 3.10.J

The reactor coolant system flow limit for this specification is specified in the COLR.

The bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

2.3 Specification 6.7.A.6 was added to the reporting requirements of the Administrative Controls section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, these specifications require that the values of these limits be established using NRC approved methodologies and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

(a) NSPNAD-8101-A, "Qualification of Reactor Physics Methods for Application of PI Units" (latest approved version).

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

(c) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

(d) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code" August 1985.

(e) WCAP-10924-P-A, "Westinghouse Large-Break LOCA Best-Estimate Methodology" December 1988.

(f) 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.

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits. These administrative changes do not in any way alter the action noticed or affect the inteal determination.

On basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in the TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that these changes are administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable. However, the staff requires that separate COLRs be issued for each Prairie Island Unit. This matter was discussed and agreed with by the licensee.

The licensee also proposed a number of Technical Specification changes that are not part of the changes needed to implement Generic Letter 88-16. The staff inadvertently omitted from the notice the deletion of the TS 3.10.D.3 concerning rod insertion limits, a provision that was applicable to the first fuel cycle only and has no effect on any subsequent fuel cycles for both units. The other administrative change appear in TS 3.10.B.7(a) where word "its" is replaced with the word "the". This is an editorial change which does not change the restrictive level or the intent of the specification. These administrative changes do not significantly alter the action noticed or affect the initial determination. These changes are administrative in nature and, therefore, we conclude that they are acceptable.

## 3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20, and changes recordkeeping, reporting, or administrative procedures or requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendements involve no significant hazards, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

## 4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and (3) security or to the health and safety of the public.

## 5.0 REFERENCES

1. Letter from Thomas M. Parker (NSP) to NRC, dated November 17, 1989.

2. Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

Principal Contributor: Daniel Fieno

Dated: March 13, 1990

ENCLOSURE 1

# SUPPLEMENT TO THE SAFETY EVALUATION

# BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENTS NOS. 87 AND 80 TO

# FACILITY OPERATING LICENSES NOS. DPR-42 AND DPR-60

# NORTHERN STATES POWER COMPANY

# PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS NOS. 1 AND 2

# DOCKETS NOS. 50-282 AND 50-306

## 1.0 INTRODUCTION

By letter dated April 3, 1989 from Dominic C. DiIanni to D. M. Musolf the design of the Median Signal Selector (MSS) and the ensuing Technical Specification changes were found to be acceptable by the staff. However, this acceptance was conditional in that the staff had not reviewed the verification and validation procedures that were in place for the software associated with the MSS. In other words, the final acceptance of the MSS was contingent upon the staff finding an adequate verification and validation program in place during a forthcoming staff audit at the vendor's site.

## 2.0 SOFTWARE DESCRIPTION

The Westinghouse MSS is programmed using the graphics Process Control Language. This high level language will enable the programmer to use menu-driven screens and interactive editing to configure process control loops, create a data base of input/output points and display the loops as configured during operation. The graphics language is comprised of four subsets. These subsets include Data Base Generation, Standard Modulating Control, Ladder Logic Control and Customer Control Schemes.

The MSS employs signal validation for input signals in order to reduce the probability of a spurious input signal (failed sensor or surveillance test error) which would cause an upset in the plant. A complete signal algorithm is applied and the signal validation algorithms use multiple measurements of each level variable. This validation rejects a failed channel where three channels are operational which is the normal condition for the level inputs to the MSS. Where only two channels are available, an arbitration signal selection method is used. If the two channels disagree significantly, then the determination as to which is correct is made by comparing them to another signal that is related to the primary measurements. The MSS is fault tolerant to any single channel failure. As a result of distributing redundant measurements across input cards, the signal validation algorithms also provide fault tolerance to an input card failure. In the event of multiple failures, selection of the proper value cannot be assured. Therefore, the MSS has been designed to automatically switch to the manual mode if multiple failures of an input variable are detected. This will prevent the failure from propagating to a disturbance of an output. An alarm/annunciator is actuated if any channel failure is detected. A separate alarm/annunciator is also actuated for an automatic switch to the manual mode which is indicative of a multiple failure.

## 3.0 REVIEW DISCUSSION

The primary objectives of the MSS are to eliminate the need for the low feedwater flow reactor trip and to enhance the reliability of the Feedwater Control System. These are both accomplished by preventing a failed instrument channel from causing a control system to fail which would initiate a planned transient that may require a protection system action. Prairie Island and the vendor (Westinghouse) state that since no adverse control system action may now result from a single, failed protection system as would otherwise have taken place for the old design, IEEE Std 279-1971 need not be considered for the MSS design. Because of the importance of the MSS software performing its function in the correct manner which is to totally eliminate the control/protection system interaction concern, the staff concluded that an audit of the verification and validation process utilized by the vendor for MSS software development should be performed by the staff.

The staff performed the verification and validation audit on July 13 and 14, 1989 at the Westinghouse site. During the staff audit of the verification and validation plan and its implementation, the advanced digital feedwater control system (ADFCS) requirements and the functional diagrams were reviewed by the staff. The actual requirements reviewed were included with the Revision 2 version dated October, 1988. The documents reviewed that were incorporated by the functional requirement were the following: Input Signal Validation, Feedwater Flow Controller-High/Low Power Modes, Cv Demand Calculation, Control Valve Sequencing and Tracking Logic and the Median Signal Selector Logic, Arbitrator Signal Select Logic, and the Index of Symbols for ADFCS Functional Diagrams.

The staff reviewed a process that was part of the software flow chart which was called the Configuration Certification Programs. Configuration Certification is a formal activity devised to minimize design errors and provide an overall assurance that the specified functional requirements are implemented in the hardware and software as a system. Configuration Certification is accomplished through: 1. software development through a structured process using documented procedures, 2. independent review of design documentation to ensure that the median signal selector functional requirements are adequately translated to support design requirement, and 3. independent testing and evaluation to demonstrate median signal document decomposed the functional requirements into detailed sub-requirements. For each sub-requirement, a test or series of tests were identified to ensure that the specific sub-requirement was satisfied. The median signal selector design basis was reviewed along with the software development program which included the design cycle and the maintenance cycle. An overview of the median signal selector testing program was performed. This overview included testing in a dynamic simulation lab, factory acceptance testing and the generic algorithm testing that was performed. A summary of the testing discrepancies and their resolutions were presented to the staff. The staff concluded that adequate attention and depth had been maintained during the testing and the resolution phase of the software development program.

The MSS design incorporates a self-diagnostic testing feature. The self-diagnostics are automatically executed during normal operation of the system and do not disrupt the real time performance of the process. The major diagnostic features are as follows: 1. if a signal is out of range the trouble alarm is actuated and the median of the three level input signal is used for control

purposes. 2. the input/output cards have status lights on their card edges to aid in trouble shooting and a test card is available to provide additional diagnostics on the I/O bus controller. 3. should an active DPU failover to the backup, a trouble alarm will be generated.

The MSS is provided with the capability for on-line testing. Signal selector testing consists of monitoring the three steam generator level input signals and the selected median signal at an engineering work station. Comparisons for correctness can then be made. The MSS can be tested concurrently with the protection system inputs. As the protection system input signal is varied, that instrument channel which represents the median signal will also be altered allowing the technician to ensure that an improper signal is not passed through the MSS.

The required frequency of testing of the MSS is identical to other control instrumentation which is every refueling outage. However, the licensee has stated that the MSS is presently tested concurrently with the monthly required testing of the steam generator level channels. Satisfactory results are based on observing that an intentionally failed channel is not selected by the MSS for control. The MSS function is checked for both the high and low failure of the input signal. The staff agrees with these voluntary monthly testing actions associated with the MSS. The staff strongly recommends that these monthly testing actions be undertaken for several cycles of operation due to the importance of the MSS design.

In addition the staff recommends that the licensee maintain a log that will list the troubles encountered during this testing period. This log should also be used to document the changes made to the MSS during these initial cycles of operation. This more formal means of documentation and tracking log will provide an aid for evaluating and maintaining the reliability of the MSS design.

## 4.0 SUMMARY

Since the MSS is within the feedwater control system which is a non-safety-related system, the staff had concluded that the guidance provided in the American National Standard ANS/IEEE-ANS-7.4.3.2.-1982, "Application Criteria for Programmable Digital Computer System in Safety Systems of Nuclear Power Generating Stations", did not have to be followed in its entirety. However, by employing the MSS design, the licensee was able to delete a reactor trip that had been initially placed in the Prairie Island design to ensure that IEEE 279-1971 requirements were met. Taking this into consideration along with the relative importance of the feedwater control system that (even though it is not relied on to perform safety functions following anticipated operational occurrences or accidents) does control a plant process which has a significant impact on plant safety, the staff established a review objective. The objective of the verification and validation audit was to confirm that there was assurance that an acceptable level of ANSI/IEEE-ANS-7.4.3.2 was followed by the licensee and the vendor. The staff position was that verification and validation independence was not necessary, however the remaining guidance of this standard should be followed or adequate justification provided.

On the basis of our review of the interface between the MSS and the reactor protection system, the staff concludes that the system satisfies IEEE-279 with regard to control and protection system interaction. Therefore, the staff finds that GDC 24 is satisfied. Furthermore, the staff concludes that there was an acceptable structured and extensive verification and validation plan in place that would detect errors and oversights in the software and the plan was sufficiently broad in scope to address discrepancies that could have occurred in the design process. The documentation reviewed was structured and provided an adequate measure of traceability.

Therefore, on the basis of our review of the software design and its verification and validation process, the staff concludes that the MSS meets an acceptable level of the guidelines provided in ANSI/IEEE-ANS-7.4.3.2 and Regulatory Guide 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants".

In summary, we conclude that the MSS meets all of the applicable guidelines and regulations and that its utilization as discussed in the previous safety evaluation is acceptable. Therefore the staff concludes that the licensee has demonstrated an acceptable verification and validation program and the technical specification TS 2.3.A.3 (c) dealing with reactor trip initiated by "Low Steam generator water level , 15% narrow range instrument in coincidence with steam/feedwater mismatch flow $1.0 \times 10^6$ lbs/hrs" may be deleted. The staff recommends the following: 1. The monthly testing actions proposed by the licensee and recommended by the vendor be continued for several cycles of operation. 2. The licensee should maintain a log that lists the troubles encountered during the above testing period and the modifications made to the MSS during these initial cycles. This log should be maintained by the licensee so that a basis will be provided for an ongoing evaluation of the reliability of the MSS.

## 5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on April 3, 1989 (54 FR 13445) which is applicable to this supplement.

Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this supplement to amendments Nos. 87 and 80 will not have a significant effect on the quality of the human environment.

## 6.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Jerry Mauck

Date: March 13, 1990