UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 30, 2010

Mr. Mark A. Schimmel
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power Company – Minnesota
1717 Wakonade Drive East
Welch, MN 55089

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

Dear Mr. Schimmel:

By letter dated April 11, 2008, Nuclear Management Company, LLC, now known as Northern States Power Company, Minnesota, submitted an application pursuant to Title 10 of the *Code of the Federal Regulations* Part 54, to renew the operating license for Prairie Island Nuclear Generating Plant, Units 1 and 2, for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The staff has identified, in the enclosure, areas where additional information is needed to complete the review. Further requests for additional information may be issued in the future.

Items in the enclosure were discussed with Gene Eckholt, of your staff, and 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 301-415-1427 or by e-mail at Richard.Plasse@nrc.gov.

Sincerely,

Richard Plasse, Project Manager
Projects Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Dockets Nos. 50-282 and 50-306

Enclosure:
As stated

cc w/encl: Distribution via Listserv

RAI B.2.1.30

Background:
By letter dated November 5, 2010, the applicant provided supplemental information to license renewal application (LRA) Section B.2.1.30, One Time Inspection of ASME Code Class 1 Small Bore Piping Program. In Enclosure 4 of the supplement, the applicant stated that it excluded Nominal Pipe Size (NPS) 1 inch from the scope of program.

Issue:
Generic Aging Lessons Learned (GALL) aging management program (AMP) XI.M35 states in "scope" program element that, "This program is a one-time inspection of a sample of ASME Code Class 1 piping less than NPS 4 and greater than or equal to NPS 1."

Request:
Incorporate ASME Code Class 1 piping of NPS 1 inch into the scope of the One Time Inspection of ASME Code Class 1 Small Bore Piping Program, consistent with GALL AMP XI.M35, or justify the adequacy of the AMP when excluding NPS 1 inch from the scope of the program.

RAI 3.1.2.2.16

Background:
Standard Review Plan-License Renewal (SRP-LR) Section 3.1.2.2.16 identifies that cracking due to primary water stress corrosion cracking (PWSCC) could occur on the primary coolant side of PWR steel steam generator (SG) tube-to-tube sheet welds made or clad with nickel alloy. The GALL Report recommends ASME Section XI ISI and control of water chemistry to manage this aging. The GALL Report also recommends no further aging management review for PWSCC of nickel alloy if the applicant complies with applicable NRC Orders and provides a commitment in the Final Safety Analysis Report (FSAR) supplement to implement applicable (1) Bulletins and Generic Letters and (2) staff-accepted industry guidelines. In GALL Report Revision 1, Volume 2, this aging is addressed in item IV.D2-4, applicable only to once-through SGs, but not to recirculating SGs.

The staff noted that ASME Code Section XI does not require any inspection of the tube-to-tubesheet welds. In addition, no NRC Orders or bulletins require examination of this weld. The staff's concern is that, if the tubesheet cladding is Alloy 600 or the associated Alloy 600 weld materials, the tube-to-tubesheet weld region may have insufficient Chromium content to prevent initiation of PWSCC, even when the SG tubes are made from Alloy 690TT. Consequently, such a PWSCC crack initiated in this region, close to a tube, could propagate into/through the weld, causing a failure of the weld and of the reactor coolant pressure boundary, for both recirculating and once-through steam generators.

In LRA Table 3.1.1, the applicant stated that item Number 3.1.1-35 is not applicable because this line is applicable to once-through SGs and not to recirculating SGs used at the applicant's plant.

In Update Safety Analysis Report (USAR) Table 4.1-1, the applicant described that the Unit 1 replacement Framatome Model 56/19 SG tubes are fabricated from Alloy 690TT and the cladding for the tubesheets from Alloys 82 and 182, and that the Unit 2 Westinghouse Model 51 SG tubes are fabricated from Alloy 600MA and the cladding for the tubesheets from Inconel. In USAR Section 4.3.2.4, the applicant stated that the NRC has approved an amendment to its Technical Specifications which allows Unit 2 tubes to remain in service if the required length of hard roll expansion is intact above the highest degradation in the tubesheet crevice region (F* Alternate Repair Criteria). However, the staff noted that the applicant will replace Unit 2 SGs before the period of extended operation. Therefore, the staff does not have information about the configuration of the replacement SGs tube-to-tubesheet welds and the necessity to manage the potential aging effect of cracking due to PWSCC in these welds.

Issue:
Unless the NRC has approved a redefinition of the pressure boundary in which the autogenous tube-to-tubesheet weld is no longer included, or the tubesheet cladding and welds are not susceptible to PWSCC, the staff considers that the effectiveness of the primary water chemistry program should be verified to ensure PWSCC cracking is not occurring.

Request:
(1) For Unit 1 SGs tube-to-tubesheet welds, provide either a plant-specific AMP that will complement the primary water chemistry program, in order to verify the effectiveness of the primary water chemistry program and ensure that cracking due to PWSCC is not occurring in tube-to-tubesheet welds, or justify why such a verification program is not needed.

(2) For Unit 2, clarify when you will replace the original SGs and discuss whether this is before or after the start of the period of extended operation. Describe whether the tubesheet cladding and the tube-to-tubesheet welds of your replacement steam generators are susceptible to PWSCC and provide the materials of construction. If these materials are potentially susceptible to PWSCC (e.g., Alloy 600 and/or its associated weld metals), provide an AMP, along with the primary water chemistry program, that will verify the effectiveness of the primary water chemistry program and will ensure that cracking due to PWSCC is not occurring in the tube-to-tubesheet welds.

RAI 4.3.3

Background:
In LRA Section 4.3.3, Commitment 33, and the letter dated April 28, 2009 to close Commitment Number 36, the applicant provided a discussion on the methodology used to determine the locations that required environmentally assisted fatigue (EAF) analyses, consistent with NUREG/CR-6260. The staff recognized that there are seven plant-specific components developed, in LRA Table 4.3-8, based on the six generic locations identified in NUREG/CR-6260. The staff noted that it was stated in the LRA the limiting pressurizer surge line location is at the safe end connected to the hot leg nozzle.

Issue:
GALL AMP X.M1 states the impact of the reactor coolant environment on a sample of critical components should include the locations identified in NUREG/CR-6260, as a minimum, with an implication that additional locations should be included. During its review, the staff was

concerned whether the applicant had verified that the plant-specific components listed in Table 4.3-8 per NUREG/CR-6260 were bounding for the generic NUREG/CR-6260 locations (charging system nozzle). Furthermore, the staff noted that the applicant's plant-specific configuration may contain more limiting locations that should be analyzed for the effects of reactor coolant environment, other than those generic locations identified in NUREG/CR-6260. The staff noted this may include locations, for example, (1) that are limiting or bounding for a particular plant-specific configuration or (2) that have calculated cumulative usage factors (CUF) values that are greater than those for the locations identified in NUREG/CR-6260.

Request:

(1) Confirm and justify that the plant-specific components listed in LRA Table 4.3-8 were bounding for the generic NUREG/CR-6260 locations.

(2) Confirm and justify that the NUREG/CR-6260 locations selected for EAF analyses consists of the most limiting and bounding for the whole plant (in addition to NUREG/CR-6260 locations). If these locations do not include the most limiting and bounding for the plant, clarify the locations that require an EAF analysis and the actions that will be taken for these additional locations. If the limiting component identified consists of nickel alloy, clarify that the methodology used to perform EAF calculation for nickel alloy is consistent with NUREG/CR-6909. If not, justify the method chosen and why NUREG/CR-6909 is not used.

RAI B2.1.29-1

Background:

GALL AMP XI.M32, "One-Time Inspection" states in element 4, "detection of aging effects" that the inspection includes a representative sample of the system population, and, where practical, focuses on the bounding or lead components most susceptible to aging due to time in service, severity of operating conditions, and lowest design margin.

LRA Section B2.1.29, One-Time Inspection, states that the program elements include: (a) determination of the sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience; and (b) identification of inspection locations in the system, component, or structure based on the aging effect. During the audit, the staff noted that the applicant's program basis document refers to the methodology discussed in EPRI TR-107514, "Age-Related Degradation Inspection Methods and Demonstration: In Behalf of Calvert Cliffs Nuclear Power Plant License Renewal Application," as one industry source to be considered in developing the One-Time Inspection Program.

Issue:

Due to the uncertainty in determining the most susceptible locations and the potential for aging to occur in other locations, the staff noted that large (at least 20%) sample sizes may be required in order to adequately confirm an aging effect is not occurring. The applicant's One-Time Inspection Program did not include specific information regarding how the population of components to be sampled or the sample size will be determined.

Request:

Provide specific information regarding how the population of components to be sampled will be determined and the size of the sample of components that will be inspected.

RAI B2.1.36-2

Background:
GALL AMP XI.M33, "Selective Leaching of Materials" states in element 1, "scope of program" that the program includes a one-time visual inspection and hardness measurement of a selected set of sample components to determine whether loss of material due to selective leaching is not occurring for the period of extended operation.

LRA Section B2.1.36, Selective Leaching of Materials, states that plant and industry operating experience will be used to establish sample size, inspection locations, and inspection techniques.

Issue:
Due to the uncertainty in determining the most susceptible locations and the potential for aging to occur in other locations, the staff noted that large (at least 20%) sample sizes may be required in order to adequately confirm an aging effect is not occurring. The applicant's Selective Leaching Program did not include specific information regarding how the selected set of components to be sampled or the sample size will be determined.

Request:
Provide specific information regarding how the selected set of components to be sampled will be determined and the size of the sample of components that will be inspected.

RAI B2.1.38

Background:
NRC staff review has determined that adequate acceptance criteria for the Structures Monitoring Program should include quantitative limits for characterizing degradation. Chapter 5 of ACI 349.3R provides acceptable criteria for concrete structures. If the acceptance criteria in ACI 349.3R are not used, plant-specific criteria should be described and a technical basis for deviation from ACI 349.3R should be provided.

Issue:
Although the LRA states that the applicant's Structures Monitoring Program incorporates inspection guidance based on recommendations contained in ACI 349.3R, it does not clearly identify quantitative acceptance criteria for inspections.

Request:
a) Confirm that quantitative acceptance criteria are used for the Structures Monitoring Program. If the criteria deviate from those discussed in ACI 349.3R, provide technical justification for the differences.
b) If quantitative acceptance criteria will be added to the program as an enhancement, provide plans and a schedule to conduct an inspection with the quantitative acceptance criteria prior to the period of extended operation.

November 30, 2010

Mr. Mark A. Schimmel
Site Vice President
Prairie Island Nuclear Generating Plant
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/RA/

Richard Plasse, Project Manager
Projects Branch 2
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Office of Nuclear Reactor Regulation

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DATE	11/24/10	11/29/10	11/30/10

OFFICIAL RECORD COPY

Letter to M. Schimmel from R. Plasse dated November 30, 2010

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