October 30, 1998

Mr. Roger O. Anderson, Director
Nuclear Energy Engineering
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: INOPERABLE ROD POSITION INDICATOR CHANNELS (TAC NOS. MA3876 AND MA3877)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 139 to Facility Operating License No. DPR-42 and Amendment No. 130 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated October 23, 1998, as supplemented October 26, 1998. This request was treated as an emergency amendment in accordance with 10 CFR 50.91(a)(5).

The amendments clarify the conditions that constitute operable Individual Rod Position Indication (IRPI) system channels, provide for an allowed out of service time for inoperable IRPI indicator channels, and provide compensatory measures to be taken when any channel is determined to be inoperable.

A copy of our related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
Carl F. Lyon
Carl F. Lyon, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures: 1. Amendment No. 139 to DPR-42
2. Amendment No. 130 to DPR-60
3. Safety Evaluation

cc w/encl: See next page

DISTRIBUTION: See attached page

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DATED: October 30, 1998

AMENDMENT NO. 139 TO FACILITY OPERATING LICENSE NO. DPR-42-PRAIRIE ISLAND UNIT 1
AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. DPR-60-PRAIRIE ISLAND UNIT 2

Docket File (50-282, 50-306)
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 139
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 October 23, 1998, as supplemented October 26, 1998, 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-42 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 139 , 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Cynthia A Carpenter

Cynthia A. Carpenter, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 30, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 139

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 vertical lines indicating the area of change.

REMOVE	INSERT
TS-iv	TS-iv
TS-x	TS-x
TS.3.10-6	TS.3.10-6
--	TS.3.10-6A
B.3.10-9	B.3.10-9
--	B.3.10-9A
--	B.3.10-9B

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3.10.E. Rod Misalignment Limitations

1. If a rod cluster control assembly (RCCA) is misaligned from its bank by more than 24 steps, the rod will be realigned or the core power peaking factors shall be determined within 2 hours, and Specification 3.10.B applied. If peaking factors are not determined within 2 hours, the high neutron flux trip setpoint shall be reduced to 85 percent of rating.

2. If the misaligned RCCA is not realigned within a total of 8 hours, the RCCA shall be declared inoperable.

3.10.F. Rod Position Indication System

1. In MODE 1 each channel of the Rod Position Indication System shall be OPERABLE, capable of determining the control rod positions within the following (except as specified in 3.10.F.2 or 3.10.F.3 below):

   a. With bank demand position greater than or equal to 215 steps, or less than or equal to 30 steps, the difference between the individual rod position indication and the demand position for the corresponding group step counter shall be no greater than ± 24 steps, or

   b. With bank demand position between 30 and 215 steps, the difference between the individual rod position indication and the demand position for the corresponding group step counter shall be no greater than ± 12 steps.

2. In MODE 1 with one rod position indicator per group inoperable for one or more groups either:

   a. Verify the position of rod(s) with inoperable position indicator(s) indirectly using the moveable incore detectors at least once per 8 hours, or

   b. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

3. In MODE 1 with more than one rod position indicator per group inoperable for one or more groups:

   a. Verify the position of rods with inoperable position indicators indirectly using the moveable incore detectors at least once per 8 hours, and

   b. Verify the position of rods with inoperable position indicators indirectly using the moveable incore detectors within 4 hours after rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of their position, and

   c. Monitor and record the demand position for the corresponding group step counters for rods with inoperable position indicators at least once per hour, and

   d. Monitor and record reactor coolant system average temperature at least once per hour, and

   e. Restore inoperable position indicators to OPERABLE status within 24 hours such that a maximum of one rod position indicator per group is inoperable.

4. If the requirements of Specification 3.10.F.3 cannot be met, then place the affected unit in at least HOT SHUTDOWN within the following 6 hours.

5. If a control rod with an inoperable rod position indicator is found to be misaligned during the verification of rod position required by Specifications 3.10.F.2.a, 3.10.F.3.a or 3.10.F.3.b above, then apply the requirements of Specification 3.10.E.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

D. Rod Insertion Limits (continued)

as stated above. Therefore, this specification has been written to further minimize the likelihood of any hypothesized event during the performance of these tests later in life. This is accomplished by limiting to two hours per year the time the reactor can be in this type of configuration, and requiring that a rod drop test is performed on the rod to be measured prior to performance of test.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

E. Rod Misalignment Limitations

Rod misalignment requirements are specified to ensure that power distributions more severe than those assumed in the safety analyses do not occur.

The rod misalignment limitations are linked closely with the Rod Position Indication System operability requirements of 3.10.F. The relaxed rod position indicator channel operability requirements at less than or equal to 30 steps or greater than or equal to 215 steps of up to ± 24 steps indicated position is allowed since the reactivity worths of control rods in these ranges are sufficiently small that this misalignment will have no appreciable effect on core power distributions.

F. Rod Position Indication System

The rod position indicator channel is sufficiently accurate to detect a rod ±7.5 inches away from its demand position in the center region of the core. A misalignment less than 15 inches in the center region of the core does not lead to over-limit power peaking factors. In the peripheral core regions (less than or equal to 30 steps or greater than or equal to 215 steps) a misalignment less than 22.5 inches will not lead to over-limit power peaking factors due to small control rod reactivity worth in this region of the core. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or core thermocouples, and/or movable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 15-inch misalignment would have no effect on power distributions. Therefore, it is necessary to apply the indirect checks following significant rod motion.

Specifications 3.10.F.2 and 3.10.F.3 provide actions to be taken when rod position indicators are determined to be inoperable. The actions to be taken are dependent on how many rod position indicators are inoperable per group. When dealing with rod position indicators associated with a control rod bank that does not contain multiple groups, the bank should be considered a single group for the purposes of entry into Specifications 3.10.F.2 or 3.10.F.3.

Specification 3.10.F.3.c requires that the demand position for the corresponding group step counters for rods with inoperable position indicators be monitored and recorded on an hourly basis. This requirement is intended to provide a periodic assessment of rod position such that it can be determined if rod movement in excess of 24 steps has occurred since the last determination of rod position. If rod movement in excess of 24 steps has occurred since the last determination of rod position, the requirements of Specification 3.10.F.3.b are to be implemented.

Specification 3.10.F.3.d requires that reactor coolant system average temperature be monitored and recorded on an hourly basis. Monitoring and recording of the reactor coolant system average temperature may provide early detection of mispositioned or dropped rods.

Specifications 3.10.F.2.a and 3.10.F.3.a require that the position of rods with inoperable position indicators be verified indirectly using the moveable incore detectors every 8 hours. The verification of rod position every 8 hours is adequate for continued plant operation since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

The reduction of THERMAL POWER to less than 50% of RATED THERMAL POWER required by Specification 3.10.F.2.b puts the core into a condition where rod position is not significantly affecting core peaking factors. The allowed completion time of 8 hours is reasonable, based on operating experience, for reducing power to less than 50% RATED THERMAL POWER from full power conditions without challenging plant systems.

Specification 3.10.F.3.b ensures that verification of rod position is initiated promptly following the movement of rods with inoperable position indication in excess of 24 steps in one direction, since the rod position was last determined or was last available from an OPERABLE rod position indication channel. The four hour allowance for completion of this action allows adequate time for personnel to be called in and for them to complete the rod position verification using the moveable incore detectors.

When more than one rod position indication channel per group is inoperable, the position of the rod(s) can still be determined by use of the moveable incore detectors. Based on experience, normal power operation does not require excessive movement of control rods. If one or more banks has been significantly moved, the action specified by Specification 3.10.F.3.b is required. Therefore, verification of rod position within every 8 hours per Specification 3.10.F.3.a is adequate for allowing continued full power operation for up to 24 hours, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour allowed out of service time also provides sufficient time to troubleshoot and restore the IRPI system to operation following a component failure in the system, while avoiding the challenges associated with a plant shutdown.

G. Control Rod Operability Limitations

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully-inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The four-week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130
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 October 23, 1998, as supplemented October 26, 1998, 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. 130 , 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Cynthia A. Carpenter

Cynthia A. Carpenter, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 30, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 130

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 vertical lines indicating the area of change.

REMOVE	INSERT
TS-iv	TS-iv
TS-x	TS-x
TS.3.10-6	TS.3.10-6
--	TS.3.10-6A
B.3.10-9	B.3.10-9
--	B.3.10-9A
--	B.3.10-9B

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3.10.E. Rod Misalignment Limitations

1. If a rod cluster control assembly (RCCA) is misaligned from its bank by more than 24 steps, the rod will be realigned or the core power peaking factors shall be determined within 2 hours, and Specification 3.10.B applied. If peaking factors are not determined within 2 hours, the high neutron flux trip setpoint shall be reduced to 85 percent of rating.

2. If the misaligned RCCA is not realigned within a total of 8 hours, the RCCA shall be declared inoperable.

3.10.F. Rod Position Indication System

1. In MODE 1 each channel of the Rod Position Indication System shall be OPERABLE, capable of determining the control rod positions within the following (except as specified in 3.10.F.2 or 3.10.F.3 below):

   a. With bank demand position greater than or equal to 215 steps, or less than or equal to 30 steps, the difference between the individual rod position indication and the demand position for the corresponding group step counter shall be no greater than ± 24 steps, or

   b. With bank demand position between 30 and 215 steps, the difference between the individual rod position indication and the demand position for the corresponding group step counter shall be no greater than ± 12 steps.

2. In MODE 1 with one rod position indicator per group inoperable for one or more groups either:

   a. Verify the position of rod(s) with inoperable position indicator(s) indirectly using the moveable incore detectors at least once per 8 hours, or

   b. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

3. In MODE 1 with more than one rod position indicator per group inoperable for one or more groups:

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   b. Verify the position of rods with inoperable position indicators indirectly using the moveable incore detectors within 4 hours after rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of their position, and

   c. Monitor and record the demand position for the corresponding group step counters for rods with inoperable position indicators at least once per hour, and

   d. Monitor and record reactor coolant system average temperature at least once per hour, and

   e. Restore inoperable position indicators to OPERABLE status within 24 hours such that a maximum of one rod position indicator per group is inoperable.

4. If the requirements of Specification 3.10.F.3 cannot be met, then place the affected unit in at least HOT SHUTDOWN within the following 6 hours.

5. If a control rod with an inoperable rod position indicator is found to be misaligned during the verification of rod position required by Specifications 3.10.F.2.a, 3.10.F.3.a or 3.10.F.3.b above, then apply the requirements of Specification 3.10.E.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

D. Rod Insertion Limits (continued)

as stated above. Therefore, this specification has been written to further minimize the likelihood of any hypothesized event during the performance of these tests later in life. This is accomplished by limiting to two hours per year the time the reactor can be in this type of configuration, and requiring that a rod drop test is performed on the rod to be measured prior to performance of test.

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Specifications 3.10.F.2 and 3.10.F.3 provide actions to be taken when rod position indicators are determined to be inoperable. The actions to be taken are dependent on how many rod position indicators are inoperable per group. When dealing with rod position indicators associated with a control rod bank that does not contain multiple groups, the bank should be considered a single group for the purposes of entry into Specifications 3.10.F.2 or 3.10.F.3.

Specification 3.10.F.3.c requires that the demand position for the corresponding group step counters for rods with inoperable position indicators be monitored and recorded on an hourly basis. This requirement is intended to provide a periodic assessment of rod position such that it can be determined if rod movement in excess of 24 steps has occurred since the last determination of rod position. If rod movement in excess of 24 steps has occurred since the last determination of rod position, the requirements of Specification 3.10.F.3.b are to be implemented.

Specification 3.10.F.3.d requires that reactor coolant system average temperature be monitored and recorded on an hourly basis. Monitoring and recording of the reactor coolant system average temperature may provide early detection of mispositioned or dropped rods.

Specifications 3.10.F.2.a and 3.10.F.3.a require that the position of rods with inoperable position indicators be verified indirectly using the moveable incore detectors every 8 hours. The verification of rod position every 8 hours is adequate for continued plant operation since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

The reduction of THERMAL POWER to less than 50% of RATED THERMAL POWER required by Specification 3.10.F.2.b puts the core into a condition where rod position is not significantly affecting core peaking factors. The allowed completion time of 8 hours is reasonable, based on operating experience, for reducing power to less than 50% RATED THERMAL POWER from full power conditions without challenging plant systems.

Specification 3.10.F.3.b ensures that verification of rod position is initiated promptly following the movement of rods with inoperable position indication in excess of 24 steps in one direction, since the rod position was last determined or was last available from an OPERABLE rod position indication channel. The four hour allowance for completion of this action allows adequate time for personnel to be called in and for them to complete the rod position verification using the moveable incore detectors.

When more than one rod position indication channel per group is inoperable, the position of the rod(s) can still be determined by use of the moveable incore detectors. Based on experience, normal power operation does not require excessive movement of control rods. If one or more banks has been significantly moved, the action specified by Specification 3.10.F.3.b is required. Therefore, verification of rod position within every 8 hours per Specification 3.10.F.3.a is adequate for allowing continued full power operation for up to 24 hours, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour allowed out of service time also provides sufficient time to troubleshoot and restore the IRPI system to operation following a component failure in the system, while avoiding the challenges associated with a plant shutdown.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 139

TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 130 TO FACILITY OPERATION LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated October 23, 1998, as supplemented October 26, 1998, the Northern States Power Company (NSP or the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License No. DPR-42 for the Prairie Island Nuclear Generating Plant, Unit 1, and Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit 2. The proposed amendments would clarify the conditions that constitute operable Individual Rod Position Indication (IRPI) system channels, provide for an allowed out of service time for inoperable IRPI indicator channels, and provide compensatory measures to be taken when any channel is determined to be inoperable. The licensee requested that the proposed amendments be treated as emergency amendments as discussed in Section 3.0 of this Safety Evaluation.

2.0 EVALUATION

2.1 Background

Prairie Island Units 1 and 2 have frequently experienced IRPI deviations of greater than 12 steps from the bank demand position involving more than one rod control cluster assembly (RCCA) during startups, shutdowns, and large power changes. The instrument accuracy of the IRPI at Prairie Island is affected by steady-state non-linearity in the relationship between actual rod position and analog indication and by transient thermal drift. These characteristics are generic to all Westinghouse-supplied analog IRPI systems. If the bank demand position is between 30 and 215 steps and the IRPI channel differs by more than 12 steps, the licensee considers the RCCA misaligned, in accordance with TS 3.10.E.2.b. In order to verify that the RCCA is not misaligned, the position of the RCCA is checked using core instrumentation (excore detector and/or thermocouples and/or movable incore detectors), as directed by TS 3.10.F.1.a. If the actual position of the RCCA is verified using core instrumentation, the licensee considers the RCCA to be aligned and operable. NSP also considers the IRPI to be inaccurate, but operable, as long as it tracks relative rod motion and can indicate a dropped rod. Since the inaccurate IRPI is unable to provide a valid input to the Rod Deviation Monitor, the licensee considers the Rod Deviation Monitor to be inoperable for the affected IRPI channels, in accordance with TS 3.10.I.

However, on October 14, 1998, the staff informed the licensee that they disagree with this interpretation and informed the licensee that it is inappropriate to consider the IRPI inoperable in order to determine actual RCCA position in accordance with TS 3.10.F.1.a, but operable to avoid a TS prescribed shutdown in accordance with TS 3.10.F.2.

The licensee agreed with the staff position and requested changes to the Prairie Island TS that would allow additional flexibility with respect to plant operation with IRPI channels. This license amendment request proposes changes based on changes previously approved for Callaway and Wolf Creek plants and on the guidance in the improved Standard Technical Specifications (NUREG-1431).

2.2 Evaluation of Proposed Change

In accordance with TS 3.10.F.2, should more than one RPI channel per group or more than two RPI channels per bank be found to be inoperable, the plant must be brought to the Hot Shutdown condition. As described above, this presents problems for any plant evolution requiring power changes; the current TS do not provide adequate flexibility to perform such evolutions as required plant maintenance and testing at power. Power changes needed to perform such required plant evolutions could result in an unnecessary plant shutdown. To correct this situation, the licensee has requested that its current TS 3.10.F, Inoperable Rod Position Indicator Channels, and TS 3.10.E, Rod Misalignment Limitations, be modified based on the current Callaway and Wolf Creek TS. The request would provide a completion time for conducting troubleshooting, repair, and replacement of RPI components during power operations. This safety evaluation addresses the licensee's proposed TS changes.

The licensee justifies the TS change request based upon the fact that the safety analyses do not assume any dependence on the operator to monitor and properly position control rods. The safety analyses assume that the control rods are maintained within the TS Alignment and Insertion Limits. Whenever an IRPI indicates that a rod is misaligned, the actual rod position must be verified utilizing the moveable incore detectors. Assurance that the control rods do not exceed their insertion limits is provided by the Rod Insertion Limit Monitor in the emergency response computer system (ERCS), which receives input directly from a pulse-to-analog converter fed from the rod control system. Proper function of the Rod Insertion Limit Monitor is not dependent upon the operability of IRPI, except for the condition where power is lost to the IRPI cabinet. Maintaining control rods aligned and within insertion limits ensures that acceptable power limits are maintained and that minimum shutdown margin is maintained, and that there is not a decrease in the minimum departure from nucleate boiling ratio (DNBR).

Continued plant operation with IRPI inoperable due to either a temperature-induced increase in the output signal nonlinear bias or some other mechanism will not increase the probability that operators will either misposition the control rods or fail to observe a control rod misalignment. With no other indication of misalignment, there is no reason to expect that a control rod with an out-of-specification IRPI indicator channel is misaligned. Prairie Island has not had a rod misalignment due to any reason. Random rod misalignment failure together with an associated IRPI failure is a very low probability event.

The Prairie Island license amendment request adequately addresses the specific TS changes. With inoperable IRPI, the licensee's compensatory TS actions require that rod position/alignment be determined indirectly using the moveable incore flux detectors, that reactor coolant system temperature be monitored and recorded, and that the demand position for group step counters with inoperable IRPI be monitored and recorded. Since the licensee's safety analyses do not assume any dependence upon the operator to monitor and properly position control rods, the staff accepts that these actions will sufficiently ensure the minimum DNBR limit will not be violated. These proposed changes to the TS will not impact the ability of the plant staff to maintain operation within the TS core thermal limits. Furthermore, the reactor protection system detection features and the engineered safety feature mitigation features are unaffected by these proposed changes.

The addition of a 24-hour allowed out-of-service time provides sufficient time to troubleshoot and restore inoperable IRPI channels, while avoiding the challenges associated with a plant shutdown. Overall plant safety would be enhanced by having a reasonable opportunity to maintain steady-state operation rather than being immediately forced, without control rod position indication, to perform the large number of control rod movements required during a plant shutdown.

The staff finds the TS changes proposed by the licensee to be acceptable.

3.0 DESCRIPTION OF EMERGENCY CIRCUMSTANCES

The Commission's regulations in 10 CFR 50.91 contain provisions for issuance of an amendment where the Commission finds that emergency circumstances exist, in that failure to act in a timely way would result in derating or shutdown of a nuclear power plant. The emergency exists in this case in that the proposed amendments are needed to prevent shutdown of Prairie Island.

At times during rod movement the difference in rod position between the IRPI system outputs and the control rod group demand counters has exceeded the rod misalignment limitations in the Prairie Island TS 3.10.E.2.b. In each instance the performance of plant procedures to verify actual rod positions has confirmed that no control rod has been misaligned from its group. Since the control rods were not misaligned, the operability of IRPI was next examined, but current TS 3.10.F does not clearly define IRPI operability. The licensee initiated NCR (nonconformance report) 19970613 to track investigation and resolution of the issue.

The instrument accuracy of the IRPI system at Prairie Island is affected by the following characteristics which are generic to all Westinghouse-supplied analog IRPI systems:

1) Steady-state non-linearity in the relationship between actual rod position and analog indication. This non-linearity is fixed, measurable, and reproducible. The IRPI output produces an S-shaped curve that can be calibrated to intersect group demand counter

output at two points. These points are chosen to provide a "best fit," but some offset between the two indications will exist over portions of the range of control rod motion.

2) Transient thermal drift, which is due to the change in detector characteristics caused by temperature changes after rod motion. This transient effect is measurable and predictable. After completion of rod motion the detector output returns to essentially the steady-state value within about 1 hour.

The licensee examined the Updated Safety Analysis Report (USAR) and the TS bases to identify the required functions of the IRPI system, which are to:

1) Provide relative IRPI in order to allow verification of rod motion upon demand.

2) Provide actual rod position indication to control board indicator and input signals to the ERCS (emergency response computer system) Rod Deviation Monitor for detection of a misaligned rod and annunciation of the control rod deviation.

3) Provide indication of each control rod at the bottom of the core following a reactor trip or plant shutdown.

The licensee had concluded that only two of the above capabilities must be maintained to provide for operability of the IRPI system and that this definition of operability was independent of system accuracy:

1) The IRPI channel(s) must trend with the control rod group demand position indication and provide the operator with the ability to manually detect rod misalignment.

2) The IRPI channel(s) must provide indication of the control rods at the bottom of the core.

The licensee concluded that an IRPI channel continued to be operable when the control rods were not misaligned but the difference between demand indication and IRPI indication was greater than 12 steps but less than 24 steps. With a difference greater than 24 steps, the IRPI channel should be logged out-of-service.

On October 14, 1998, the licensee was informed that the NRC staff had completed a review of the definition of IRPI operability utilized at Prairie Island and had concluded that the licensee's definition of operability was not adequate. The NRC staff clarified during a call with the licensee staff on October 15, 1998, that an IRPI channel was inoperable anytime the IRPI indicated position differed from the group demand position by greater than the rod misalignment limitations in TS.

As discussed above, changes in rod position, as required during power changes, normally result in some amount of additional bias in the IRPI indicated rod positions, which can result in an increase in the difference between the IRPI indicated position and the group demand position. It is typical during significant power changes to see several IRPI channels affected by this phenomenon, some of which may exceed the rod misalignment limitations in TS between IRPI indicated position and group demand position.

Since operability of the IRPI channels is based on not exceeding a difference between IPRI indicated position and demand position by greater than the rod misalignment limitations in TS, then any significant power change can result in multiple IRPI channels being declared inoperable. Per the requirements of current Prairie Island TS 3.10.F.2, should more than one IRPI channel per group or more than two IPRI channels per bank be found to be inoperable, the plant must be brought to the Hot Shutdown condition.

This presents problems for any plant evolution requiring a significant power reduction, such as turbine valve testing or condenser cleaning, where the unit is not taken to the Hot Shutdown condition. Based on past performance of the IRPI channels and the clarified definition of IRPI operability, the current Prairie Island TS do not provide adequate flexibility to perform such required plant maintenance and testing at power. The reduction in power required to perform the testing would likely result in a plant shutdown.

In addition, because the IRPI drift phenomenon is also experienced during power increases, the application of the ± 12 step operability criteria in combination with the requirements of current TS 3.10.F.2, may prevent the plant from returning to power following shutdowns or outages.

A power reduction of Prairie Island Unit 1 for turbine valve testing and condenser tube cleaning was scheduled to be performed the weekend of October 17, 1998. This power reduction was scheduled prior to receiving the call from the NRC staff on October 14, 1998. Implementation of the revised definition of IRPI operability, combined with the restrictions in the current Prairie Island TS, made it impossible to perform the power reduction and testing scheduled for October 17, 1998. Initiation of the power reduction would have likely resulted in Unit 1 being taken to the Hot Shutdown condition and may have prevented the unit from restarting. As a result of the revised definition of IRPI operability, the power reduction was postponed until October 31, 1998.

The turbine valve testing originally scheduled to be performed on October 17, 1998, is required by TS 4.1. It must be completed by December 3, 1998. It has been scheduled to be performed the weekend of October 31, 1998, so that it does not have to be performed during the Unit 2 refueling outage scheduled to begin on November 7, 1998.

The licensee does not consider it prudent to perform a planned Unit 1 power reduction and turbine valve test during the Unit 2 refueling outage. The safety concerns focus on a potential for increased distractions to operating personnel that could occur by performing a Unit 1 power reduction and valve test during a refueling outage on the other unit. Such distractions have been shown through analysis of industry events to contribute to errors that could lead to undesirable plant transients or events. Plant safety is enhanced by performing the Unit 1 valve testing prior to the Unit 2 refueling outage. The current Prairie Island TS do not provide adequate flexibility for either unit to perform a significant power reduction for routine planned maintenance or for emergent unplanned maintenance. The reduction in power would likely result in a plant shutdown. Additionally, the units could be prevented from restarting following shutdown by the same IRPI phenomenon. The staff has determined that the licensee used its best efforts to make a timely application.

Accordingly, the Commission has determined that emergency circumstances exist pursuant to 10 CFR 50.91(a)(5) and could not have been avoided, that the submittal was timely, and that the licensee did not create the emergency condition.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of the facility in accordance with the amendment would not:

(1) involve a significant increase in the probability or consequences of any accident previously evaluated;

(2) create the possibility of a new or different kind of accident from any previously evaluated; or

(3) involve a significant reduction in a margin of safety.

These amendments have been evaluated against the standards in 10 CFR 50.92 and the staff's final determination is presented below. They do not involve a significant hazards consideration because the change would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not affect any system that is a contributor to initiating events for previously evaluated design basis accidents. Neither do the changes significantly affect any system that is used to mitigate any previously evaluated design basis accidents. Therefore, the proposed changes do not involve a significant increase in the probability or consequence of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed changes do not alter the design, function, or operation of any plant component and do not install any new or different equipment; therefore, the possibility of a new or different kind of accident from those previously analyzed has not been created.

While continued operation of the plant for 24 hours with a number of IRPI channels inoperable does slightly increase the possibility that a misaligned control rod or mispositioned control group might go undetected by the operators for some time period, this possibility is not new and has been addressed in previously performed safety analyses, where bounding conservative assumptions on rod position were used.

3. Involve a significant reduction in the margin of safety.

The proposed changes do not alter the initial conditions assumed in deterministic analyses associated with either the reactor coolant system boundary or fuel cladding; therefore, these changes do not involve a significant reduction in the margins of safety.

Accordingly, the Commission has determined that the amendments involve no significant hazards consideration.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC 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 made a final finding that the amendments involve no significant hazards consideration. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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