September 13, 2000

Mr. Joel Sorensen
Site General Manager
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - SAFETY EVALUATION ON TOPICAL REPORT, NSPNAD-8101, REVISION 2, "QUALIFICATION OF REACTOR PHYSICS METHODS FOR APPLICATION TO PRAIRIE ISLAND UNITS 1 AND 2" (TAC NOS. MA7997 AND MA7998)

Dear Mr. Sorensen:

By letter dated January 3, 2000, the Northern States Power Company (NSP) requested NRC approval of the subject topical report regarding qualification of new reactor physics methods for application to Prairie Island Nuclear Generating Plant, Units 1 and 2. The proposed reactor physics methods are based on the CASMO-4/SIMULATE-3 computer codes, and are intended for safety analyses supporting core reload and operations.

We have reviewed the subject topical report and our evaluation is enclosed. We have concluded that the proposed methodology (NSPNAD-8101, Revision 2), as validated, can be applied to the steady-state physics calculations for safety analyses associated with core reload and operations. Our approval is subject to, and limited to, the range of fuel configurations and core design parameters as stated and referenced in NSP's January 3, 2000, submittal. Introduction of significantly different or new fuel designs will require further validation of the above stated physics methods, and subsequent staff approval.

Sincerely,

/RA/

Tae Kim, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosure: Safety Evaluation

cc w/encl: See next page

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Prairie Island Nuclear Generating Plant,
Units 1 and 2

cc:

J. E. Silberg, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N. W.
Washington, DC 20037

Plant Manager
Prairie Island Nuclear Generating Plant
Nuclear Mana
gement Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

Adonis A. Neblett
Assistant Attorney General
Office of the Attorney General
455 Minnesota Street
Suite 900
St. Paul, MN 55101-2127

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
1719 Wakonade Drive East
Welch, MN 55089-9642

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Mr. Stephen Bloom, Administrator
Goodhue County Courthouse
Box 408
Red Wing, MN 55066-0408

Commissioner
Minnesota Department of Commerce
121 Seventh Place East
Suite 200
St. Paul, MN 55101-2145

Site Licensing
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

Tribal Council
Prairie Island Indian Community
ATTN: Environmental Department
5636 Sturgeon Lake Road
Welch, MN 55089

Site General Manager
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

NUCLEAR MANAGEMENT COMPANY, LLC

TOPICAL REPORT, NSPNAD-8101, REVISION 2

QUALIFICATION OF REACTOR PHYSICS METHODS

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated January 3, 2000, the licensee requested approval of Topical Report, NSPNAD-8101, Revision 2, (Reference 1), regarding qualification of new reactor physics methods for the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2. The current reactor physics methods, as described in NSPNAD-8101, Revision 1, was approved in 1982 (Reference 2).

Topical Report NSPNAD-8101, Revision 2, provides description of the new reactor physics methods which are based on the CASMO-4/SIMULATE-3 computer codes. The proposed methods are similar in many respects to the methods previously approved (NSPNAD-8101, Revision 1) which is based on the CASMO/PDQ/DP5 computer codes.

Topical Report NSPNAD-8101, Revision 2, includes discussions on qualification and quantification of reliability factors (RF), and applications of the proposed methods to operations and reload safety analyses. The January 3, 2000, letter indicated that the proposed reactor physics methods will be used as inputs in the following applications:

- NSPNAD-8102-PA, Revision 7, July 1999, “Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units;”

- NSPNAD-93003-A, Revision 0, April 1993, “Prairie Island Units 1 and 2 Transient Power Distribution Methodology;” and

- NSPNAD-8408-A, Revision 2, May 1991, “Prairie Island Units 1 and 2 Rod Swap Methodology.”

As part of the staff’s review of the subject topical report, a teleconference was held between the NRC staff and the licensee on May 17, 2000, to clarify certain aspects of the information provided in the topical report.

2.0 EVALUATION

2.1 COMPUTER CODES DESCRIPTIONS AND METHODOLOGY

Topical Report NSPNAD-8101, Revision 2, describes the licensee's plant-specific reactor physics methods (a calculational model), which are based on the CASMO-4 and SIMULATE-3 computer codes developed by Studsvik (References 3 and 4). The CASMO-4 computer code is used to generate the lattice physics parameters (i.e., the cross sections, nuclide concentrations, pin power distributions, and other nuclear data) that are used as inputs to SIMULATE-3. CASMO-4 is a multi-group, two-dimensional transport theory code for depletion and branch calculations for a single assembly.

Some of the characteristics of CASMO-4 include:

- 40-energy-group cross section library;
- energy groups are used during the two-dimensional transport calculations;
- Gadolinium and other burnable absorbers are depleted microscopically;
- the predictor-corrector approach is used for depletion; and
- effective resonance cross sections are calculated individually for each pin.

SIMULATE-3 is a two-group, three-dimensional nodal code based on the NRC-approved QPANDA neutronics code. SIMULATE-3 utilizes the modified coarse mesh (nodal) diffusion theory calculational technique, coupled with thermal hydraulic and Doppler feedback. The code includes the following modeling capabilities: 1) solution of the two group neutron diffusion equation in three dimensions; 2) fuel assembly homogenization, 3) explicit reflector cross-section model, and 4) cross-section depletion and pin power reconstruction. In order to ensure flux continuity at nodal interfaces and accurate determination of pin-wise power distributions, SIMULATE-3 uses assembly discontinuity factors that are pre-calculated by CASMO-4. These factors are related to the ratio of the nodal surface flux in the actual heterogenous geometry to the cell averaged flux in an equivalent homogeneous model, and are determined for each energy group as a function of exposure, moderator density, and control-rod-state.

The assembly homogenization employs the flux discontinuity correction factors from CASMO-4 to combine the global (nodal) flux shape and the assembly heterogeneous flux distribution. The flux discontinuity concept is also applied to the baffle/reflector region in both radial and axial directions to eliminate the need for normalization, or other adjustment at the core/reflector interface.

The SIMULATE-3 fuel depletion model uses tabular and functionalized macroscopic or microscopic, or both, cross sections to account for fuel exposure without tracking the individual nuclide concentrations. Depletion history effects are calculated by CASMO-4 and then processed by the TABLES-3 code (Reference 5) for generation of the cross section library used by SIMULATE-3.

SIMULATE-3 can be used to calculate the three-dimensional pin-by-pin power distribution in a manner that accounts for individual pin burnup and spectral effects. SIMULATE-3 also calculates control rod worth as well as feedback from the effects of moderator temperature, Doppler, and Xenon.

ESCORE (References 6-9) is an Electric Power Research Institute (EPRI)-developed computer code for steady-state fuel performance analysis. This code is capable of modeling the thermal and mechanical responses of light-water reactor fuel and is used to generate fuel temperature predictions to be used as inputs to CASMO-4 and SIMULATE-3.

2.2 BENCHMARKING AND MODEL VERIFICATION

Data collected from several operating cycles at PINGP, Units 1 and 2, were used to benchmark the CASMO-4/SIMULATE-3 model to quantify the reliability factors. The subject topical report defines the term RF as the allowances to be assure conservatism. The uncertainty factor (1σ) is used to account for the actual model accuracy. The RF is always larger than the uncertainty factor. The term bias is defined as the statistical difference between a measured value and the calculated value.

The qualification benchmarking compares the outputs from the CASMO-4/SIMULATE-3 model with measurements obtained from the plant data covering cycles 16 through 19 for PINGP, Units 1 and 2. The plant analyses were performed over a wide range of conditions from cold (ambient) temperature to hot full power operation.

During each cycle, measured and calculated parameters will be compared in order to validate and update the RFs. Results of the validation and any needed updates for each parameter are documented as part of the reload safety evaluation for the reload in which the updated values will be used. The updates to the reliability factors will be conducted in accordance with methodology outlined in this submittal (Reference 1).

The licensee stated that if the comparisons for the new cycles give higher reliability factors, the higher RF will be used for reload safety evaluations going forward until comparisons indicate that the RF of Table 3.0.1 in the subject topical report are met. The licensee stated that RFs lower than those listed in Table 3.0.1 will never be used for any reload safety evaluations, even if comparisons indicate that they would be valid.

The licensee also stated that it intends to use these methods for steady-state pressurized-water reactor core physics reload design and licensing applications including generation of core physics parameters (i.e., rod and boron worth benchmarking, temperature coefficient benchmarking and startup predictions, and reactivity coefficients) for transient and safety analyses input.

2.3 STATISTICS

The calculational RF for any of the reactor physics parameters such as F_Q is defined as a single value (i.e., delta F_Q), such that the calculated value of F_Q at any core location plus delta F_Q has a 95-percent probability with a 95-percent confidence level of being conservative with respect to the measured F_Q at that location.

3.0 CONCLUSION

Topical Report NSPNAD-8101, Revision 2, provided the results of extensive bench-marking studies on the proposed CASMO-4/SIMULATE-3 model. This included detailed comparisons of calculated key physics parameters with measurements obtained from several operating cycles at PINGP. These results were used to determine the set of 95/95 (probability/confidence) tolerance limits for application to the calculation of the stated physics parameters.

Based on the information presented in the January 3, 2000, submittal, the staff has concluded that the licensee's proposed reactor physics methods based on the CASMO-4/SIMULATE-3 model, and as validated by the licensee, can be applied to the steady-state physics calculations for applications as described in the preceding sections of this evaluation. The staff's approval is subject to, and limited to, the range of fuel configurations and core design parameters as stated and referenced in the topical report. Introduction of significantly different or new fuel designs will require further validation of the above stated physics methods by the licensee, and approval by the NRC staff.

4.0 REFERENCES

1. Northern States Power letter from Joel P. Sorensen (NSP) to NRC, "NSPNAD-8101, Revision 2: Prairie Island Nuclear Generating Plant Qualification of Reactor Physics Methods for application to Prairie Island," January 3, 2000.

2. NSP Topical, "Qualification of reactor physics methods for Application to PI Units," NSPAND-08101-A, Revision 1, December 1982.

3. A.S. Digiovine, J.D. Rhodes, III, K.S. Smith, D.M. Ver Planck, and J.A. Umbarger, "SIMULATE-3 Advanced three-dimensional Two-group Reactor Analysis Code User's Manual," Studsvik of America, Inc., SOA-95/15 Rev. 0, October 1995.

4. J.T. Cronin, K.S. Smith, D.M. Ver Planck, J.A. Umbarger, and M.Edenius, "SIMULATE-3 Advanced three-dimensional Two-group Reactor Analysis Code User's Manual," Studsvik of America, Inc. SOA-95/18 Rev. 0, October, 1995.

5. K.S. Smith, D.M. Ver Planck, and J.A. Umbarger, "TABLES-3 Library Preparation Code for SIMULATE-3," Studsvik of America, Inc., SOA-95/15 Rev. 0, October, 1995.

6. I.B. Fiero, M.A. Krammen, H.R. Freeburn, et al., "ESCORE-The EPRI Steady-state core reload evaluator code: General Description," EPRI-NP-5100-L-A, April 1991.

7. M.A. Krammen, H.R. Freeburn, et al., "ESCORE-The EPRI Steady-state core reload evaluator code Volume 1: Theory Manual," Electric Power Research Institute, EPRI-NP-4492-CCMP, August 1986.

8. M.A. Krammen, R. B. Fancher, et al., "ESCORE-The EPRI Steady-state core reload evaluator code Volume 2: User's Manual," Electric Power Research Institute, EPRI-NP-4492-CCMP, August 1986.

9. M.A. Krammen, R. B. Fancher, et al., “ESCORE-The EPRI Steady-state core reload evaluator code Volume 3: Programer’s Manual,” Electric Power Research Institute, EPRI-NP-4492-CCMP, August 1986.

Principal Contributor: A. Attard

Date: September 13, 2000