December 24, 2008

Mr. Michael D. Wadley
Site Vice President
Prairie Island Nuclear Generating Plant, Units 1 and 2
Northern States Power Company, Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE REVIEW OF THE LICENSE RENEWAL APPLICATION FOR PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 (TAC NOS. MD8528 AND MD8529)

Dear Mr. Wadley:

By letter to Northern States Power Co. (NSP), dated October 23, 2008, the U.S. Nuclear Regulatory Commission (NRC or staff) requested that additional information regarding Severe Accident Mitigation Alternatives (SAMAs) be provided in order for the staff to complete its review of the Prairie Island Nuclear Generating Plant, Units 1 and 2, license renewal application. By letter dated November 21, 2008, NSP provided responses to the staff's requests for additional information (RAIs). Upon review of your responses, the staff has identified areas where additional information is needed in order to complete its review.

Enclosed is the staff's additional RAI regarding SAMAs. As discussed with Mr. James Holthaus of your staff, a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at (301) 415-2703 or by e-mail at Nathan.Goodman@nrc.gov.

Sincerely,

\RA

Nathan Goodman, Project Manager
Projects Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosure:
As stated

cc w/encl: See next page

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OFFICE	LA:DLR	PM:DLR:RPB2	OGC	BC:DLR:RPB2
NAME	IKing	NGoodman	BMizuno	DWrona w/comments
DATE	12/22/08	12/22/08	12/23/08	12/24/08

OFFICIAL RECORD COPY

Letter to NSP from N. Goodman, dated December 24, 2008

DISTRIBUTION:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE REVIEW OF THE LICENSE RENEWAL APPLICATION FOR PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 (TAC NOS. MD8528 AND MD8529)

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Request for Additional Information (RAI)
Regarding the Analysis of Severe Accident Mitigation Alternatives for Prairie Island Nuclear Generating Plant, Units 1 and 2

RAI 1e – The response (p. 10) noted a probabilistic risk assessment (PRA) maintenance and an updated fact and observation (F&O) had been identified and resolved. Describe the F&O and its resolution. In addition, describe the scope and personnel qualifications of the three reviews identified as being part of the self-assessment process (p. 11).

RAI 1f – The Prairie Island Nuclear Generating Plant (PINGP) PRA uses a Westinghouse reactor coolant pump seal loss-of-coolant accident (LOCA) model (WCAP-10541, 1986), that pre-dates the Westinghouse Owners Group (WOG) 2000 model approved by the NRC in 2003 for plants using high-temperature O-rings. The peer review of the PINGP PRA occurred prior to the approval of the WOG 2000 model, and as such would not have identified this as an issue. Provide an assessment of the impact on the severe accident mitigation alternative (SAMA) identification and screening if the PINGP PRA utilized the WOG 2000 model.

RAI 1h – Based on the description provided, the dominant internal flooding sequence (involving cooling water header rupture) would result in core damage at both units. The benefits of any related SAMAs should therefore be doubled. Identify all sequences resulting in core damage at both units. Confirm that the benefits for related SAMAs were appropriately assessed, i.e., doubled where appropriate. (Also see RAI 5.b)

RAI 2b – The last paragraph of the RAI response provides a qualitative comparison of the conditional probabilities of the steam generator tube rupture (SGTR) under specific primary and secondary side conditions, but does not include a characterization of the PINGP-specific results for induced SGTR. Provide the frequency-weighted conditional probability of temperature induced-SGTR (over all sequences involving high primary side and low secondary side pressure, and a dry secondary side) for PINGP. Provide an assessment of the impact on the SAMA identification and screening if a conditional probability of 0.25 (similar to NUREG-1570) is assumed for these sequences.

RAI 3a and 3b – In order to support the assumption that the fire core damage frequency (CDF) is comparable to the internal events CDF (9.79E-6 for Unit 1 and 1.21E-5 for Unit 2), it should be shown, preferably through sensitivity analysis or other quantitative arguments, that the individual plant examination of external events (IPEEE) fire CDF value (4.9E-5) is conservative by a factor of 4 to 5 (for Units 2 and 1, respectively). The information provided in the RAI response is general and qualitative in nature, and does not sufficiently demonstrate that such a large reduction in the fire CDF is appropriate (For example, the discussion of control room fires [65% contributor] states that partitioning of a cabinet within a panel zone was not credited. What is not stated is that the main control panel is a contiguous arrangement of panel sections without barriers or boundaries. The IPEEE used partitioning process of overlapping zones [25 zones] to subdivide the panel based on consideration of nominal panel fire heat rate, nominal heat value of the cable bundle, available fire suppression time of 15 minutes and the general vertical propagation tendency of fire in open back panels. Therefore, the zones are subdivided panel sections. In addition, the statement that manual suppression credit was only applied to cutsets representing <13% of the internal fires CDF appears to be misleading. The IPEEE indicates that manual suppression was applied to all control room fires with a 10 minute fire suppression failure probability of 1.6E-2.). However, the 46 percent reduction in the conditional core damage probability (CCDP) since the IPEEE cited in the response (p. 23) would suggest that a factor of 2 reduction in CDF might be justified. Provide additional information on how the CCDP was computed, and on how the events on which it is based relate to the dominant fire events. Clarify whether the CCDP value includes station blackout events.

RAI 5a. – It is understood from the response that improved training will not provide any additional benefit. However, the failure probabilities of 1.9E-02 and 5.3E-02 appear to have room for improvement. Explain the characteristics of these actions (and the calculator used to determine its value) that prevents lower calculated values given excellent training and emergency operating plan-driven direction.

RAI 5b – Three candidate SAMAs (6, 6a, and 13) and 2 IPE-identified enhancements related to internal flooding were dismissed on the basis of a cooling water header piping modification in 1992, and deterministic considerations described in a 1995 engineering calculation/white paper. However, the IPE and 7 subsequent PRA updates (up to and including the current PRA) continue to model the rupture of the cooling water header. Justify why the piping modification should be credited (for eliminating cooling water header ruptures) in the SAMA evaluation, in view of the fact that the IPE and subsequent PRA updates continue to model these pipe breaks, and that the American Society of Mechanical Engineers PRA standard would call for treatment of such flood sources. Provide a quantitative evaluation of the costs and benefits of each of the aforementioned SAMAs / enhancements based on the current PRA treatment of cooling water header pipe breaks.

RAI 5d and 5e – A review of Table F.5-3 finds that several screened Phase I candidates (i.e., SAMAs 6, 6a, 7, 8, 13, 14 and 16) do not appear to meet the environmental report (ER) Section 4.17.1 screening criteria. In addition, the discussion for the basis for screening SAMA 14 does not appear to address the benefit of improved operator training for power-operated relief valve failure to re-seat. Its screening appears to be based on model limitations as opposed to actual benefit. As such, the ER Section F.5.2 criteria do not appear to be consistent with the ER Section 4.17.1 criteria. Confirm that both sets of screening criteria are used. Explicitly identify the ER Section F.5.2 and the ER Section 4.17.1 screening criterion used for each screened SAMA.

RAI 5g - Based on the description provided, the dominant internal flooding sequence (involving cooling water header rupture) would result in core damage at both units. Provide an evaluation of a less extensive, alternative SAMA that would limit water damage to the systems, structures, and components for a single unit (so that core damage would be limited to one unit). Provide the costs and benefits for this alternative.

RAI 6c – The life-cycle cost is identified as $100K. However, SAMA 20 changes a normally-open motor-operated valve to normally-closed. Demonstrate that this change will add $100K additional life-cycle cost to an existing valve. In addition, the noted design cost reduction of 30% does not yield the reduced second unit cost. Address this apparent discrepancy.

RAI 6.g – The corrected treatment of uncertainties shows SAMA 19a as potentially cost beneficial. Discuss Nuclear Management Company's, plans for further evaluation or implementation of this SAMA.

cc:

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