Northern States Power Company

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November 24, 1993

10 CFR Part 50
Section 50.90

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

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Revision 2 to License Amendment Request Dated September 21, 1992
Instrumentation Specification Changes

Attached is a revision to the September 21, 1992 License Amendment Request which proposed changes to surveillance test intervals and allowed outage times for the engineered safety features and reactor protection system instrumentation. This revision incorporates changes which resulted from NRC Staff review comments and comments received during plant operations and technical staff training on the proposed changes.

Exhibit A for the original License Amendment Request, which contained the description of the proposed changes, the reasons for requesting the changes and the supporting safety evaluation/significant hazards determination has been revised to incorporate the proposed revisions. The revisions to Exhibit A are denoted by sidebars. The proposed revisions do not change the conclusions of the original significant hazards evaluation.

Exhibit B contains marked up Technical Specification pages with the original proposed changes and all revisions incorporated. Exhibit C contains revised Technical Specification pages with the original proposed changes and all revisions incorporated. Exhibit D consists of the Technical Specification pages submitted by the original September 21, 1992 License Amendment Request and the December 29, 1992 revision, marked up to indicate the changes being incorporated into the pages by this revision.

A001

Please contact Gene Eckholt (612-388-1121 Ext 4663) if you have any questions related to this revision of the September 21, 1992 License Amendment Request.

Roger O Anderson

Roger O Anderson
Director
Licensing and Management Issues

c: Regional Administrator-III, NRC
NRR Project Manager, NRC
Senior Resident Inspector, NRC
State of Minnesota
Attn: Kris Sanda
J E Silberg

Attachments: Affidavit

Exhibit A - Evaluation of Proposed Changes to the Technical Specifications

Exhibit B - Proposed Changes Marked Up on Existing Technical Specification Pages

Exhibit C - Revised Technical Specification Pages

Exhibit D - Changes to Technical Specification Pages Since December 29, 1992 Revision

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT DOCKET NO. 50-282
50-306

REVISED REQUEST FOR AMENDMENT TO
OPERATING LICENSES DPR-42 & DPR-60

NOVEMBER 24, 1993 REVISION TO
LICENSE AMENDMENT REQUEST DATED September 21, 1992
INSTRUMENTATION SPECIFICATION CHANGES

Northern States Power Company, a Minnesota corporation, requests authorization for changes to Appendix A of the Prairie Island Operating License as shown on the attachments labeled Attachments A, B, C and D. Attachment A describes the proposed changes, reasons for the changes, and a significant hazards evaluation. Attachments B, C and D are copies of the Prairie Island Technical Specifications incorporating the proposed changes.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By Roger O. Anderson
Roger O Anderson
Director
Licensing and Management Issues

On this 24 day of November 1993 before me a notary public in and for said County, personally appeared Roger O Anderson, Director, Licensing and Management Issues, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, that he knows the contents thereof, and that to the best of his knowledge, information, and belief the statements made in it are true and that it is not interposed for delay.

Judy L. Klapperick

JUDY L. KLAPPERICK
NOTARY PUBLIC-MINNESOTA
ANOKA COUNTY
My Commission Expires Sept. 29, 1997

Exhibit A

Prairie Island Nuclear Generating Plant
November 24, 1993 Revision to
License Amendment Request Dated September 21, 1992

Evaluation of Proposed Changes to the
Technical Specifications Appendix A of
Operating License DPR-42 and DPR-60

Pursuant to 10 CFR Part 50, Sections 50.59 and 50.90, the holders of Operating Licenses DPR-42 and DPR-60 hereby propose the following changes to Appendix A, Technical Specifications:

Background

a. History:

In response to growing concerns of the impact of current testing and maintenance requirements on plant operation, particularly as related to instrumentation systems, the Westinghouse Owners Group (WOG) initiated a program to develop a justification to be used to revise generic and plant specific instrumentation technical specifications. Operating plants experienced many inadvertent reactor trips and safeguards actuations during performance of instrumentation surveillance, causing unnecessary transients and challenges to safety systems. Significant time and effort on the part of the operating staff was devoted to performing, reviewing, documenting and tracking the various surveillance activities, which in many instances seemed unwarranted based on the high reliability of the equipment. Significant benefits for operating plants appeared to be achievable through revision of instrumentation test and maintenance requirements.

In their letter dated February 21, 1985 (Reference 1), the NRC issued the Safety Evaluation Report (SER) for WCAP-10271 and Supplement 1. The SER approved quarterly testing, 6 hours to place a failed channel in a tripped mode, increased Allowed Outage Time (AOT) for test and testing in bypass for analog channels of the Reactor Protection System (RPS). The quarterly testing had to be conducted on a staggered basis.

In their letter dated February 22, 1989 (Reference 2), the NRC issued the SER for WCAP-10271 Supplement 2 and Supplement 2, Revision 1. The SER approved quarterly testing, 6 hours to place a failed channel in a tripped mode, increased Allowed Outage Time for test and testing in bypass for analog channels of the Engineered Safety Features (ESF). The Engineered Safety Features functions approved in the SER were those presented in Appendix A1 of the reference WCAPs. These functions are all included in the Westinghouse Standard Technical Specifications. Staggered testing was not required for Engineered Safety Features analog channels and the requirement was removed from the Reactor Protection System analog channels.

In their letter dated April 30, 1990 (Reference 5), the NRC issued the Supplemental SER (SSER) for WCAP-10271 Supplement 2 and Supplement 2, Revision 1. The Supplemental SER approved Surveillance Test Interval (STI) and Allowed Outage Time extensions for the Engineered Safety Features functions that were included in Appendix A2 of WCAP-10271, Supplement 2, Revision 1. The functions approved are associated with the Safety Injection, Steam Line Isolation, Main Feedwater Isolation, and Auxiliary Feedwater Pump Start signals. The configurations contained in the Appendix A2 are those that are not contained in the Westinghouse Standard Technical Specifications.

With the issuance of the SER and the Supplemental SER, the relaxations for the analog channels of the Reactor Protection System and Engineered Safety Features are now the same and the special conditions applied to shared analog channels are no longer applicable.

To facilitate the incorporation of the revised Surveillance Test Intervals and Allowed Outage Times with appropriate ACTION requirements into the Prairie Island Technical Specifications, the Tables applicable to the instrumentation Technical Specifications have been reformatted to be consistent with the format of the Westinghouse Standard Technical Specifications and OPERATIONAL MODES have been defined. Where a requirement in the Prairie Island Instrumentation Tables is not included in Standard Technical Specifications Instrumentation specifications, and is adequately covered elsewhere in the Technical Specifications, the requirement is requested to be deleted from the Prairie Island Instrumentation Technical Specifications. Where a surveillance requirement is not adequately covered elsewhere, even when it is not a requirement in Standard Technical Specifications, it has been retained in Technical Specification Table TS.4.1-1C.

2. Hardware Modification.

No plant modifications are required to implement the items requested in this License Amendment Request. Increased allowed outage time and allowed testing in bypass mode will be accomplished with the present plant configuration. At present Prairie Island Nuclear Generating Plant does not have bypass testing capability for any of the analog instrumentation associated with the Reactor Protection System or Engineered Safety Features.

If in the future Prairie Island Nuclear Generating Plant does elect to test in bypass, plant modifications will be required. Any future bypass testing modification would be accomplished without reliance upon lifted leads or temporary jumpers and would provide bypass status indications to the plant operators in the control room.

Proposed Changes

This License Amendment Request proposes to revise the Prairie Island Technical Specifications and associated bases as described below. The specific wording changes to the Technical Specifications are shown in Exhibits B and C. Changes made to the proposed Technical Specification pages since the December 29, 1992 revision to this License Amendment Request are shown in Exhibit D.

1. Definitions 1.0

   a. Added new definitions to support new Instrumentation Specifications. Definitions are consistent with current industry Standard Technical Specification Revision 4a.

      - ACTION
      - OPERATIONAL MODE - MODE
      - STAGGERED TEST BASIS
      - New Table TS.1-1 with OPERATIONAL MODES

      The definition of DEGREE OF INSTRUMENTATION REDUNDANCY is being deleted because that term will no longer be utilized in the Prairie Island Technical Specifications.

   b. Replaced definitions for COLD SHUTDOWN, HOT SHUTDOWN, POWER OPERATION and REFUELING with new Table TS.1-1 "OPERATIONAL MODES". The proposed MODE table (Table TS.1-1) is consistent with the MODE table in the revised Westinghouse Standard Technical Specifications with the following exceptions:

      1) The titles for MODES 2, 3 and 4 are not consistent with the Revised Standard Technical Specifications. An asterisked statement noting this inconsistency is included in the proposed Table TS.1-1.

      2) The RATED THERMAL POWER conditions for MODES 1 and 2 are based on 2% rather than 5%.

      3) The status of the reactor vessel head closure bolts is specified in a separate column rather than as an asterisked statement.

      4) The reactivity conditions are specified using the terms "Critical" and "Subcritical" instead of the k_{eff} values used in the Revised Standard Technical Specifications.

   c. Associated with the elimination of the definition of HOT SHUTDOWN, the current SHUTDOWN MARGIN requirements of Technical Specification 3.10.A are being replaced with a new definition for SHUTDOWN MARGIN and new SHUTDOWN MARGIN requirements. The proposed SHUTDOWN MARGIN definition in Section 1.0 and the proposed SHUTDOWN MARGIN requirements in Section 3.10.A are consistent with the Revised Westinghouse Standard Technical Specifications except that portions of the Revised Standard Technical Specification SHUTDOWN MARGIN definition have been incorporated into the bases for Section 3.10.A rather than Section 1.0. In addition, the

bases for the shutdown margin requirements of Technical Specification Section 3.10 are being revised to support the revisions to Section 3.10.A. A note is also being incorporated into the Bases for Section 3.10 to clarify that shutdown margin is a function of hot full power boron concentration in Figure TS.3.10-1.

Changes to specification 3.10.A.1 and the Bases to Section 3.10 proposed by the December 29, 1992 revision to this License Amendment Request contained erroneous references to HOT STANDBY with $k_{eff} < 1.0$. Those references have been eliminated by this revision as shown in Exhibit D.

2. Technical Specification 2.3.A.2.g

   a. Technical Specification 2.3.A.2.g and the associated bases are being revised to clarify that the Technical Specification required reactor coolant pump bus undervoltage reactor trip is the direct undervoltage reactor trip, not the reactor coolant pump circuit breaker undervoltage trip which indirectly results in a reactor trip.

3. Technical Specification 3.5 and Table TS.3.5-2 through TS.3.5-6

   a. Technical Specification Section 3.5 is revised to refer to new Tables TS.3.5-2A and TS.3.5-2B. Parts C and D of specification 3.5 have been replaced by incorporating ACTIONS or notes into the new Tables as appropriate.

   b. Table TS.3.5-2A replaces old Table TS.3.5-2. The new Table is consistent with the format and content of Standard Technical Specifications Revision 4a and also incorporates the Allowed Outage Times approved in References 1 and 5.

      Actions 3.b and 5 from the proposed Table 3.5-2A have been clarified as shown in Exhibit D of this revision.

   c. Table TS.3.5-2B replaces old Tables TS.3.5-3, TS.3.5-4 (except Functional Unit 4), and TS.3.5-6. The new Tables are consistent with the format and content of Standard Technical Specifications Revision 4a and also incorporate the Allowed Outage Times approved in References 2 and 5.

      The following changes have been incorporated into the proposed Table 3.5-2B as shown in Exhibit D of this revision:

      1. The descriptions of Functional Units 1.c, 5.d, 6.c and 7.d have been clarified.

      2. The cross references "See 1 above..." have been revised to "See Functional Unit 1 above..." for clarity.

3. Action 26 has been revised to allow one channel of the main feedwater pump trip auxiliary feedwater pump actuation instrumentation to be inoperable for 72 hours rather than the 48 hours originally proposed. The 72 hour allowed out of service time is consistent with the out of service time allowed for the auxiliary feedwater pumps by specification 3.4.B.2.

4. The requirement to take the unit to INTERMEDIATE SHUTDOWN if the inoperable channel was not returned to service within the allowed out of service time was deleted from the proposed Action 26. It is only necessary to take the unit to MODE 3 (HOT SHUTDOWN) because the main feedwater pump trip auxiliary feedwater actuation instrumentation is only required for MODES 1 and 2.

5. A new Action (Action 34) was added for Functional Unit 7.a, the manual auxiliary feedwater pump actuation circuitry. This change is required because of the changes to Action 26 described above. The new Action 34 is consistent with the old Action 26 except that the allowed out of service time has been increased from 48 hours to 72 hours for the reasons discussed above.

6. A new Action (Action 30) was added for Functional Unit 7.f, the auxiliary feedwater pump actuation logic and actuation relays. The new Action 30 is consistent with the old Action 20 except that the allowed out of service time has been increased from 6 hours to 72 hours. The increase in allowed out of service time is based on the auxiliary feedwater actuation logic design. Each actuation logic channel only starts one auxiliary feedwater pump. Therefore, if one logic channel is inoperable, only one auxiliary feedwater pump is affected. There is no increased risk associated with allowing an auxiliary feedwater actuation logic channel to be out of service for 72 hours because the associated pump is allowed to out of service for 72 hours.

   The new Action 30 also only requires the unit to be taken to INTERMEDIATE SHUTDOWN if the inoperable channel is not returned to service within the allowed out of service time. It is only necessary to take the unit to MODE 4 (INTERMEDIATE SHUTDOWN) because the auxiliary feedwater actuation logic and actuation relays are only required to be operable in MODES 1, 2 and 3.

7. The MINIMUM CHANNELS OPERABLE requirements for Functional Units 8.a and b were revised to "3/Bus" to accurately reflect the system design.

8. Action 29 was replaced with new Actions 31, 32 and 33 in Functional Units 8.a and b. This was required because Action 29 was found to be inappropriate for the degraded voltage and undervoltage instrumentation installed as part of the recent station blackout modifications.

9. Actions 21 and 29 were revised to only allow one inoperable channel to be bypassed at a time for surveillance testing. In both these cases it was determined to be inappropriate to allow more than one channel to be bypassed at a time.

d. Functional Unit 10 of Table TS.3.5-2 specifies the requirements for the single loop and two loop loss of flow reactor trips the two trips are listed separately. In Functional Unit 12 of Tables 3.5-2A and TS.4.1-1A the loss of reactor coolant flow reactor trip is listed as a single item, with no reference to single loop or two loop trips.

e. Functional Unit 15 of Table TS.3.5-2 is deleted since the control rod misalignment monitor is not associated with the reactor protection system and because Technical Specification Section 3.10.I specifically addresses the actions to be taken if rod position deviation or quadrant power tilt monitors are inoperable.

f. Functional Unit 4 of Table TS.3.5-4 is deleted since this requirement is adequately addressed by the requirements of Technical Specification 3.4.C and the definition of operability.

The original License Amendment Request proposed inserting specific references to the steam exclusion system temperature sensors into specification 3.4.C. Those references have been eliminated by this revision to the License Amendment Request. It is unnecessary to make specific reference to instrumentation supporting the operability of a system in the LCO for that system. The consideration of the affect of the inoperability of an individual system component on the operability of the whole system is addressed by the operability definition.

g. Table TS.3.5-5 is deleted since this requirement is adequately addressed in specifications 3.6.F and 3.6.H. The original License Amendment Request proposed inserting a discussion of the operability of the Shield Building and Auxiliary Building Special Ventilation systems with respect to the operability of safety injection actuation and pressure difference inputs to the systems. into the Bases to Section 3.10 of the Technical Specifications. That discussion has been eliminated by this revision to the License Amendment Request. It is unnecessary to make specific reference to system actuation inputs and their effect on system operability into a Technical Specification Bases. The consideration of the affect of the inoperability of an actuation input on the operability of the whole system is addressed by the operability definition.

4. Technical Specifications 4.1.A, 4.1.D, and Table TS.4.1-1

a. Technical Specification 4.1.A is revised to refer to new Tables TS.4.1-1A through TS.4.1-1C.

b. Technical Specification 4.1.D is revised to delete the sentence about APPLICABILITY at all times. APPLICABILITY has been incorporated into the individual new Tables.

c. Tables TS.4.1-1A and TS.4.1-1B replace old Table TS.4.1-1 for Reactor Trip and Engineered Safety Features Surveillance Requirements. Those functions not related to Reactor Trip or Engineered Safety Features have been incorporated into a new Table TS.4.1-1C for miscellaneous instrumentation surveillance requirements. The new Tables are consistent with the format and content of Standard Technical Specifications Revision 4a. In addition, Tables TS.4.1-1A and TS.4.1-1B incorporate the Surveillance Frequencies approved in References 1, 2 and 5.

d. Specific surveillance requirements for the auxiliary feedwater system actuation instrumentation which are consistent with current requirements in the Prairie Island Technical Specifications or the Standard Technical Specifications have been incorporated into Functional Unit 7 in Table TS.4.1-1B.

e. Functional Unit 43 of Table TS.4.1-1 is deleted since the Control Room Ventilation System Chlorine Monitors are no longer required by the Prairie Island Technical Specifications.

f. The following changes have been incorporated into the proposed Table 4.1-1A as shown in Exhibit D of this revision:

   1) A requirement for a refueling frequency calibration of the Power Range Neutron Flux High Setpoint channels, which was mistakenly not included in the original submittal, has been incorporated into Functional Unit 2.a.

   2) The required MODE for Functional Unit 6.a has been corrected from MODE 1 to MODE 2. The source range neutron flux instrumentation is de-activated by the time the reactor reaches MODE 1.

   3) Note 19, which is not used at this time, was added to support the numbering changes of the notes in Tables 4.1-1B and 4.1-1C.

g. The following changes have been incorporated into the proposed Table 4.1-1B as shown in Exhibit D of this revision:

   1) The descriptions of Functional Units 1.c, 5.d, 6.c and 7.d have been clarified.

   2) The cross references "See 1 above..." have been revised to "See Functional Unit 1 above..." for clarity.

   3) The notes at the end of the table have been changed such that the numbering of the notes is unique for each of the 4.1-1 tables.

h. The following changes have been incorporated into the proposed Table 4.1-1C as shown in Exhibit D of this revision:

   1) The description of Functional Unit 9 has been clarified to make it clear that the specified surveillance is applicable to volume control tank level instrumentation. This reference to level was inadvertently dropped from this Functional Unit by License Amendments 61 and 55 dated February 2, 1983.

   2) The description of Functional Unit 19 has been clarified to make it clear that the specified surveillance is applicable to CRDM cooling shroud exhaust air temperature instrumentation. This reference to exhaust air temperature was inadvertently dropped from this Functional Unit by License Amendments 68 and 62 dated February 21, 1984.

   3) The functional test frequency for Functional Unit 15, Turbine First Stage Pressure, has been changed from Monthly to Quarterly to be consistent with the testing of the associated reactor protection system and engineered safety features instrumentation.

   4) The notes at the end of the table have been changed such that the numbering of the notes is unique for each of the 4.1-1 tables.

5. Change to the bases to insert the necessary wording for referencing the WCAP-10271 and supplements.

6. Editorial Technical Specification Changes

   a. Technical Specification 2.3.B.1 through 2.3.B.5 have added headings of the interlock names for clarity.

   b. Deleted footnote "See Specification 4.1.D" at the bottom of Table TS.4.1-2B page 2 of 2 and added a footnote "Required at all times" at the bottom of page 1 of 2 in this table.

   c. In addition to the changes to page TS.3.10-1 described above, the term "power operation" in the objective section of Technical Specification Section 3.10 is being fully capitalized because it is a defined term.

<u>Justification</u>

Increasing the Surveillance Test Interval for the Reactor Protection System and Engineered Safety Features instrumentation minimizes the potential number of inadvertent Engineered Safety Features actuations and reactor trips during surveillance testing. Less frequent surveillance testing has been estimated to result in 0.5 fewer inadvertent reactor trips, per unit, per year. Also, increasing the surveillance interval enhances the operational effectiveness of plant personnel. The amount of time plant personnel spend performing surveillance testing will be reduced. This allows manpower to be used for other tasks such as preventative maintenance. The increased Allowed Outage Time has been shown to result in fewer human factor errors, since more time is allowed to perform an action.

WCAP-10271 results show that the reduction in testing and the increase in testing and maintenance Allowed Outage Times do not adversely affect public health and safety. The proposed revision will reduce the number of inadvertent Engineered Safety Features actuations and reactor trips and allow Prairie Island to better manage resources to maintain the plant.

Reformatting the Tables in the instrumentation Technical Specifications to a format and content consistent with Revision 4a of the Westinghouse Standard Technical Specifications ensures implementation of the approved Allowed Outage Times and Surveillance Test Intervals in a manner consistent with the SERs and the Supplemental SER of References 1, 2 and 5. By defining ACTION, OPERATIONAL MODE-MODE and STAGGERED TEST BASIS, and creating Table TS.1-1 to define OPERATIONAL MODES, a consistent set of Action Statements, with Applicable MODE requirements can be established.

Definitions for COLD SHUTDOWN, POWER OPERATION, REFUELING and HOT SHUTDOWN would be deleted and replaced by the OPERATIONAL MODES defined in Table TS.1-1. The proposed MODE table (Table TS.1-1) is consistent with the MODE table in the revised Westinghouse Standard Technical Specifications with the following exceptions:

1) The titles for MODES 2, 3 and 4 are not consistent with the Revised Standard Technical Specifications. The proposed titles for MODES 2, 3 and 4 are "HOT STANDBY", "HOT SHUTDOWN" and "INTERMEDIATE SHUTDOWN" respectively. The Standard Technical Specification titles are not being used because the term "HOT SHUTDOWN" is used throughout the Prairie Island Technical Specifications. Use of the Standard Technical Specification title for MODE 3 (HOT STANDBY) would involve a significant revision to the Prairie Island Technical Specifications, which is beyond the scope of this amendment request. The title "HOT STANDBY" is used for MODE 2 because plant procedures define the MODE 2 conditions as HOT STANDBY and because the Standard Technical Specification title for MODE 2 is "STARTUP" which could be confused with the term "STARTUP OPERATIONS" currently defined in the Prairie Island Technical Specifications.

   An asterisked statement, noting this inconsistency, is included in the proposed Table TS.1-1.

2) The RATED THERMAL POWER conditions for MODES 1 and 2 are based on 2% rather than the 5% specified in the Revised Standard Technical Specifications. The 2% RATED THERMAL POWER conditions for MODES 1 and 2 is consistent with current Prairie Island Technical Specification and procedural definitions. Changing the MODES 1 and 2 RATED THERMAL POWER condition to 5% would provide no improvement in plant safety and could increase the possibility of human error.

3) The status of the reactor vessel head closure bolts is specified in a separate column to eliminate additional asterisked notes from the table and to more clearly define reactor head bolt requirements and the difference between the COLD SHUTDOWN and REFUELING MODES.

4) The HOT SHUTDOWN reactivity requirements, in the current definition, are based on the shutdown margin requirements in Table TS.3.10-1. Because of this past history of MODES being associated with shutdown margin, the reactivity conditions in the proposed MODE table are specified using the terms "Critical" and "Subcritical" instead of the k_{eff} values used in the Revised Standard Technical Specifications. The terms "Critical" and "Subcritical" meet the intent of the k_{eff} conditions specified in the Revised Standard Technical Specifications.

POWER OPERATION AND COLD SHUTDOWN, as defined in new Table TS.1-1, correspond to the definitions being deleted and therefore do not represent changes.

HOT STANDBY AND INTERMEDIATE SHUTDOWN and the conditions they represent are not currently defined in the Prairie Island Technical Specifications. However, the proposed MODES they represent (MODES 2 and 5) are consistent (except as discussed above) with the Standard Technical Specification MODES.

The definition of HOT SHUTDOWN in the new OPERATIONAL MODES table differs from the current definition in two respects, the reactivity conditions and the average reactor coolant temperature.

In the current definitions of HOT SHUTDOWN, reactivity requirements are based on the SHUTDOWN MARGIN requirements in Table TS.3.10-1. The definition of HOT SHUTDOWN per new Table TS.1-1 specifies reactivity conditions equivalent to those in the Westinghouse Standard Technical Specifications. Associated with these proposed changes, the current SHUTDOWN MARGIN requirements of Technical Specification 3.10.A are being replaced with a new definition for SHUTDOWN MARGIN and new SHUTDOWN MARGIN requirements. This change to Specification 3.10.A is being made to ensure there is no confusion between the HOT SHUTDOWN reactivity conditions and the requirements of Table TS.3.10-1 and to ensure that shutdown margin requirements exist for all plant modes.

The bases for the shutdown margin requirements of Technical Specification Section 3.10 are revised to support the revisions to Section 3.10.A and to further clarify the shutdown margin requirements. A note is also being incorporated into the Bases for Section 3.10 to clarify that shutdown margin is a function of hot full power boron concentration in Figure TS.3.10-1.

The new SHUTDOWN MARGIN requirements in Section 3.10.A and the new SHUTDOWN MARGIN definition in Section 1.0 are consistent with the revised Westinghouse Standard Technical Specifications except that portions of the revised Standard Technical Specification SHUTDOWN MARGIN definition have been incorporated into the bases for Section 3.10.A rather than Section 1.0. The portions of the Revised Standard Technical Specification SHUTDOWN MARGIN definition placed in the Section 3.10 Bases involve guidance on how to calculate SHUTDOWN MARGIN when in MODES 1 or 2 or with rods not fully inserted and as such do not specifically define SHUTDOWN MARGIN.

These changes to the definition of HOT SHUTDOWN, to Specification 3.10.A and the bases to Section 3.10 will expand shutdown margin requirements to the INTERMEDIATE and COLD SHUTDOWN conditions, but will have no affect on the actual shutdown margin limits in Figure TS.3.10.1.

The adoption of Standard Technical Specification OPERATIONAL MODES will result in the temperature conditions for HOT SHUTDOWN (MODE 3) changing from a specific temperature (547°F) to a range of temperatures (≥350°F). The addition of SHUTDOWN MARGIN requirements for temperatures below 547°F, as discussed above, will ensure that adequate SHUTDOWN MARGIN will be maintained for the full HOT SHUTDOWN (MODE 3) temperature range.

The definition of REFUELING currently includes a 140°F temperature condition. This temperature is not associated with any plant safety analysis and was eliminated from the OPERATIONAL MODE Table in the Revised Westinghouse Standard Technical Specifications. Therefore, the 140°F temperature limitation for the REFUELING MODE was not included in the proposed OPERATIONAL MODE Table. The propose REFUELING MODE definition is consistent with the Revised Westinghouse Standard Technical Specifications.

Technical Specification 2.3.A.2.g currently lists the reactor coolant pump circuit breaker undervoltage trip setpoint as a protective instrumentation setting for reactor trip. While the trip of a reactor coolant pump breaker as the result of this undervoltage instrumentation would indirectly result in a reactor trip because of the opening of the breaker, this is not the undervoltage trip utilized in the analysis of the loss of flow accident in the Updated Safety Analysis Report. Per Section 14.4.8.1 of the Updated Safety Analysis Report, the direct reactor coolant pump bus undervoltage reactor trip is used in the analysis of the reactor coolant system flow coastdown event. Because the direct reactor coolant pump bus undervoltage reactor trip is utilized in the plant safety analysis, Technical Specification 2.3.A.2.g is being revised to refer to that undervoltage trip rather than the reactor coolant pump breaker undervoltage trip currently referenced.

The loss of reactor coolant flow trip is being listed as a single item, with no reference to the number of loops, in Functional Unit 12 of Tables 3.5-2A and TS.4.1-1A, because the P-7 and P-8 interlocks, which enable the single loop and two loop loss of flow trips, have the same setpoint (>10% power). Because the single loop and two loop loss of flow trips are enabled at the same power level, and because these two trips utilize the same flow instrumentation, there is no need to list both trips separately. A reactor coolant system flow channel failure would affect both the single loop and the two loop loss of flow trips, and the actions taken in response to the failure would be the same. Listing only a single loss of reactor coolant flow functional unit in Table TS.3.5-2A will make the response to a loss of a reactor coolant flow channel less confusing and will simplify the Technical Specifications.

Functional Unit 15 of Table TS.3.5-2 is being deleted since the control rod misalignment monitor is not associated with the reactor protection system and because Technical Specification Section 3.10.I specifically addresses the actions to be taken if rod position deviation or quadrant power tilt monitors are inoperable.

Functional Unit 4 of Table TS.3.5-4 is not proposed to be incorporated into Table TS.3.5-2B since this requirement is adequately addressed by the requirements of Technical Specification 3.4.C and the definition of operability. Because there is only one temperature sensor per channel, the inoperability of a steam exclusion system temperature sensor would result, per the definition of operability, in the inoperability of the associated train of the steam exclusion system. Thus, the actions required by specification 3.4.C for the inoperability of the steam exclusion system are directly applicable to the inoperability of the associated temperature sensors. There is no need for separate Limiting Conditions for Operation and Action Statements in Section 3.5 for the steam exclusion temperature sensors. The proposed change increases the time the temperature sensors may be inoperable, but makes the action to be taken consistent with the current Technical Specification actions for the actuation logic and actuated components. These channels are not in the instrumentation Standard Technical Specifications.

Action statements for Functional Units 1 and 2 of Table TS.3.5-6 are being revised to provide more appropriate actions in response to failures of the degraded voltage and undervoltage channels. These functions actuate from a logic of 1 of 2 channels on one phase coincident with 1 of 2 channels on an alternate phase indicating below setpoint. Aside from initiating voltage restoration upon detection of low voltage, these functions must also provide a permissive for subsequent load restoration when voltage is reestablished. For a single channel failure (e.g., failed degraded voltage relay), placing the failed channel in bypass prevents a subsequent failure from preventing load restoration, yet still permits voltage restoration. For a two channel failure (e.g., failed potential transformer on one phase, or failed degraded voltage relay on each phase), placing one channel in the tripped condition ensures actuation will work in the case of two channels inoperable on a single phase. The other inoperable channel is placed in bypass such that a constant degraded or undervoltage actuation signal does not exist for a failure on each phase as this would result in voltage restoration but no load restoration. An added requirement that all channels associated with the redundant 4kV safeguards bus be operable was added for appropriateness. The time permitted to perform LCO actions was extended to 6 hours. These actions will need to be performed by an electrician, possibly during off-shift hours. Six hours provides time to safely perform the required actions and is consistent with similar times throughout the Section 3.5 action statements.

The Instrument Operating Conditions contained in the current Table TS.3.5-5 are deleted as these are adequately covered in the normal operability determination for the ventilation systems in Technical Specifications 3.6.F and 3.6.H. This does not represent a change in requirements. These channels are not in the instrumentation Standard Technical Specifications.

Adding headings to Technical Specification 2.3.B.1 through 2.3.B.5 is proposed for clarity and does not represent a change in requirements.

Deleting the footnote "See Specification 4.1.D" at the bottom of Table TS.4.1-2B page 2 of 2 and adding a footnote "Required at all times" at the bottom of page 1 of 2 in the same table does not represent a change in requirements. The original footnote on page 2 of 2 only applied to a function on page 1 of 2. Specification 4.1.D required the asterisked items in this table to be operable at all times. Changing the footnote as proposed therefore is only makes the existing requirement more readily apparent to the operators.

The surveillance requirements for the auxiliary feedwater system actuation instrumentation were incorporated into Table TS.4.1-1B for consistency with the Standard Technical Specifications. Those surveillance requirements for the auxiliary feedwater actuation instrumentation which are not consistent with the Standard Technical Specification requirements are consistent with current requirements in Prairie Island Technical Specification Section 4.8.

The functional test frequency for Functional Unit 15 of Table TS.4.1-1C, Turbine First Stage Pressure, has been changed from Monthly to Quarterly to be consistent with the testing of the associated reactor protection and engineered safety feature actuation instrumentation. The turbine first stage pressure instrumentation addressed by Functional Unit 15 provides permissive interlock input to the reactor protection and engineered safety feature actuation systems. This instrumentation is physically located in the same racks and is tested under the same procedures as the reactor protection and engineered safety feature actuation instrumentation. This turbine first stage pressure instrumentation was evaluated as part of the setpoint drift analysis completed for the reactor protection and engineered safety feature actuation instrumentation and was found acceptable.

Functional Unit 43 of Table TS.4.1-1 is deleted since the Control Room Ventilation System Chlorine Monitors are no longer required by the Prairie Island Technical Specifications. The Control Room Ventilation System Chlorine Monitors were eliminated from the Prairie Island Technical Specifications by License Amendments DPR-42/102 and DPR-60/95 dated September 29, 1992. The chlorine monitor surveillance requirements were mistakenly not included in the License Amendment request which requested the elimination of the chlorine monitors.

The changes to the bases are consistent with the NRC requirements included in the References 1, 2 and 5 and only add the applicable references for the revised Allowed Outage Times and Surveillance Test Intervals.

Safety Evaluation

In WCAP-10271 and its supplements, the Westinghouse Owners Group evaluated the impact of the proposed Surveillance Test Interval and Allowed Outage Time changes on core damage frequency and public risk. The NRC staff concluded in its evaluation (Reference 2) of the Westinghouse Owners Group evaluation that an overall upper bound of the core damage frequency increase due to the proposed Surveillance Test Interval/Allowed Outage Time changes is less than 6 percent for Westinghouse Pressurized Water Reactors (PWR) plants. The NRC Staff also concluded that actual core damage frequency increases for individual plants are expected to be substantially less than 6 percent. The NRC Staff considered this core damage frequency increase to be small compared to the range of uncertainty in the core damage frequency analyses and therefore acceptable.

Additionally the NRC Staff concluded that a staggered test strategy need not be implemented for Engineered Safety Features analog channel testing and is no longer required for Reactor Protection System analog channel testing. This conclusion was based on the small relative contribution of the analog channels to Reactor Protection System/Engineered Safety Features unavailability, process parameter signal diversity and normal operational testing sequencing.

The proposed changes in Surveillance Test Intervals and Allowed Outage Times are consistent with NRC Safety Evaluation Reports dated February 21, 1985 (Reference 1), February 22, 1989 (Reference 2), and April 30, 1990 (Reference 5) regarding WCAP-10271, WCAP-10271 Supplement 1, WCAP-10271 Supplement 2, and WCAP-10271 Supplement 2, Revision 1, (References 1, 2 and 5). The changes to make the Tables, the MODES and the ACTIONS in the Standard Technical Specification format are consistent with the assumptions used in the analyses of WCAP-10271 and the supplements. The SERs and the Supplemental SER therefore apply to the format and content changes as proposed. Where a Functional Unit in the current Prairie Island Technical Specifications is not included in the Standard Technical Specifications, it is retained in the new Table 4.1-1C. In the few cases where an existing Functional Unit has not been included in one of the new Tables, the requirement is retained elsewhere in the Technical Specifications and the safety function is maintained. The only exceptions are the surveillance requirements for the Control Room Ventilation System Chlorine Monitors which are no longer required by the Prairie Island Technical Specifications and have been removed from service.

The NRC Staff has stated that approval of the changes approved in their SERs is contingent upon confirmation that certain conditions are met. Although the Safety Evaluation Reports of References 2 and 5 apply to Engineered Safety Features instrumentation, conditions given in the Reference 1 SER for the Reactor Protection System instrumentation also apply to Engineered Safety Features where appropriate. The Prairie Island response to these conditions is provided below.

1. Reactor Protection System SER Conditions:

   a. SER Condition - NRC Staff stated in the Reactor Protection System SER (Reference 1, page 10) that approval of an increase in Surveillance Test Interval for the analog channel operational tests from once per month to once per quarter is contingent on performance of the testing on a staggered test basis. In the Engineered Safety Features SER (Reference 2, page 4 of enclosure 1) this requirement was removed.

      Response - This SER Condition is not a concern for Prairie Island as the changes proposed in this LAR implement Reactor Protection System and Engineered Safety Features at the same time. As the increase in Surveillance Test Interval for the analog channel operational tests from once per month to once per quarter with the contingency to perform the testing on a staggered test basis was not implemented for Reactor Protection System functions, it is not necessary to remove this requirement.

   b. SER Condition - NRC Staff stated in the Reactor Protection System SER (Reference 1, page 10) that approval of items related to extending Surveillance Test Intervals is contingent on procedures being in place to require evaluation of failures for common cause and to require additional testing if necessary.

      Response - Prairie Island has implemented procedures and procedural steps to evaluate failures for common cause and require additional testing as necessary in accordance with the Westinghouse Owners Group position given in "Westinghouse Owners Group Guidelines for Preparing Submittals Requesting Revision of Reactor Protection System Technical Specification, Revision 1". These guidelines were reviewed and approved by NRC Staff.

   c. SER Condition - NRC Staff stated in the Reactor Protection System SER (Reference 1, page 10) that for channels which provide dual inputs to other safety related systems such as Engineered Safety Features, the approval of items that extend Surveillance Test Intervals and Allowed Outage Times apply only to the Reactor Protection System function.

      Response - The Engineered Safety Features SER has been issued (References 2 and 5). The extensions approved for the Engineered Safety Features analog Channels are the same as the Reactor Protection System and so this SER Condition is not a concern for Prairie Island.

   d. SER Condition - NRC Staff stated in the Reactor Protection System SER (Reference 1, page 10) that approval of channel testing in a bypassed condition is contingent on the capability of the Reactor Protection System design to allow such testing without lifting leads or installing temporary jumpers.

      Response - At present Prairie Island does not have bypass testing capability for any of the analog instrumentation associated with the Reactor Protection System or Engineered Safety Features with the exception of the source range and intermediate range reactor trips.

If in the future Prairie Island does elect to test other channels in bypass, plant modifications will be required. Any future bypass testing modification would be accomplished without reliance upon lifted leads or temporary jumpers and will provide bypass status indications to the plant operators in the control room.

e. SER Condition - NRC Staff stated in the Reactor Protection System SER (Reference 1, page 9) that acceptance was contingent on confirmation that the instrument setpoint methodology includes sufficient margin to offset the drift anticipated as a result of less frequent surveillance.

Response - Prairie Island implemented a program to evaluate setpoint drift in accordance with the Westinghouse Owners Group position given in the "Westinghouse Owners Group Guidelines for Preparing Submittals Requesting Revision of Reactor Protection System Technical Specification, Revision 1". These guidelines were reviewed and approved by NRC Staff.

Prairie Island has determined that the values used in the setpoint methodology properly account for drift due to extended Surveillance Test Intervals.

2. Engineered Safety Features SER Conditions:

a. SER Condition - NRC Staff stated in the Engineered Safety Features SER (Reference 2, Table 1 of enclosure 1) that the licensee must confirm the applicability of the generic analyses to the plant.

Response - The generic analyses used in WCAP-10271 and Supplements is applicable to Prairie Island. Prairie Island uses the Foxboro H-Line Process Control System and the Westinghouse Relay Protection System for both the Engineered Safety Features and Reactor Protection System. Both of these systems were specifically modelled in the generic analyses. The Engineered Safety Features Functional Units implemented at Prairie Island are all addressed by the generic analyses.

b. SER Condition - NRC Staff stated in the Engineered Safety Features SER (Reference 2, Table 1 of enclosure 1) that the licensee must confirm that any increase in instrument drift due to the extended Surveillance Test Intervals is properly accounted for in the setpoint calculation methodology.

Response - Same as Reactor Protection System SER Condition e. above.

The changes being made to the definition of HOT SHUTDOWN separate shutdown margin from the plant mode definitions and make the definition of HOT SHUTDOWN more consistent with the Standard Technical Specifications. Because these proposed changes are consistent with the guidance in the Standard Technical Specifications and because the proposed changes to Section 3.10.A will ensure that shutdown margin will be maintained during HOT SHUTDOWN conditions, there will be no reduction in plant safety.

The change to Specification 3.10.A and associated bases will ensure there is no confusion between the HOT SHUTDOWN reactivity conditions and the requirements of Table TS.3.10-1. The changes to Specification 3.10.A and the associated bases will expand shutdown margin requirements to the INTERMEDIATE and COLD SHUTDOWN conditions consistent with the Westinghouse Standard Technical Specifications, and thus will result in more restrictive Technical Specification requirements. The proposed changes will have no affect on the actual shutdown margin limits in Figure TS.3.10.1. Because the proposed shutdown margin requirements are consistent with the guidance in the Westinghouse Standard Technical Specifications and are more restrictive than the current Technical Specification requirements, they will not reduce the plants margin of safety.

Technical Specification 2.3.A.2.g currently lists the reactor coolant pump circuit breaker undervoltage trip setpoint as a protective instrumentation setting for reactor trip. While the trip of a reactor coolant pump breaker as the result of this undervoltage instrumentation would indirectly result in a reactor trip because of the opening of the breaker, this is not the undervoltage trip utilized in the analysis of the loss of flow accident in the Updated Safety Analysis Report. The proposed change to Section 2.3.A.2.g will correct the Technical Specifications to reference the reactor coolant pump bus undervoltage trip utilized in the plant safety analysis and will thus help ensure that protective function is maintained operable.

In conclusion, Northern States Power believes there is reasonable assurance that the health and safety of the public will not be adversely affected by the proposed Technical Specification changes.

Determination of Significant Hazards Considerations

The proposed changes to the Operating License have been evaluated to determine whether they constitute a significant hazards consideration as required by 10 CFR Part 50, Section 50.91 using the standards provided in Section 50.92. This analysis is provided below:

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The determination that the results of the proposed change are within all acceptable criteria have been established in the SERs prepared for WCAP-10271, WCAP-10271 Supplement 1, WCAP-10271 Supplement 2 and WCAP-10271 Supplement 2, Revision 1 issued by References 1, 2 and 5. Implementation of the proposed changes is expected to result in an acceptable increase in total Reactor Protection and Engineered Safety Features Systems yearly unavailability. This increase, which is primarily due to less frequent surveillance, results in a increase of similar magnitude in the probability of an Anticipated Transient

Without Scram (ATWS) and in the probability of core melt resulting from an ATWS and also results in a small increase in core damage frequency (CD) due to Engineered Safety Features unavailability.

Implementation of the proposed changes is expected to result in a significant reduction in the probability of core melt from inadvertent reactor trips. This is a result of a reduction in the number of inadvertent reactor trips (0.5 fewer inadvertent reactor trips per unit per year) occurring during testing of Reactor Protection System instrumentation. This reduction is primarily attributable to less frequent surveillance.

The reduction in inadvertent core melt frequency is sufficiently large to counter the increase in ATWS core melt probability resulting in an overall reduction in total core melt probability.

The values determined by the Westinghouse Owners Group and presented in the WCAP for the increase in core damage frequency were verified by Brookhaven National Laboratory (BNL) as part of an audit and sensitivity analyses for the NRC Staff. Based on the small value of the increase compared to the range of uncertainty in the core damage frequency, the increase is considered acceptable.

The changes of an editorial nature, including the change to Standard Technical Specification format for the instrumentation Technical Specifications and mode definitions, have no impact on the severity or consequences of an accident previously evaluated.

The proposed changes do not result in an increase in the severity or consequences of an accident previously evaluated. Implementation of the proposed changes affects the probability of failure of the Reactor Protection System and Engineered Safety Features but does not alter the manner in which protection is afforded nor the manner in which limiting criteria are established.

2. The proposed amendment will not create the possibility of a new or <u>different kind of accident from any accident previously analyzed.</u>

The proposed changes do not involve hardware changes and do not result in a change in the manner in which the Reactor Protection System and Engineered Safety Features provide plant protection. No change is being made which alters the functioning of the Reactor Protection System or Engineered Safety Features. Rather the likelihood or probability of the Reactor Protection System or Engineered Safety Features functioning properly is affected as described above. Therefore the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The changes of an editorial nature, including the change to Standard Technical Specification format for the instrumentation Technical Specifications and mode definitions does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

The proposed changes do not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operation are determined. The impact of reduced testing other than as addressed above is to allow a longer time interval over which instrument uncertainties (e.g., drift) may act. Experience has shown that the initial uncertainty assumptions are valid for reduced testing.

Implementation of the proposed changes is expected to result in an overall improvement in safety by:

a. Less frequent testing will result in less inadvertent reactor trips and actuation of Engineered Safety Features components.

b. Higher quality repairs leading to improved equipment reliability due to longer repair times.

c. Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation. This is due to less frequent distraction of the operator and shift supervisor to attend to instrumentation testing.

The changes of an editorial nature, including the change to Standard Technical Specification format for the instrumentation Technical Specifications and mode definitions does not lead to a reduction in any margin of safety.

Based on the evaluation described above, and pursuant to 10 CFR Part 50, Section 50.91, Northern States Power Company has determined that operation of the Prairie Island Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by NRC regulations in 10 CFR Part 50, Section 50.92.

Environmental Assessment

Northern States Power has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration,

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR Part 51, Section 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51, Section 51.22(b), an environmental assessment of the proposed changes is not required.

References

1. Letter from C. O. Thomas (NRC) to J. J. Sheppard (WOG) dated February 21, 1985 - "Safety Evaluation by the Office of Nuclear Reactor Regulation WCAP-10271, Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System".

2. Letter from Charles E. Rossi (NRC) to Roger A. Newton (WOG) dated February 22, 1989 - "Safety Evaluation by the Office of Nuclear Reactor Regulation Review of Westinghouse Report WCAP-10271 Supplement 2 and WCAP-10271 Supplement 2, Revision 1 on Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features".

3. WCAP-10271 Supplement 1-P-A, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System", May 1986.

4. WCAP-10271-P-A Supplement 2, Revision 1 "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features", May 1989.

5. Letter Charles E. Rossi (NRC) to Gerard T. Goering (WOG) dated April 30, 1990 (NRC Supplemental Safety Evaluation for WCAP-10271 Supplement 2, Revision 1).