November 4, 2004

Mr. Joseph M. Solymossy
Site Vice-President
Prairie Island Nuclear Generating Plant
Nuclear Management Company
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - RESPONSE TO COMMENTS ON THE LICENSE AMENDMENT AND SAFETY EVALUATION ISSUED ON APRIL 28, 2004 (TAC NOS. MC3676 AND MC3677)

Dear Mr. Solymossy:

By letter dated June 28, 2004, the Nuclear Management Company, LLC (NMC) provided comments on License Amendment Nos. 162 and 153 and the associated safety evaluation (SE) issued on April 28, 2004, for the Prairie Island Nuclear Generating Plant. The U.S. Nuclear Regulatory Commission has reviewed those comments and provides the following responses, which were discussed with your representatives during a teleconference on September 23, 2004. The affected pages of SE for Amendments 162 and 153 issued on April 28, 2004, are included as an enclosure.

1. Comments 1, 3, 5, 6, 9, 11 and 12 - These comments are editorial in nature and do not affect the technical content of our SE. These changes have been incorporated in the SE.

2. Comment 2 - This comment indicated that NMC proposed to "increase" pressurizer pressure-low Allowable Value rather than "decrease," as indicated in the second paragraph on SE page 3.

   The SE has been revised to reflect the change from "a decrease" to "an increase" for pressurizer pressure-low Allowable Value (AV) to be consistent throughout the SE. The use of word "decrease" in the background information in Section 3.0 of the original SE referred to a decrease in the allowance between the actual plant setting (APS) and the AV from 140 psig to 55 psig. The original AV was set at 1760 psig and the new setting for AV is 1845 psig. The value of APS is 1900 psig and was not changed. This change does not affect the technical basis of our SE, since our evaluation (as documented in SE Section 3.3.6) was based on the NMC's proposed pressurizer pressure-low AV and the results of the measurement uncertainty analysis.

3. Comment 4 - This comment indicated that under "LOFTRAN" on SE page 9, "rod drop" should be "dropped rod."

SE Page 9 under "LOFTRAN" stated that "...[t]he licensee proposed to use LOFTRAN to analyze the rod drop event..." Our use of the terminology, "the rod drop event," is consistent with that used in Table RAI1-11 of the NMC's response documented in Attachment 6 to NMC's letter dated January 14, 2004. The last paragraph in Table RAI1-11 (page 17 of 17) stated that "[t]he LOFTRAN code was only used in the analysis of the Rod Drop [underline added] transient (USAR 14.4.3) for Prairie Island..." The same table also includes "Rod Drop" in the list of transients for which LOFTRAN can be used. Therefore, the terminology, "rod drop," used in our SE remains unchanged.

4. Comment 7 - The comment stated that, Safety Evaluation, page 14 under "RCP Coastdown Delay Times," references General Design Criteria 10 and 15. In accordance with the Updated Safety Evaluation Report, the Prairie Island Nuclear Generating Plant was designed and constructed to comply with the intent of the Atomic Energy Commission General Design Criteria for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967.

   The SE has been changed to replace "the GDC 10 and 15 requirements" with "the requirements of USAR Sections 3.1.2.1 and 4.3.1.2.1" on SE page 14. The change is consistent with that discussed in SE Section 2.0, "REGULATORY EVALUATION," and Section 4.0, "SUMMARY." Both SE sections indicated that the review of the results of non-LOCA analyses for Prairie Island was based on the requirements of USAR Sections 3.1.2.1 and 4.3.1.2.1. Therefore, the change does not affect the technical basis and conclusions in our SE.

5. Comment 8 - This comment stated that, Safety Evaluation, page 14, Section 3.2.1, "ATWS Analysis," although the Safety Evaluation recognizes that the AMSAC/DSS was credited in the "ATWS" (Anticipated Transient Without Scram) analyses, it should be clarified that these are not the classical ATWS analyses. Specifically, ATWS is not included in the list of transients presented in the RETRAN topical report Safety Evaluation for which RETRAN can be used. Safety Evaluation Reference 5 does not refer to "ATWS" events, but rather "AMSAC/DSS" events.

   SE page 14, Section 3.2.1, "ATWS Analysis," discusses the evaluation of the results of the analysis of ATWS transients that credited the AMSAC/DSS for mitigation of consequences. Our evaluation and use of the terminology, "ATWS analysis," are consistent with the information provided in the NMC's response documented in Attachment 5 to NMC's letter dated January 14, 2004. Specifically, the fourth paragraph on page 5-272 of the NMC's submittal stated that, "[t]he analyses performed by NSP demonstrated that the DSS provides protection for the ATWS events [underline added], such that all of the applicable acceptance criteria are met. As part of the Safety Analysis Transition Program for Prairie Island, Westinghouse has reanalyzed several of these ATWS events [underline added] to reconfirm that the DSS provides adequate protection..." Therefore, SE page 14 for the ATWS analysis remains unchanged.

6. Comment 10 - This comment stated that, Safety Evaluation, page 16, middle of top paragraph, should say "... in Reference 24 that the least negative value of MTC [Moderator Temperature Coefficient]..." not the "most" negative. The least negative MTC is conservative for ATWS.

SE page 16, middle of top paragraph stated that, "...[i]n response, the licensee indicated in Reference 24 that the most negative value of MTC used in the analysis was -4.2 pcm/ EF, which bounded 95 percent of those for a representative cycle..." The SE statement is consistent with the information provided on page 5 of Attachment 1 to NMC's letter dated February 23, 2004. NMC stated on page 5 that, "[t]he most [underline added] negative (least conservative) MTC assumed in the AMSAC/DSS analyses of Section 5.1.15 (Attachment 5) is -4.2 pcm/ EF, which, as indicated on page 5-319, is bounding for 95 percent of a representative fuel cycle. Transient-specific MTCs assumed are as followed:

- Partial Loss of Reactor Coolant Flow (Section 5.1.15.3), 0.0 pcm/ EF,
- Loss of Normal Feedwater (Section 5.1.15.4), -2.0 pcm/ EF,
- Loss of AC Power (Section 5.1.15.5), -2.0 pcm/ EF,
- Loss of Load/Turbine Trip (Section 5.1.15.6), -4.1 pcm/ EF,
- RCCA Bank Withdrawal at Power (Section 5.1.15.7), -2.0 pcm/ EF,
- Uncontrolled Boron Dilution (Section 5.1.15.8), -4.2 pcm/ EF."

NMC indicated that the least negative MTC is conservative for the analysis of ATWS transients. However, its analysis (referenced above) also indicated that the most negative MTC of -4.2 pcm/EF used in the analysis bounded 95 percent of a representative cycle. The staff position on the MTC value used in the analysis is provided in the last sentence of the first paragraph of SE page 16. The SE indicated that the NRC staff found that this 95 percent probability level for the captured cycle time is equivalent to the probability level to assure that UET (unfavorable exposure time) will not be greater than 5 percent, and determined that it is acceptable. Therefore, SE page 16 remains unchanged.

If you have any further questions regarding our response to your comments, please contact me at (301) 415-8371.

Sincerely,

/RA/

Mahesh L. Chawla, Project Manager, Section 1
Project Directorate III
Division Of Licensing Project Management
Office Of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures: 1. Summary of changes
2. Affected pages of safety evaluation

SE page 16, middle of top paragraph stated that,"...[i]n response, the licensee indicated in Reference 24 that the most negative value of MTC used in the analysis was -4.2 pcm/ EF, which bounded 95 percent of those for a representative cycle..." The SE statement is consistent with the information provided on page 5 of Attachment 1 to NMC's letter dated February 23, 2004. NMC stated on page 5 that, "[t]he most [underline added] negative (least conservative) MTC assumed in the AMSAC/DSS analyses of Section 5.1.15 (Attachment 5) is -4.2 pcm/ EF, which, as indicated on page 5-319, is bounding for 95 percent of a representative fuel cycle. Transient-specific MTCs assumed are as followed:

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ADAMS ACCESSION NO: ML042950224

OFFICE	PDIII-1/PM	PDIII-1/LA	SC:SRXB	PDIII-1/SC
NAME	MChawla	THarris	JUhle	LRaghavan
DATE	10/28/04	10/28/04	11/02/04	11/04/04

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Summary of Changes

1. Attachment to License Amendment: add “2.0-2" to the list for the pages that were removed from the TSs for Units 1 and 2.

2. Second paragraph on SE page 3: change “a decrease ...Allowable Value” to “an increase ....Allowable Value.”

3. Second paragraph on SE page 9: change “pressurize-” to “pressurized-.”

4. Second paragraph on SE page 10: change “PHONIX-P” to “PHOENIX-P.”

5. Fourth paragraph on SE page 10: change “Experiment ... at low...” to “Experimental data from a European country indicated that failure of high burnup fuels occurs at low...”

6. Third paragraph on SE page 14 : change “the GDC 10 and 15 requirements” to “the requirements of USAR Sections 3.1.2.1 and 4.3.1.2.1.”

7. The third bulleted item on SE page 15: change “the AMSAC/DSS setpoint” to “the time that the AMSAC/DSS setpoint.”

8. The fourth bulleted item on SE page 15: change “AMSAC/DSS setpoint” to “the AMSAC/DSS setpoint.”

9. Page 19, third paragraph, next to last line and Page 21, third paragraph of SE: change “PING” to “PINGP”

10. Table 1 on SE page 22 - Loss of Reactor Coolant Flow: change “>1.607" to “1.607" and “>1.333" to “1.333.”
Table 1 on SE page 23 - Zero Power: change “WRB-1" to “W-3.”
Table 1 on SE Page 23 - RCCA Ejection: change “<168.6" to ”#168.6.”

ATTACHMENT TO LICENSE AMENDMENT NO. 162

FACILITY OPERATING LICENSE NO. DPR-42

DOCKET NO. 50-282

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

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

ATTACHMENT TO LICENSE AMENDMENT NO. 153

FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NO. 50-306

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

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

3.0 TECHNICAL EVALUATION

Background

In a March 25, 2003, letter (Ref. 1), supplemented by letters of June 16, 2003 (Ref. 2), January 14, 2004 (Ref. 3) , February 11 (Ref.24), February 23, 2004 (Ref. 25), and April 7, 2004 (Ref.26), NMC, the licensee for the PINGP, Units 1 and 2, proposed a license amendment request for approval of conversion to Westinghouse non-LOCA methodologies and related TS changes.

NMC has performed non-LOCA transient analyses that support operation of the PINGP with its own methods for many years. It plans to rely on Westinghouse methodologies to perform non-LOCA transient analyses in the future and, therefore, requested the approval of use of Westinghouse methodologies for non-LOCA analyses in this amendment request. In the process of converting to Westinghouse safety analyses, NMC also proposed TS changes in the following areas: (1) implementation of relaxed axial offset control (RAOC) of the reactor core; (2) implementation of Westinghouse methodologies for determining selected core operating parameter values; (3) relocation of selected operating parameters to the COLR; and (4) an increase of the pressurizer pressure-low Allowable Value. The TSs involved are: TS 2.1.1, “Reactor Core SLs;” TS 3.1.8, “Physics Tests Exceptions - Mode 2;” TS 3.2.1, “Heat Flux Hot Channel Factor $F_Q(Z)$;” TS 3.2.3, “Axial Flux Difference (AFD);” TS Table 3.3.1-1, “Reactor Trip System Instrumentation;” and TS 5.6.5, “Core Operating Limits Report (COLR).”

The proposed TS changes would reflect conversion to Westinghouse non-LOCA methodologies used in transient analyses, increase plant availability and operating flexibility.

On July 26, 2002, the NRC approved the conversion of the PINGP, Units 1 and 2, TS to the Improved Technical Specifications (ITS). In the safety evaluation report (SER) for the ITS conversion, Section G.2.1, “Instrument Setpoint Methodology and New Allowable Value,” specifically addresses the PINGP instrument setpoint methodology.

The licensee submitted Revision 0 of Section 3.3.4.1 of the Engineering Manual, which is the Engineering Design Standard for Instrument Setpoint/Uncertainty Calculations and is used to calculate instrument allowable values. The NRC staff reviewed and accepted licensee’s setpoint methodology based on regulatory requirements and guidance. The calculations submitted in this license amendment request (LAR) are performed in accordance with this methodology.

Review Scope

In support of the request for approval of conversion to Westinghouse methods for non-LOCA analyses, the licensee provided information in References 1 and 2 for the NRC staff to review. References 1 and 2 provide a description of the proposed TS changes with associated supporting justification, as well as reference WCAP Topical Reports (TRs) documenting Westinghouse methods that will be used in the non-LOCA analysis for the PINGP. The licensee indicated that the Westinghouse TRs were previously approved by the NRC and claimed that the TRs are acceptable for referencing in the specific plant licensing application. However, the staff found that additional information was needed to justify the acceptability of the TRs to the specific plant.

CHF correlations with use of VIPRE-01 were previously approved (Ref. 22) by the NRC for Westinghouse plants, and the PINGP specific application of the CHF correlations (discussed in Section 3.2 below) are within the applicable ranges of the approved correlations, the NRC staff concludes that the use of the correlations with the associated safety DNBR limits is acceptable.

RETRAN-02: This code simulates a multi-loop system using a model containing a reactor vessel, hot and cold-leg piping, SGs and pressurizer. The code also includes point kinetics and reactivity effects of the moderator, fuel, boron, and control rods. The secondary side of the SG uses a homogeneous, saturated mixture for analyses of thermal transients and water level responses for indication and control. As documented in WCAP-14882-P-A, the code has been generically approved (Ref. 23) by the NRC for Westinghouse to analyze system responses to non-LOCA transients for Westinghouse pressurized-water reactors. The licensee proposed (Ref. 6) to use RETRAN-02 to perform analyses of the following events: (1) uncontrolled rod cluster control assembly (RCCA) withdrawal at power; (2) excessive heat removal due to feedwater system malfunctions; (3) loss of reactor coolant flow; (4) locked pump rotor; (5) loss of external electrical load; (6) loss of normal feedwater; (7) loss of all AC power to the station auxiliaries; and (8) rupture of a steam pipe. The NRC staff found that: (1) each event proposed to be analyzed for the PINGP using RETRAN-02 is listed in Table 1 of the NRC staff's SE approving RETRAN-02; (2) the code is reliable in calculating the system responses during transients; and (3) the results of the transient analysis (discussed in Section 3.2 below) meet the acceptance criteria for the analysis of record. The NRC staff, therefore, concludes that the licensee's use of RETRAN-02 in performing non-LOCA analysis for the PINGP is acceptable.

LOFTRAN: The LOFTRAN code provides a simulation of the RCS response and calculates system parameters such as core power, RCS flow, RCA primary and secondary pressures and temperatures. The code was previously approved (Ref. 15) for Westinghouse plants for use in performing the design-basis non-LOCA transients. The licensee proposed to use LOFTRAN to analyze the rod drop event. The NRC staff found that the event proposed to be analyzed using LOFTRAN is listed in the NRC's SE approving LOFTRAN, and the results of the analysis (discussed in Section 3.2 below) show that the acceptance criteria for the analysis of record are met. The NRC staff, therefore, concludes that the licensee's use of LOFTRAN in performing the rod drop event analysis for the PINGP is acceptable.

TWINKLE: This multi-dimensional spatial neutronics code uses an implicit finite-difference method to solve the two group transient neutronics equations in one, two, and three dimensions. This code is documented in WCAP-7979-P-A (Ref. 17). The licensee proposed to apply TWINKLE to the PINGP for analysis of the uncontrolled RCCA withdrawal from a subcritical condition and the RCCA ejection event. The NRC staff found that the TWINKLE code was previously approved by NRC for Westinghouse plants in calculating the kinetics response of a reactor for transients such as the uncontrolled RCCA withdrawal from a subcritical condition and the RCCA ejection event. Since the TWINKLE code is a generic neutron kinetics code approved by the NRC, and the licensee applied TWINKLE using the nuclear characteristics specific to the PINGP fuel for two events (Ref. 5), the application of the TWINKLE code for the proposed use is acceptable.

FACTRAN: As documented in WCAP-7908-P-A (Ref. 16), the NRC has approved the code for Westinghouse plants in calculating the transient heat flux at the surface of a rod. Since FACTRAN is an NRC-approved code for calculating thermal transients in a fuel rod and the licensee complied (Ref. 6) with the SE restrictions (related to initial fuel temperatures, the gap heat transfer coefficient and number of concentric rings used for a fuel rod, for example) imposed on use of FACTRAN, the licensee's application of FACTRAN is acceptable.

PHOENIX-P and ANC: Both codes address three-dimensional features of the nuclear characteristics of the fuel. PHOENIX-P is used to generate the cycle-specific nuclear cross sections. ANC with the input from PHOENIX-P is used to calculate the nuclear characteristics such as power distributions, control rod worth, and reactivity feedback coefficients. As documented in WCAP-11596-P-A (Ref. 20) and WCAP-10965-P-A (Ref. 18), the NRC has approved both PHOENIX-P and ANC for Westinghouse plants in calculating the nuclear characteristics. Since the codes were approved by the NRC and the licensee applied the codes using fuel design characteristics specific to the PINGP, the licensee's application of the codes is acceptable.

Method for the Rod Ejection Analysis: As documented in WCAP-7588 Rev. 1-A (Ref. 14), the NRC has approved the method which relies on spatial kinetics models for the Westinghouse plants to perform the rod ejection analysis. Since the methods were approved by the NRC and the licensee applied the method using plant conditions specific to the PINGP, the licensee's application of the method is acceptable.

The rod ejection analysis method limits the calculated radial average fuel enthalpy to below 200 cal/gm, which is less than the acceptance criterion of 280 cal/gm specified in RG 1.77. Experimental data from a European country indicated that failure of high burnup fuels occurs at lower values of enthalpy than the limits specified in RG 1.77. However, generic analyses performed by the industry that assumed low enthalpy for fuel failure showed that the radiological consequences of the rod ejection event meet the acceptance criteria of Standard Review Plan (SRP) 15.4.8 (Appendix A). The generic analyses are predicated on conservative treatment of experimental fuel data applied to existing and planned cores within approved burnup limits for pessurized-water reactors (PWRs). In addition, there is a broad agreement among the NRC, the industry, and the international community, that burnup degradation in the margin for low-enthalpy fuel failure is likely to be regained by application of more detailed 3-dimensional analysis methods of the fuel response to rod ejection events. Therefore, the NRC staff concludes that although the RG 1.77 fuel failure enthalpy limit may not be conservative, the generic analyses provide reasonable assurance that radiological consequences of the rod ejection event will not violate the acceptance criterion in SRP Section 15.4.8 for the PWR cores operating within the current NRC-approved burnup limits. The NRC staff will not approve further extension of burnup limits until additional experimental information on fuel behavior is available to demonstrate that the fuel cladding will satisfy the regulatory acceptance criteria used in the rod ejection analysis for licensing applications.

Models in two Westinghouse TRs: WCAP-11394-P-A (Ref. 19) documents the methodology for analysis of the dropped rod event, and WCAP-12910 Rev. 1-A (Ref. 21) documents the methodology used in assessing and applying the pressurizer safety valve response to the transient analysis. The NRC staff found that the NRC had previously approved both TRs for Westinghouse plants in performing plant transient analysis. Since the TRs were approved by the NRC, and the PINGP is a Westinghouse-designed plant, the licensee's proposed use of the TRs is consistent with current licensing practice and is acceptable.

full-power values of 44 percent NRS plus 11 percent NRS uncertainty). In relation to the low-low and high-high SG level setpoints, the bottom and top of the NRS (0 percent NRS and 100 percent NRS) were assumed, respectively, in the transient analyses. Applicable uncertainties which are consistent with the assumptions used in the transient analyses are accounted for in the establishment of the TS Allowable Values for the low-low SG level setpoint. As for the ATWS analyses, the low SG level setpoint was assumed to be 35 percent wide range level (WRL), which included allowance for instrumentation uncertainties. The actual plant setpoint is 42.5 percent WRL. Since the licensee used the NRC-approved methods for setpoint calculations and the effects of instrumentation uncertainties discussed in the Westinghouse NSALs and the associated WCAP-16115 were accounted for in the transient analyses, the NRC staff concludes that the issue regarding the SG level measurement errors is resolved.

<u>RCP Coastdown Delay Times</u>

The licensee indicated in Reference 5 that in the analysis of the LOAC power event, the RCPs were assumed to lose power and begin coasting down 2 seconds following the turbine trip, resulting from the reactor trip. For the steam line break (SLB) analysis, the RCPs were assumed to begin coasting down 3 seconds after SLB initiation for the case without offsite power. The NRC staff requested the licensee to justify the validity of the use of different RCP coastdown delay times in the analysis for the LOAC and SLB events. In response, the licensee indicated (Ref. 24) that for both cases, it was assumed that loss of offsite power (LOOP) occurred as a consequence of instability on the power grid caused by the unit trip. The analysis assumed that for the cases with LOOP, the loss of power to RCPs caused RCPs coastdown which occurred 2 or 3 seconds following the turbine trip. The licensee stated that the RCP coastdown delay time is not an important parameter for the analysis of both LOAC and SLB events. Based on its evaluation, the licensee indicated that transient results, including DNBR, pressurizer pressure and the peak pressurizer water volume, will be negligibly affected if a zero delay time were assumed in the analysis. Also, the LOAC result in terms of the maximum pressurizer water volume was significantly less limiting than that for the LONF event in which continuous operation of the RCPs was assumed. For the SLB analysis, the result in terms of minimum DNBR for the case with LOOP was less limiting than the case with offsite power available. Therefore, the licensee stated and the staff agreed that the different RCPs coastdown delay times assumed in the analysis of LOAC and SLB events are acceptable.

The staff reviewed the non-LOCA analysis and summarized the computer codes used for each case and the results of the analysis for the most limiting cases in Table 1. The staff found that: (1) the licensee used the approved codes and methodologies to perform transient analyses; (2) the values used for the input parameters are conservative in predicting the worst consequences; (3) the computer codes are reliable in calculating core power, pressure, temperature, RCS flow and pressurizer water volume during the transient; and (4) the results of the analysis show that the acceptance criteria, satisfying the requirements of USAR Sections 3.1.2.1 and 4.3.1.2.1, regarding the fuel integrity and RCS pressure boundary integrity, for the analysis of record are met. Therefore, the NRC staff concludes that the analyses are acceptable.

3.2.1 ATWS Analysis

An ATWS event is defined as an anticipated operational occurrence (AOO) combined with an assumed failure of the reactor tip system to shutdown the reactor. For Westinghouse plants, the ATWS rule, 10 CFR 50.62, requires the installation of an ATWS mitigating system actuation circuitry (AMSAC) system to initiate a turbine trip and actuate AFW flow independent of the reactor protection system. The licensee has met the ATWS rule by installing an AMSAC at the PINGP. In addressing concerns regarding the operability of the AFW pump during an ATWS, the licensee subsequently installed an NRC-approved diverse scram system (DSS) at the PINGP. The DSS provides a reactor trip signal on a low SG wide range level signal and on RCP breaker position signal. In support of the request for NRC approval of the Westinghouse non-LOCA methods conversion, the licensee reanalyzed ATWS events to confirm that the AMSAC/DSS will provide adequate protection. The licensee presented its ATWS reanalysis in Reference 5. The licensee evaluated the ATWS analysis presented in the UFSAR and considered the need to reanalyze each AOO under ATWS conditions. As a result, the licensee identified five events that are not needed to explicitly reanalyze for ATWS conditions, because the events result in consequences bounded by other events. The events are: (1) uncontrolled RCCA withdrawal from a subcritical condition; (2) RCCA misalignment; (3) excessive heat removal due to feedwater system malfunctions, (4) excessive load incident; and (5) isolation of main condenser. The licensee performed ATWS reanalysis with the Westinghouse non-LOCA methods for the remaining USAR AOOs. The reanalyzed events include: (1) partial loss of reactor coolant flow; (2) LONF; (3) loss of AC to the station auxiliaries; (4) loss of external electrical load; (5) uncontrolled RCCA withdrawal at power; and (6) uncontrolled boron dilution. The licensee analyzed the ATWS events using the NRC-approved RETRAN-02 code.

Consistent with the NRC staff's position that was applied to the acceptable ATWS analyses, the licensee assumed nominal plant conditions in the reanalysis as initial plant boundary conditions. For example, the reactor power was assumed to be at 100 percent of the rated power, the initial pressure and temperature were assumed at their nominal values, the MSSVs and pressurizer power-operated relief valves were assumed to operate normally.

In the analysis, the licensee credited the DSS in conjunction with the AMSAC for event mitigation. The licensee assumed that upon actuation of the AMSAC/DSS signal, the turbine was tripped, the AFW pumps were started, and the DSS was actuated to insert the control rods. The following assumptions regarding the AMSAC/DSS were made for the analysis:

- The AMSAC/DSS SG WRL trip safety analysis setpoint is 35 percent WRL (as compared to 42.5 percent WRL for the normal trip setpoint).

- The AMSAC/DSS output signal is generated within 5 seconds.

- The AMSAC /DSS turbine trip delay is 10 seconds from the time that the AMSAC/DSS setpoint is reached.

- The AMSAC/DSS AFW start delay time is 65 seconds from the time that the AMSAC/DSS setpoint is reached.

The NRC staff found that the credit of the AMSAC/DSS for the control rod insertion, the turbine trip and the AFW pump actuation was assumed in the analysis of record. Therefore, the NRC staff concludes that the credit of the AMSAC/DSS in the reanalysis is acceptable.

In accordance with the NRC staff's review position on the ATWS analysis for the existing PWRs, such as the PINGP, the staff considers that an acceptable ATWS analysis shows that the unfavorable exposure time (UET), given the cycle design (including the moderator

3.3.6 TS TABLE 3.3.1-1 - Reactor Trip System Instrumentation Related to Item 8.a, Pressurizer Pressure-Low

In response to the NRC staff's questions regarding the need to increase the allowable value for pressurizer pressure reactor trip from the existing value of 1760 psig to 1845 psig, the licensee provided the following explanation in their letter dated January 14, 2004.

The licensee states:
"The proposed change to this value is in the conservative direction. As described and illustrated in Appendix B of WCAP-8745-P-A, the low and high pressurizer reactor trips limit the range of conditions over which the OTΔT and OPΔT trips must provide protection. The more limited this range is, the less restrictive the setpoint constants need to be, which provides increased operational flexibility while still ensuring that safety limits are met. The current Allowable Value for pressurizer pressure - low is much lower than the actual plant setting. To credit and thus derive the benefit from this higher than actual plant setting in the ΔT trip setpoints it is necessary to increase the safety analysis setpoint, which makes it necessary to formally increase the Allowable Value for pressurizer pressure - low in the Technical Specifications."

The pressurizer pressure-low trip function is designed to ensure that protection is provided against violating the DNB due to low pressure. Current TS defines the setpoint of greater than or equal to 1760 psig as Item 8.a in TS Table 3.3.1-1. The proposed TS increases the setpoint to greater than or equal to 1845 psig. The licensee further confirmed (Ref. 6) that the proposed TS value bounds the value (1835 psig) used in the PINGP safety analysis discussed Reference 5. The licensee also stated that the difference between the actual plant setting and the TS allowable value provides sufficient margin for calibration tolerances and measurable instrument uncertainties such as instrument accuracy, setting drift, and so on. The NRC staff has reviewed the calculation provided by the licensee for general adequacy and adherence to the PINGP setpoint methodology and agrees with the licensee's determination. Therefore, the NRC staff concludes that the change is acceptable.

3.3.7 TS 5.6.5 - COLR

TS 5.6.5.a provide a list of TSs for which the core operating limits are documented in the COLR. The proposed TS 5.6.5.a (Refs. 1 and 2) adds to list TS 2.1.1, "Reactor Core SLs," and LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation Overtemperature ΔT and Overpower ΔT Parameter Values for Table 3.3.1-1." The changes are acceptable because the added TS and LCO specify the cycle-specific parameters that are allowed to be relocated to the COLR based on the evaluation discussed in Sections 3.3.1 and 3.3.5 above.

TS 5.6.5.b provides a list of titles of TRs documenting the NRC-approved methodologies used to determine the values of cycle-specific parameters that are removed to the COLR. The proposed TS adds the following titles of TRs to the reference list (the Reference numbers refer to those in the proposed TS 5.6.5.b (Ref. 1)) :

- Reference 13 - WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control/F_Q Surveillance Technical Specifications;"
- Reference 14 - WCAP 8745-P-A, "Design Bases for the Thermal Overpower ΔT and Overtemperature ΔT Trip Functions;"

. References 6 and 7 of TS 5.6.5.b are identical when their date, volume and addendum are removed. The NRC staff finds that the removal of Reference 6 is also acceptable.

4.0 SUMMARY

Based on the review, the NRC staff has determined that the proposed Westinghouse methodologies and computer codes for non-LOCA analyses discussed in Sections 3.1 and 3.2 above are acceptable for PINGP, Units 1 and 2, because they were previously approved by the NRC for Westinghouse plants. The licensee's use of the methodologies and computer codes in performing reload analysis for the PINGP complies with the NRC staff's SE limitations imposed on the use of the approved methods or codes, and the results of the non-LOCA analyses using the Westinghouse methods satisfy the requirements of USAR Sections 3.1.2.1 and 4.3.1.2.1 with respect to integrity of the fuel and RCS pressure boundary.

This LAR referenced the Engineering Manual Section 3.3.4.1, "Engineering Design Standard for Instrument Setpoint/ Uncertainty Calculations," for the instrument setpoint and uncertainty calculation, which was accepted by the NRC staff during ITS conversion. The NRC staff's review of the submitted calculations determined that the PINGP setpoint methodology has followed the guidance provided in RG 1.105, Revision 3, and the requirements of 10 CFR 50.36 and, therefore, is acceptable.

The NRC staff also determined that the associated proposed changes to TSs 2.1.1, 3.1.8, 3.2.1, 3.2.3, TS Table 3.3.1-1, and 5.6.5 are acceptable because they meet the current regulation (10 CFR 50.36) and are consistent with the GL 88-16 guidance and NUREG-1431 (Revision 2), "Standard Technical Specifications - Westinghouse Plants," discussed in Section 3.3.

Table 1 - The NON-LOCA Analysis Results and Computer Codes and Critical Heat Flux Correlations Used in the Analysis

Events Analyzed	Codes Used	Result parameters	Results for limiting case
Uncontrolled RCCA Withdrawal from a Subcritical Condition	TWINKLE FACTRAN VIPRE	Min. DNBR below first mixing grid (W-3 correlation, thimble/typical) Min. DNBR above first mixing grid (WRB-1 correlation, thimble/typical) Max. fuel centerline temp., EF	1.703/1.849 2.047/2.075 2,480
Uncontrolled RCCA Withdrawal at Power	RETRAN	Min. DNBR (WRB-1) Peak RCS pressure, psia Peak SG pressure, psia	1.432 2,572.7 1,185.4

Events Analyzed	Codes Used	Result parameters	Results for limiting case
Rod Cluster Control Assembly Misalignment	LOFTRAN ANC VIPRE	Min. DNBR (WRB-1) Peak linear heat generation, kW/ft Peak uniform cladding strain, %	>1.34 <22.54 <1.0
Chemical and Volume Control System Malfunction Mode 1 with manual rod control Mode 1 with auto. rod control Mode 2 Modes 3, 4 , 5 and 6	N/A	Min. time to loss of shutdown margin, minutes	 >20 >21 >17 >24
Excessive Heat Removal due to Feedwater System Malfunctions	RETRAN VIPRE	Min. DNBR (WRB-1)	1.41
Excessive Load Increase Incident	RETRAN VIPRE	Min. DNBR (WRB-1) Peak core heat flux, %	1.49 117.1
Loss of Reactor Coolant Flow - Partial Loss of Flow - Complete Loss of Flow Locked-Rotor Accident	RETRAN VIPRE	 Min. DNBR (WRB-1) Min. DNBR (WRB-1) Max. percent rods in DNB, % Peak RCS pressure, psia Peak cladding temp., EF Max. Zirc-water reaction, %	 1.607 1.333 18.4 2,562 2,010 0.6
Loss of External Electrical Load	RETRAN VIPRE	Min. DNBR (WRB-1) Peak RCS pressure, psia Peak SG pressure, psia	1.77 2,701 1,207.3
Loss of Normal Feedwater	RETRAN	Max. pressurizer mixture volume, ft^3	934.1
Loss of All AC Power to the Station Auxiliaries	RETRAN	Max. pressurizer mixture volume, ft^3	653.0

Events Analyzed	Codes Used	Result parameters	Results for limiting case
Rupture of a Steam Pipe-Core Response - Zero Power - Full Power	RETRAN VIPRE	Min. DNBR (W-3) Min. DNBR below first mixing vane grid (W-3) (thimble/typical) Min. DNBR above first mixing vane grid (WRB-1) (thimble/typical) Peak linear heat generation (kW/ft)	2.535 1.448/1.681 1.537/1.578 <22.54
RCCA Ejection	TWINKLE FACTRAN	Max. fuel pellet average enthalpy, cal/g	# 168.6

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (68 FR 22750). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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

Prairie Island Nuclear Generating Plant,
Units 1 and 2

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