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

June 10, 2010

Mr. Mark A. Schimmel
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota
1717 Wakonade Drive East
Welch, MN 55089-9642

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION RELATED TO LICENSE AMENDMENT REQUEST TO EXCLUDE THE DYNAMIC EFFECTS ASSOCIATED WITH CERTAIN POSTULATED PIPE RUPTURES FROM THE LICENSING BASIS BASED UPON APPLICATION OF LEAK-BEFORE-BREAK METHODOLOGY (TAC NOS. ME2976 AND ME2977)

Dear Mr. Schimmel:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated December 22, 2009 (Agencywide Documents Access and Management System Package No. ML100200129), Northern States Power Company, a Minnesota corporation (the licensee), doing business as Xcel Energy, submitted a request for application of a leak-before-break methodology to piping systems attached to the reactor coolant system for the Prairie Island Nuclear Generating Plant, Units 1 and 2.

The NRC staff is reviewing your submittal and has determined that additional information is required to complete the review. The specific information requested is addressed in the enclosure to this letter. During a discussion with your staff on May 18, 2010, it was agreed that you would provide a response within 45 days of the date of this letter.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. If circumstances result in the need to revise the requested response date, please contact me at (301) 415-4037.

Sincerely,

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosure:
Request for Additional Information

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REQUEST FOR ADDITIONAL INFORMATION

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

By letter dated December 22, 2009 (Agencywide Documents Access and Management System Package Accession No. ML100200129), the Northern States Power Company, a Minnesota corporation (the licensee), doing business as Xcel Energy, requested a license amendment to allow implementation of leak-before-break (LBB) on the safety injection lines, residual heat removal lines, reactor coolant system (RCS) draindown line, at Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, and the pressurizer surge line at PINGP Unit 2. To complete its review, the NRC staff requests the following additional information.

Enclosure 1 to the December 22, 2009 Submittal

E1-1. Pages 1 and 2, Item 4 states that the safety injection line consists of a 4-inch diameter pipe and a 6-inch diameter pipe. Item 5 states that the RCS draindown line consists of a 2-inch pipe reducer and a 6-inch diameter pipe.

(1) Discuss the approximate length of the 4-inch diameter and 6-inch diameter pipe segments in the safety injection line. Discuss the approximate length of the 2-inch diameter and 6-inch diameter pipe segments in the RCS draindown line.

(2) Discuss whether there are pipe whip restraints installed on the 4-inch diameter portion of the safety injection piping and on the 2-inch portion of the RCS draindown line.

(3) If no pipe whip restraints are installed on the 2-inch diameter portion of the draindown line and 4-inch diameter portion of the safety injection line, and if pipe whip restraints on the 6-inch diameter portion of these two lines are removed as a result of the LBB approval, justify how the 6-inch diameter portion of these two lines is protected, should the 2-inch portion of the draindown line or the 4-inch diameter portion of the safety injection line fail in a double-end guillotine break.

E1-2. Page 7. The licensee stated that when LBB was applied to the RCS loop piping in an LBB evaluation in 1986, a criterion of 1 gallon per minute (gpm) in one hour for RCS leakage was used for the leak detection system capability. However, for the current submittal, the licensee used a leakage detection limit of 0.2 gpm. The use of 0.2 gpm in the proposed LBB evaluation is an improvement in the leakage detection capability from the original licensing basis of 1 gpm. However, discuss whether the design basis for the RCS leak detection system needs to be changed in the PINGP Updated Final Safety Analysis Report and plant technical specifications via a license amendment process. If not, provide justification.

Enclosure 2 to the December 22, 2009 Submittal

E2-1. Page 1-1.

(1) Identify the material specification of the subject pipes and welds for LBB (e.g., SA designation).

(2) Provide material properties of the pipes and welds because Table 4-1a provides only materials properties for the lower-bound shield metal arc welding process.

(3) Identify any piping analyzed for LBB in Enclosure 2 that contains Alloy 82/182 dissimilar metal welds.

E2-2. Page 1-5. Table 1-1 presents 12 leak detection systems at PINGP with detectable leakage and response time. The licensee stated that PINGP has a very redundant leak detection system capable of detecting leakage as low as 0.1 gpm, but it is being conservative by using a leak detection capability of 0.2 gpm in the LBB analysis. The NRC staff questions the capability of the 12 detection systems and methods having the necessary redundancy and sensitivity to meet the specifications in Regulatory Guide (RG) 1.45, Revision 1. First, of the 12 detection systems and methods listed in Table 1-1, only five monitoring methods can detect a minimum leakage of 0.2 gpm or lower. Of the five monitoring methods, the operator inspection method, the daily coolant inventory method, and the sump pump operating time method would not satisfy the RG 1.45 requirement of a response time of 1 gpm within 1 hour. The remaining two monitoring methods may be acceptable. The containment radioactive particulate monitor R-11 has an estimated response time of 1 hour for a leakage rate of 0.5 gpm and it can detect a minimum of 0.1 gpm. The licensee may also take credit for the containment relative humidity monitoring which can detect a minimum leakage of 0.2 gpm with an estimated response time of 2 hours for a 0.5 gpm leakage.

(1) Of the 12 leak detection systems in Table 1, confirm which leak detection systems satisfy RG 1.45.

(2) On page 1-4, the licensee stated that Table 1-1 was taken from Reference 5, which is related to the PINGP coolant leakage detection system performance and was submitted to the NRC on March 31, 1976. The information in Table 1-1 is more than 30 years old. Identify the leakage detection systems and methods at Units 1 and 2 that satisfy RG 1.45, Revision 1, in terms of redundancy, reliability, and sensitivity per Standard Review Plan (SRP) Section 3.6.3.III.4.

(3) Provide the response time for the detectable leakage of 0.2 gpm because Table 1-1 presents response time based on the leakage of 0.5 gpm, 1.0 gpm and 5.0 gpm, and not 0.2 gpm.

E2-3. Page 4-5, Item 3. Explain why normal operating pressure is multiplied by 1.01 for the critical flaw size calculation.

E2-4. Page 4-6, Item 8.

(1) Explain why not all the nodes reported in the subject LBB evaluation have updated loading data due to the uprate conditions and why there is no stress analysis performed on the subject piping considering uprate conditions.

(2) Explain why node 1045 in the 6-inch diameter safety injection line in Table 4-10 was selected as the limiting node even though its leakage was not limiting as shown in Table 5-9.

E2-5. Not used

E2-6. Page 4-19. Clarify why the "normal operation and [safe shutdown earthquake] SSE" moment and leakage flaw size for node 1045 of the 6-inch diameter safety injection line in Unit 1 in Table 4-10 are not the same as the moment and leakage flaw size for node 1045 in Table 5-9.

E2-7. Page 5-3, second paragraph, and Figure 5-1. For the J-R power-law representation, discuss the crack extension that was used to calculate the toughness slope dJ/da.

E2-8. Pages 5-7 to 5-12. Explain why the stress corrosion cracking morphology in leak rate calculations is discussed in this report even though it does not appear that the subject pipes contain Alloy 82/182 weld material or Alloy 600 material.

E2-9. Page 6-1, second paragraph, states that "...Although there was a safety injection transient in Unit 1 due to [steam generator] tube rupture in 1979, there have been no inadvertent safety injections since. This transient is therefore also considered unlikely and was not evaluated..." The fact that a safety injection did occur in 1979 shows that the safety injection transient is a likely event and should be considered in the fatigue crack growth calculation. Justify why the inadvertent safety injection should not be considered in the evaluation, and discuss the actions/measures that preclude the potential for having an inadvertent safety injection.

E2-10. Page 6-1, second paragraph, states that "...There are no local piping system transients for the 6-inch draindown line and the 6-inch hot leg nozzles..."

(1) Explain why there are no local piping system transients for the 6 inch draindown line and the 6-inch hot leg nozzles. Discuss the transients that were used for these two lines in the analysis.

(2) Discuss whether there are local piping system transients or design transients applied to other pipes in the LBB evaluation.

(3) Explain the "local" piping system transients as opposed to the design basis transients or non-local transients.

(4) Discuss the thermal transients of the reactor coolant system draindown line and whether the thermal transients were included in the analysis.

E2-11. Pages 6-2 and 6-3. Section 6-2 discusses various stresses for crack growth evaluation. Explain why stresses due to seismic event were not discussed.

<u>Enclosure 3 to the December 22, 2009 Submittal</u>

E3-1. Page 1-1. The licensee stated that it has installed the weld overlay on the Alloy 82/182 dissimilar metal weld on the pressurizer surge line at PINGP Unit 2.

(1) Discuss whether the weld overlay is installed on the pressurizer surge line at Unit 1.

(2) Discuss inspection results of the overlaid Alloy 82/182 dissimilar metal weld(s) at Unit 2 pressurizer surge line.

(3) Provide the weld identification number for the Alloy 82/182 welds that have been weld overlaid.

E3-2. Page 4-2, Section 4.2.

(1) Discuss whether the weight of the weld overlay is included in the applied loads in the LBB evaluation. If not, provide justification.

(2) Clarify whether the loadings used in the LBB analysis are applicable for 60 years for the period of license renewal.

E3-3. Page 4-4, second paragraph. The licensee stated that the thermal stratification loads are lower than safe shutdown earthquake (SSE) load as shown in Table 4-2 of Enclosure 3; thus, thermal stratification during heatup/cooldown is ignored. Justify why the thermal stratification loads in Table 4-2 are ignored because the moment in the y direction for the thermal stratification is not insignificant compared to the SSE load. If the moments for the thermal stratification in the y direction (My) are included with the SSE load in the analysis, the critical crack size and leakage crack size may be changed from the reported value in the submittal.

E3-4. Page 7-1, first paragraph. The licensee stated that "...Crack growth evaluations were performed in Reference 6 to indicate that combined PWSCC and fatigue crack growth for axial and circumferential postulated flaws is within acceptable limits for [time period - proprietary information] operating interval..."

(1) Provide the acceptable limits and the starting point of the stated time period.

(2) Based on the results, the postulated flaw(s) will exceed the acceptable limits before the end of the plant license and license renewal period. Discuss how the subject pipe will be monitored to prevent flaws from exceeding the acceptable limits.

<u>Enclosure 4 to the December 22, 2009 Submittal</u>

E4-1. Page 2-1. Section 2.1 implies that stress corrosion cracking in the RCS primary loop and connecting Class 1 lines is a low probability event. The pressurizer surge line at Unit 2 contains Alloy 82/182 dissimilar metal welds, which are susceptible to primary water stress corrosion cracking (PWSCC) based on pressurized-water reactor (PWR) operating experience.

(1) In light of Alloy 82/182 welds in the surge line, explain why Section 2.1 stated that stress corrosion cracking is a low probability event and did not discuss the PWSCC issue in the pressurizer surge line. The NRC staff understands that Enclosure 3 of the submittal covers the Alloy 82/182 and PWSCC issue for the Unit 2 surge line. However, Enclosure 4 should also address the issue.

(2) Provide any prior occurrences of fatigue cracking or PWSCC in the Unit 2 pressurizer surge line.

E4-2. Page 2-2, Section 2.2. Provide quantitative information about historic frequencies on water hammers in Unit 2 pressurizer surge piping.

E4-3. Page 2-1, Section 2. Based on PWR operating experience, the pressurizer surge line is susceptible to thermal stratification, which is a form of thermal-induced fatigue. SRP Section 3.6.3.III.10 does not permit LBB to be applied to piping with a history of fatigue cracking or failure. The licensee did not discuss thermal stratification in Section 2.0. Discuss whether thermal stratification is a concern in the pressurizer surge line at PINGP Unit 2. If not, provide technical basis.

E4-4. Pages 4-1 to 4-5. Section 4.4 states that load cases A, B, and C are normal operation conditions and D, E, F, and G are faulted conditions. There should be 12 loading combinations of normal operation conditions and faulted conditions: A/D, A/E, A/F, A/G, B/D, B/E, B/F, B/G, C/D, C/E, C/F, and C/G.

Explain why load combinations A/E, A/G, B/D, C/D, C/E, and C/F were not shown in Table 4-3.

E4-5. Page 4-5, Table 4-2.

(1) Explain how the temperatures are derived in each of the load cases in Table 4-2.

(2) Discuss whether loads caused by insurge and outsurge have been considered in the LBB evaluation. If not, provide justification.

E4-6. Page 4-7, Table 4-4 provides loading for critical location, Node 1320.

(1) Discuss how the critical location, Node 1320, was selected.

(2) To aid in necessary confirmatory calculations, provide all load components (i.e., moments in the x, y, and z directions and Fx) for each ASME loading category case (A, B, C and D) for Node 1320.

(3) Discuss whether the pipe loadings for the LBB evaluation include the effect of the power uprate conditions. If not, provide justification.

E4-7. Page 5-3, first paragraph, states that the crack relative roughness was obtained from fatigue crack data of stainless steel samples. Discuss the source of the stainless steel samples. Discuss how the roughness value was obtained.

E4-8. Page 5-3. Section 5.2.3 states that the crack opening area was estimated using the method of Reference 5-3. Discuss in detail exactly how the crack opening area was estimated and provide page numbers in Reference 5-3 which show the crack opening area calculation.

E4-9. Pages 5-11 to 5-14. Discuss how the curves on these pages were constructed.

E4-10. Page 6-1. The licensee did not perform a fatigue crack growth calculation for the Unit 2 pressurizer surge line. Instead, it used the results of the Unit 1 fatigue crack growth calculation to apply to the Unit 2 fatigue growth calculation. Unit 1 selected location 1 is near the reactor coolant loop nozzle and location 2 is located near the pressurizer nozzle.

(1) Discuss whether the same locations in the Unit 2 surge line have the same loading as the two locations in the Unit 1 surge line.

(2) Discuss how it was determined that the two pipe locations in the Unit 1 pressurizer surge line will be the same limiting locations in the Unit 2 pressurizer surge line.

(3) Discuss whether the applied loads and stresses at the Unit 1 surge line bound the applied loads at the Unit 2 surge line.

E4-11. Page A-1. Cite reference(s) for the equations presented in Appendix A.

June 10, 2010

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/RA/

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
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Docket Nos. 50-282 and 50-306

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