UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

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

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

LICENSE AMENDMENT REQUEST DATED September 21, 1992

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 Exhibits A, B, and C. Exhibit A describes the proposed changes, reasons for the changes, and a significant hazards evaluation. Exhibits B and C 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 Thomas M Parker
Thomas M Parker
Manager
Nuclear Support Services

On this 21st day of September 1992 before me a notary public in and for said County, personally appeared Thomas M Parker, Manager Nuclear Support Services, 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
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:

1. Instrumentation Specification Changes

<u>Background</u>

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.

1. Definitions 1.0

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

      o ACTION

      o OPERATIONAL MODE - MODE

      o STAGGERED TEST BASIS

      o New Table TS.1-1 with OPERATIONAL MODES

   b. Replaced definitions for COLD SHUTDOWN, HOT SHUTDOWN, and REFUELING with new Table TS.1-1 "OPERATIONAL MODES".

   c. The HOT SHUTDOWN reactivity requirements, in the current definition, 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. A note is being included in the new MODE definition table (Table TS.1-1) to clarify that the HOT SHUTDOWN reactivity conditions are not a shutdown margin requirement and that the shutdown margin requirements are specified in Table TS.3.10-1.

      As the result of the proposed changes to the definition of HOT SHUTDOWN, Technical Specification 3.10.A is being revised to clarify that shutdown margin is based only on HOT SHUTDOWN temperature conditions. In addition, the bases for the shutdown margin requirements of Technical Specification Section 3.10 are being replaced with wording consistent with the current Westinghouse Standard Technical Specifications.

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.

   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.

   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 revising specification 3.4.C to specifically state that the actuation logic includes the temperature sensors.

   g. Technical Specification 3.4.C is revised to include the actuation instrumentation that was previously addressed in Table TS.3 5-4 Functional Unit 4. This increases the time that the temperature sensors may be inoperable to be consistent with the time that the actuated components are allowed to be inoperable.

   h. Table TS.3.5-5 is deleted since this requirement is adequately addressed in specifications 3.6.F and 3.6.H.

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.

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.

Justification

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 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, REFUELING and HOT SHUTDOWN would be deleted and replaced by the OPERATIONAL MODES defined in Table TS.1-1. COLD SHUTDOWN and REFUELING as defined in new Table TS.1-1 correspond to the definitions being deleted and therefore do not represent changes.

The definition of HOT SHUTDOWN the new Table TS.1-1 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. This change is being made to separate shutdown margin requirements from the plant mode definitions and to make the definition of HOT SHUTDOWN more consistent with the Standard Technical Specifications. The 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. The bases for Technical Specification Section 3.10 are being revised to clarify the shutdown margin requirements. These changes to the definition of HOT SHUTDOWN, to specification 3.10.A and the bases to Section 3.10 will have no affect on the actual shutdown margin requirements.

The change in the HOT SHUTDOWN average reactor coolant temperature requirements is discussed later in this exhibit under a separate evaluation.

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 as it is more appropriate to add it to Technical Specification 3.4.C. This increases the time the temperature sensors may be inoperable and makes the action to be taken consistent with actions for the actuation logic and actuated components. In the current Technical Specification, if the temperature sensor is inoperable, the plant must be taken to HOT SHUTDOWN and then to COLD SHUTDOWN if the minimum conditions are not met in 24 hours. The actions to be taken if either the actuation logic or actuated dampers are inoperable is only to close the associated dampers. Provided the dampers are closed, plant shutdown or MODE changes are not required. The proposed change is justified because there is only one temperature sensor per channel and because the change makes the action for the sensors consistent with the actions for the actuation logic that takes its input from the sensors, and the action for the actuated component. These channels are not in the instrumentation Standard Technical Specifications.

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 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 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. 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. These changes to the definition of HOT SHUTDOWN, specification 3.10.A and associated bases, will have no affect on the actual shutdown margin requirements in the Prairie Island Technical Specifications, and therefore they will not effect plant 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 (CDF) 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.

2. Change in HOT SHUTDOWN Average Coolant Temperature Requirements

Background

The current definition for HOT SHUTDOWN specifies that a reactor is in the HOT SHUTDOWN condition when the reactor coolant average temperature is 547°F or greater. Under normal conditions, the steam dump system controls the reactor coolant average temperature to a setpoint of 547°F during HOT SHUTDOWN conditions. As with any temperature control system the temperature cannot be controlled exactly at the setpoint, the actual temperature varies within a limited band around the setpoint.

The way the current definition for HOT SHUTDOWN is worded there is no allowance for average coolant temperature to fall below 547°. Because of this deficiency in the HOT SHUTDOWN definition, and the nature of the steam dump system, it is impossible to maintain the average coolant temperature within the requirements of the HOT SHUTDOWN definition at all times. This is especially true immediately following a reactor trip when average coolant temperature can drop several degrees below 547°F while the steam dump system responds to return temperature to 547°F.

Proposed Changes

The current temperature limit for HOT SHUTDOWN of ≥547°F is being lowered to ≥535°F in the proposed Operational Modes table (Table TS.1-1) described in the previous evaluation. The specific wording changes to Technical Specifications are shown in Exhibits B and C.

Justification

The change to the HOT SHUTDOWN definition described above will lower the HOT SHUTDOWN average coolant temperature requirement to ≥535°F, well below the 547°F setpoint of the steam dump system. This will help ensure that average coolant temperature can be maintained within the Technical Specification requirements for HOT SHUTDOWN.

Safety Evaluation

The proposed change in the temperature requirements for HOT SHUTDOWN were evaluated with respect to the effect on the plant safety analysis and reactor shutdown margin during HOT SHUTDOWN conditions.

The Northern States Power reload safety evaluation methodology lists the required transients that need to be addressed for each reload safety evaluation. Of the various transients listed, only the following four were impacted by the reduction of the HOT SHUTDOWN average coolant temperature to ≥535°F:

1. Uncontrolled RCCA Withdrawal from Subcritical
2. Startup of an Inactive Coolant Loop
3. Main Steam Line Break
4. Ejected Rod

The evaluation of these four transients concluded that the plant safety analysis remained bounding at HOT SHUTDOWN with the average coolant temperature at ≥535°F.

The effect of the reduction in the HOT SHUTDOWN average coolant temperature requirements on shutdown margin was evaluated for the most limiting transients, main steam line break and ejected rod. The evaluation of shutdown margin for these two transients concluded that the Technical Specification shutdown margin requirements were adequate during both transients with the reactor at HOT SHUTDOWN and the average coolant temperature at 535°F.

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 reduction of the HOT SHUTDOWN average coolant temperature requirements.

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 proposed change in the temperature requirements for HOT SHUTDOWN were evaluated with respect to the effect on the plant safety analysis and reactor shutdown margin during HOT SHUTDOWN conditions.

   This evaluation concluded that the plant safety analysis remained bounding and the Technical Specification shutdown margin requirements were adequate at HOT SHUTDOWN with the average coolant temperature at ≥535°F.

   The proposed changes only effect the HOT SHUTDOWN average coolant temperature requirements. They do not involve any modification of plant equipment and thus will not effect the probability of an accident.

   Therefore, the proposed changes will not significantly affect the probability or consequences of an accident previously evaluated.

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

   There are no new failure modes or mechanisms associated with the proposed changes. The proposed changes do not involve any modification of plant equipment. The proposed changes will only affect the average coolant temperature maintained during HOT SHUTDOWN conditions. No new or different kind of accident can result from a change in the HOT SHUTDOWN average coolant temperature requirements.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated, and the accident analyses presented in the Updated Safety Analysis Report will remain bounding

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

   The proposed change in the temperature requirements for HOT SHUTDOWN were evaluated with respect to the effect on the plant safety analysis and reactor shutdown margin during HOT SHUTDOWN conditions.

   This evaluation concluded that the plant safety analysis remained bounding and the Technical Specification shutdown margin requirements were adequate at HOT SHUTDOWN with the average coolant temperature at ≥535°F.

   Therefore, the proposed changes will not result in any reduction in the plant's 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).