PrairieIslandNPEm Resource

From: Vincent, Robert [Robert.Vincent@xenuclear.com]
Sent: Wednesday, June 24, 2009 5:44 PM
To: Plasse, Richard; Goodman, Nathan
Cc: Eckholt, Gene F.
Subject: PINGP Letter Responding to Refueling Cavity Leakage RAIs and Revising Vessel Internals Program Commitment
Attachments: 20090624 Response to Follow-up RAI B2.1.38 & Commitment Change.pdf; 20090624 Response to Follow-up RAI B2.1.38 & Commitment Change.doc

Attached are pdf and WORD copies of a letter responding to the latest Refueling Cavity leakage Follow-up RAIs and changing the PWR Vessel Internals Program as we discussed in the June 10 telecon. The letter was signed out today.

Let me know if you have any problems with the files.

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Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Response to NRC Request for Additional Information Regarding Application for Renewed Operating Licenses

By letter dated April 11, 2008, Northern States Power Company, a Minnesota Corporation, (NSPM) submitted an Application for Renewed Operating Licenses (LRA) for the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2. During the Aging Management Audit, the NRC was briefed on water seepage from the refueling cavity into the containment sumps that had been detected during refueling outages. In a letter dated November 5, 2008, the NRC issued RAI AMP-B2.1.38-2 regarding that seepage, and PINGP responded on December 5, 2008. The matter was discussed in a public meeting on March 2, 2009. An additional Follow-up RAI was issued on March 31, 2009, and the PINGP response was provided on April 6, 2009. On May 28, 2009 the NRC visited the PINGP site to review documents related to refueling cavity leakage. On June 4, 2009, the NRC issued the Safety Evaluation Report With Open Items Related to the License Renewal of the Prairie Island Nuclear Generating Plant Units 1 and 2 (SER). The SER identified the refueling cavity seepage as Open Item 3.0.3.2.17-1, pending completion of the NRC review of the April 6, 2009, Follow-up RAI response. Subsequently, in a letter dated June 10, 2009, an additional Follow-up RAI was issued regarding that seepage. The PINGP response to that Follow-up RAI is provided in Enclosure 1.

As a separate matter, in a conference call on June 10, 2009, the PINGP PWR Vessel Internals Program was discussed. During that call, PINGP agreed to clarify Part B of the associated License Renewal Commitment No. 25 to indicate that the vessel internals inspection plan submittal would also include any LRA changes to the scoping, screening, and AMR results, and the description of the PWR Vessel Internals Program, that are necessary to reflect the final NRC-approved Inspection and Evaluation guidance. Accordingly, License Renewal Commitment No. 25 is being revised as noted below to incorporate this clarification. A complete listing of PINGP License Renewal Commitments, updated to reflect NSPM correspondence to date, is provided in Enclosure 2.

If there are any questions or if additional information is needed, please contact Mr. Eugene Eckholt, License Renewal Project Manager.

Summary of Commitments

This letter contains no new commitments. License Renewal Commitment No. 25 is revised to read as follows:

A. A PWR Vessel Internals Program will be implemented. Program features will be as described in LRA Section B2.1.32. The program will be implemented prior to the period of extended operation.
B. An inspection plan for reactor internals will be submitted for NRC review and approval at least twenty-four months prior to the period of extended operation. In addition, the submittal will include any necessary revisions to the PINGP PWR Vessel Internals Program, as well as any related changes to the PINGP scoping, screening and aging management review results for reactor internals, to conform to the NRC-approved Inspection and Evaluation Guidelines.

The implementation schedules for Parts A and B of this commitment are unchanged: 8/9/2013 (Unit 1) and 10/29/2014 (Unit 2) for Part A, and 8/9/2011 (Unit 1) and 10/29/2012 (Unit 2) for Part B.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on June 24, 2009.

Michael D Wadley

Michael D. Wadley
Site Vice President, Prairie Island Nuclear Generating Plant Units 1 and 2
Northern States Power Company - Minnesota

Enclosures (2)

cc:

Administrator, Region III, USNRC
License Renewal Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
Prairie Island Indian Community ATTN: Phil Mahowald
Minnesota Department of Commerce

Enclosure 1
Response to Follow-up RAI B2.1.38

NRC Follow-up RAI B2.1.38

In letter L-PI-08-098, dated December 5, 2008, "Responses to NRC Requests for Additional Information Dated November 5, 2008 Regarding Application for Renewed Operating Licenses," the applicant submitted responses to the staff's RAIs. In addition, the applicant provided information during a public meeting on March 2, 2009, and a previous follow-up RAI in letter L-PI-09-047, April 6, 2009.

On May 28, 2009, the NRC staff performed an audit at the Prairie Island Nuclear Generating Station (PINGP), Units 1 and 2, related to the supporting documentation for the reactor cavity leakage. The staff reviewed the following documents:

References:
1. Dominion Energy Incorporated, "Evaluation of Effects of Borated Water Leaks on Concrete Reinforcing Bars and Carbon Steel Plate of the Containment Vessels at Prairie Island Units 1 and 2," Report No. R-4448-00-01, Rev. 0.
2. Prairie Island Nuclear Generating Station, "Refueling Cavity leakage, Event Date 1988-2008," Report No. RCE 01160372-01, Volumes 1 and 2.

In order to complete our review of issues related to the PINGP reactor cavity leakage discussed in the above referenced reports, the staff requests the following additional information:

[Note: To assist the reader, the NSPM response to each part (a through i) of the RAI is provided immediately after the statement of each part. References applicable to all responses are listed after the response to part i.]

Follow-up RAI Part a

a) Section 2.2 of Reference 1 recommends that a test be performed to determine if there is high assurance that the pH present in the water between the containment steel plates and concrete is more than 12.5. Please provide a schedule for performing this test.

NSPM Response to Part a

It is well established that boric acid will be neutralized by contact with concrete. Tests of the effects of boric acid in contact with concrete have shown a rapid rise in pH of the boric acid solution. (Page 3-2 of Reference 5) It has also been shown by calculations using the EPRI MULTEQ program that the equilibrium pH will be about 12.5 with excess quantities of calcium present, as would be the case with the large amounts of calcium hydroxide in concrete. Evidence that neutralization of boric acid has occurred in concrete at Prairie Island is provided by the pH values of 7 and 7.8 that have been measured in the active leakage. (Page 4-2 of Reference 1) To provide additional confirmation of this behavior, a simple laboratory test has been performed, as discussed below.

Enclosure 1
Response to Follow-up RAI B2.1.38

To provide background for the process which results in boric acid being neutralized when in contact with concrete, the materials that form concrete and their chemical properties are summarized below.

Concrete is a composite material consisting of a binder (cement paste) and a filler of fine and/or course aggregate particles that combine to form a synthetic conglomerate. The cement is a mixture of compounds made by grinding crushed limestone, clay, sand, and iron ore together to form a homogeneous powder that is then heated at very high temperatures to form a clinker. After the clinker cools, it is ground and mixed with a small amount of gypsum to regulate setting and facilitate placement. This produces the general-purpose portland cement that is mixed with water to produce cement paste that binds the aggregate particles together. Portland cements are composed primarily of four chemical compounds: tricalcium silicate, dicalcium silicate, tricalcium aluminate, and tetracalcium aluminoferrite. The calcium silicate hydrates constitute about 75% of the mass. (Section 3, Reference 4)

The hardened cement paste consists mainly of calcium silicate hydrates, calcium hydroxide, and lower proportions of calcium sulphoaluminate hydrate. About 20% of the hardened cement paste volume is calcium hydroxide. The pore solution is normally a saturated solution of calcium hydroxide within which high concentrations of potassium and sodium hydroxides are present. (Section 3, Reference 4)

Hardening of concrete occurs as a result of hydration, which is a chemical reaction in which the major compounds in the cement form chemical bonds with water molecules and become hydrates. Since cement is the most expensive ingredient in concrete, it is desirable to utilize the minimum amount necessary to produce the desired properties and characteristics. Aggregate typically occupies 60 to 75% of the volume of concrete, with the balance of the concrete mix generally consisting of 10 to 15% cement, 15 to 20% water, and 5 to 8% air, if entrained. (Section 3, Reference 4)

Tests reported in Reference 3 indicate that water of neutral pH placed in small holes drilled into concrete reaches an equilibrium pH of about 12.8 to 13.3 after one to two weeks. This is consistent with general industry information that indicates that pore water in concrete generally has a pH of about 12.5 or higher. This pH is fully protective of rebar (Section 4.3.2 of Reference 4). These results indicate that normal fresh water in contact with concrete will reach an equilibrium pH of 12.5 or more and be protective of steel (i.e., result in insignificant corrosion). However, these results do not address the possible effects of the boron in the water on the pH. This is discussed further in the following paragraphs.

In order to more firmly establish the pH that will develop in small volumes of borated water in contact with large amounts of concrete, a simple laboratory test was performed. The test was performed by adding chemicals representative of those in concrete to an open beaker with a volume of one liter of deionized water at room temperature. After each chemical addition, the solution was stirred and the pH was measured. The steps of the test and the pH values measured are shown in Table 1. Comments regarding the solutions tested and the results are as follows:

- Solution 1 involved the addition of calcium oxide alone, and represents the case of water in contact with the calcium hydroxide that is present in large quantities in concrete. Calcium oxide forms calcium hydroxide when dissolved in water. The measured pH of 12.05 was slightly below the normally reported equilibrium value of 12.5 for a saturated solution of calcium hydroxide, which is expected to contain approximately 1,000 ppm calcium ions (as calcium hydroxide). The slightly lower-than-expected pH is attributed to the fact that full equilibrium had not been reached when the pH measurement was made (i.e., the pH was measured before the calcium oxide was completely dissolved).

- Solution 2 reflects the addition of sodium, albeit at lower concentrations than calcium. The concentration of sodium added to the test solution was consistent with that anticipated based on information reported in Reference 3 and is consistent with Reference 4 which notes that high concentrations are present in pore water. As shown in Table 1, the sodium increased the measured pH by a small amount.

- Solution 3 reflects the addition of the equivalent of 3000 ppm boric acid, which is representative of the concentration present during refueling. The measured pH of 8.85 is in the expected range indicated by calculations for a case where some boron remains in solution.

- Solutions 4 through 8 reflect the addition of increasing amounts of calcium oxide. This simulates exposure over time to the excess amounts of calcium hydroxide that are present in the concrete. Initially, the excess boric acid present in solution buffers the solution pH, and calcium oxide additions have only a small effect on the resulting pH. However, the pH increases rapidly to the 12.3 range after the concentration of calcium oxide exceeds the stoichiometric amount needed to react with the boric acid that was initially present.

Table 1 pH of Simulated Boric Acid Leakage in Contact with Chemicals From Concrete

Solution	Mass (g)				pH
	CaO	ppm Ca	NaOH	H_3BO_3	
1	1.3992	1,000			12.05
2	1.3992	1,000	3.3061		12.39
3	1.3992	1,000	3.3061	17.1585	8.85
4	6.3992	4,573	3.3061	17.1585	8.94
5	11.4016	8,149	3.3061	17.1585	9.01
6	24.7441	17,684	3.3061	17.1585	9.74
7	29.7456	21,259	3.3061	17.1585	12.28
8	34.7469	24,833	3.3061	17.1585	12.30

The primary conclusion that can be drawn from the test results summarized in Table 1 is that a high protective pH is reached by borated water that is in contact with excess amounts of calcium oxide. This indicates that a similar high pH will develop in borated water trapped between the steel containment vessel and the concrete since there are excess amounts of calcium hydroxide in the concrete.

The test discussed above was performed in accordance with written instructions, and the results were documented following normal laboratory practice. However, it was not performed in accordance with formal nuclear QA requirements. Nevertheless, the results are considered to be consistent with theoretical values and provide strong supporting evidence that high protective pH values will be reached in borated water trapped between the steel containment vessel and the concrete.

Follow-up RAI Part b

b) Section 2.2 of Reference 1 recommends removal of concrete inside the containment at the following locations:

   i. Sump C
   ii. Through the wall at elevation 695 closer to the transfer tube

   However, Northern States Power Company, Minnesota (NSPM) in a letter dated April 6, 2009, committed to remove concrete from Sump C only. Please clarify. In addition, please explain why removal of concrete from Sump C is not planned during the next scheduled outages at PINGP, Units 1 and 2.

NSPM Response to Part b

In reviewing the recommendations from Reference 1 to determine the appropriate Corrective Actions to be assigned within the Corrective Action Program, engineering management considered the value, need and sufficiency of each recommendation provided. The review concluded that there is limited value in removing concrete from the inside diameter of the containment vessel at the 697’ floor elevation, as it is not known whether this area is wetted and has a potential for corrosion to exist. The site has instead removed the grout in the RHR suction sump of both units as these areas are lower in containment elevation and consistently show wetting when refueling cavity leakage occurs. It is also believed the RHR sumps would be more likely than the 697' elevation to show any corrosion due to repeated wetting and close proximity to ambient oxygen. In addition, much of the area between the transfer tube at the 715’ elevation and annulus floor at the 706’ elevation can be monitored by ultrasonic thickness measurement from the exterior of the containment vessel, further diminishing the value of removing concrete from the interior wall. Therefore, the Corrective Action assignments did not include removal of concrete at elevation 697'.

Removal of the concrete in sump C (under the reactor vessel) will be performed in the next refueling outages following the outages during which the refueling cavity liners are repaired. This is primarily for logistical reasons. The estimated thickness of the concrete at the thinnest location in the floor of sump C is 16 inches with reinforcing bar both near the top of the proposed excavation and near the containment vessel inside diameter. The work area is relatively small and in close proximity to the reactor vessel thimble tubes which provide an ASME Class 1 pressure boundary. As such, performing the excavation safely requires considerable planning and specialized tooling. The site will use the upcoming refueling outage in each Unit to survey the excavation sites, and the time between outages to plan the excavations and secure the appropriate tools.

Follow-up RAI Part c

c) Section 4.2 of Reference 1 has identified an upper bound loss of 0.25 inch in the 1.50 inch steel containment due to borated water corrosion over a 36 year period. Please advise if the stresses in the steel containment remain within the American Society of Mechanical Engineers Code allowable values for this loss of 0.25 inches. According to Section 4.1 of Reference 1, minimum thickness required for the steel containment for all loading conditions is 1.4908 inches. In addition, please clarify if NSPM has considered the potential of continued reactor cavity leakage over the life extension period of 60 years.

NSPM Response to Part c

The evaluation estimates the likely corrosion of the containment vessel to date at no more than 0.010". This estimate accounts for the neutralization of borated water in concrete. Recent ultrasonic thickness measurements, including measurements of known wetted areas in the RHR suction sump, showed no corrosion with all thickness readings above the nominal plate thickness. If any significant loss were identified, an ASME code evaluation would be required.

As discussed in section 4.2 of Reference 1, the 0.25" value of corrosion assumes continuous wetting with aerated, concentrated, boric acid over a period of 36 years. This value does not consider the buffering effect of the concrete or the consumption of oxygen dissolved in the water. Therefore, the long term environment that could lead to this level of corrosion would not exist. The report's reference to 0.25" as an upper bound does not clearly convey its meaning. This value was provided for comparison purposes only to provide further support for the low corrosion rates expected, and does not represent an expected condition in a PINGP containment vessel. Therefore, an ASME Code analysis of the containment vessel which assumes loss of 0.25" of vessel wall has not been performed. Reference 1 does not suggest that a wall loss of this magnitude would leave the vessel capable of meeting code allowables. Indeed, the report states that any observed wall loss that reduced the vessel below the nominal 1.5" thickness would have to be evaluated in accordance with ASME Section XI. Reference 1 only provides the judgment that even with a wall loss of 0.25", the containment vessel would still be able to withstand accident pressure without a loss of containment integrity.

The areas of the steel containment vessel that are potentially subject to borated water exposure from refueling cavity leakage are the bottom head and sections of the shell behind concrete at the end of the refueling cavity and transfer pit (referred to in the

USAR as the “cold spot”). Both the shell and bottom head are fabricated from SA-516-70 material with a minimum tensile strength of 70 ksi. The Pioneer Service & Engineering Company containment vessel stress report shows the shell and bottom head were designed in accordance with ASME section VIII with a design pressure of 41.4 psig and a corresponding required thickness of 1.5 inch.

In accordance with ASME section VIII the design allowable membrane stress is limited to 25% of the material minimum tensile strength, or 17.5 ksi. Allowable stresses during a Design Basis Accident (DBA) are considerably higher. As indicated in USAR table 12.2-22, total stresses under a DBA with Design Basis Earthquake range from 24.48 ksi to 27.86 ksi. These stresses are approximately one half the DBA allowable stress of 52.5 ksi. Stresses are generally proportional to thickness. As such, even with thinning of 0.25 inch of the 1.5 inch shell thickness, total stresses would still be well below the stress limits for the load combinations that combine stresses from a DBA with those from a Design Basis Earthquake.

The site fully expects that leakage will be stopped during the next refueling outage of each unit. During the outage following the outage of repair, any water observed in Sump C will be evacuated. However, any residual water behind concrete that may not be able to be evacuated would have a very small stagnant volume with its pH elevated by the alkalinity in concrete. Any potential corrosion in such regions would be similar in magnitude to (or lower than) the 0.010" conservatively estimated for 36 years to date. With this level of corrosion, the overall conclusion remains valid that containment vessel integrity would be unaffected.

The site will continue to monitor the containment vessel and internal structures through the ASME section XI, IWE program and the Structures Monitoring program. If any new leakage is identified that indicates the refueling cavity leakage has recurred, the issue will be entered into the Corrective Action Program for evaluation and identification of corrective actions.

Follow-up RAI Part d

d) In Section 5.2.3 of Reference 1, the rate of degradation estimated for PINGP concrete is two times that used previously for Salem/Connecticut Yankee plants to account for the difference in the type of concrete aggregates at PINGP. Please advise if NSPM has performed or intends to perform any tests to confirm the use of this assumption for the degradation rate.

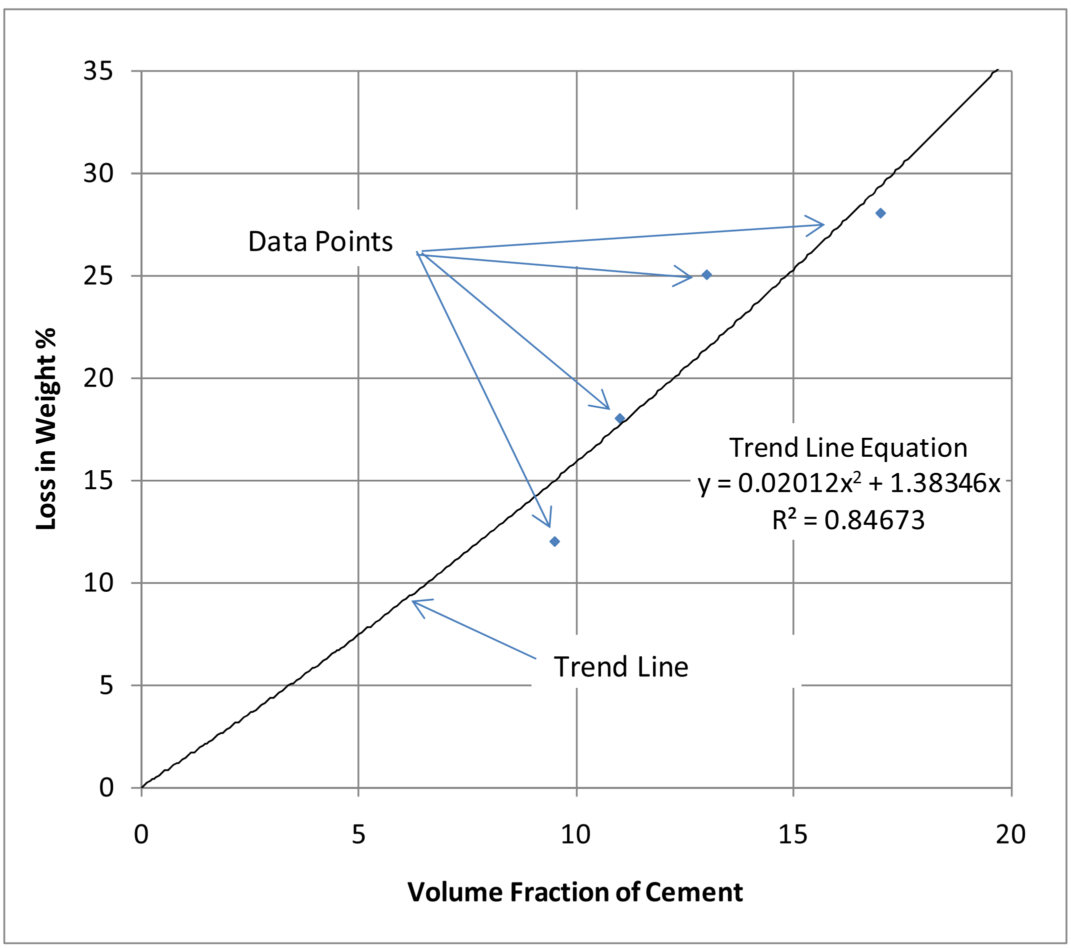
NSPM Response to Part d

The acceleration factor of 2.0 for the increased rate of degradation resulting from the presence of about 5% carbonate-based aggregate at Prairie Island relative to the 0% carbonate-based aggregate at the cited plants was based on engineering judgment.

Tests of the effects of acids on reinforced concrete show that acids weaken concrete by dissolving cement and carbonate-based aggregate (Page 29 of Reference 4 and page 4 of Reference 6). Tests discussed in Reference 7 indicate that the weight loss experienced by cubes of concrete immersed in acid increases as the volume fraction of cement in the concrete increases. These tests were performed on concrete that had non-soluble aggregate (i.e., only siliceous aggregate and no calcium carbonate-based aggregate). This increased weight loss is attributed to the fact that, as the volume fraction of cement increases, the volume fraction of the concrete that is soluble in acids increases. Based on this result, it is reasonable to assume that the effect of boric acid on concrete containing soluble aggregates will follow a similar pattern; i.e., the degradation of concrete due to boric acid will increase as the total fraction of material in the concrete that is soluble in acids increases, whether that soluble material is cement or aggregate.

Quantitative data in Reference 7 for weight loss of concrete cubes when exposed to an acid concentration of 3% is shown in Figure 1. A trend line is shown on the figure that provides an equation for quantifying this dissolution behavior. As can be seen, the weight loss increases somewhat more strongly than linearly as the volume fraction of cement, (i.e., of soluble material), increases.

Figure 1 Weight Loss in Acid vs. Volume Fraction of Cement (based on data from Reference 7)



A footnote on page 10 of Reference 4 indicates that concrete normally contains 10 to 15% of cement, or an average of about 12.5%. As noted on page 2-2 of Reference 1, the amount of carbonaceous aggregate in the concrete used at Prairie Island was about 5%. Adding this 5% to the average 12.5% cement results in a total of 17.5% soluble material in the concrete at Prairie Island as compared to the 12.5% in a concrete with all igneous (no carbonate) based aggregate such as was used at the cited plants. (Section 8.1.3 of Reference 8) Using the trend line equation shown in Figure 1, increasing the fraction of soluble material (cement in the figure, cement plus soluble aggregate in this case) from 12.5% to 17.5% increases the weight loss by a factor of 1.49. This indicates that the assumed factor of 2.0 increase in severity of acid attack to account for the presence of 5% carbonate-based aggregate is conservative by a significant margin (2 vs. 1.49).

Based on this result, it is considered that there is no need for additional tests to evaluate the effects of the carbonate-based aggregate used at Prairie Island.

Follow-up RAI Part e

e) Section 5.2.6 of Reference 1 states, "…concrete is not relied upon for tensile strength (tensile strength provided by rebar)." Please explain how formation of large thru thickness cracks will affect the transverse shear capacity of concrete slabs and walls. Shear strength of concrete is directly related to the tensile strength.

NSPM Response to Part e

The nominal shear strength of reinforced concrete is based on the combination of the nominal shear strength provided by the concrete and the nominal strength provided by shear reinforcement (steel rebar). If a reinforced concrete member were postulated to have a large idealized crack along the entire shear plane, the reinforcement would need to carry the shear force. However, there is no indication that such a crack might exist at PINGP.

The existence of a wide crack in the refueling cavity floor or walls that has traveled through the entire thickness of a concrete section along a single shear plane is highly unlikely. Plant operating experience has not identified any large cracks in either the Unit 1 or Unit 2 reactor containment concrete structures. The cracks that have been identified are often characterized as hairline or normal shrinkage cracks. Large through-member cracks are not reasonable to postulate at PINGP based on the plant's adherence to good design and construction practices. The refueling pool floor slab, at its minimum, is 4'-0" thick (3'-8" of reinforced concrete and 4" of leveling grout) and contains both top and bottom reinforcement in each direction. Where water seepage from cracks has occurred during flood up of the refueling cavity, the cracks have never been described as large or wide, which is consistent with the small amount of water that seeps from the cracks, estimated at no more than 1-2 gallons per hour. Additionally, no evidence of significant washout of material has been identified, a condition that could be indicative of the development of a widening crack.

Follow-up RAI Part f

f) Section 5.2.6 of Reference 1 states, "Degradation of concrete by exposure to borated water can also occur at cracks in the concrete. This could lead to loss of strength of concrete in a narrow band through thickness of material." A crack across thickness of material may expose rebars to corrosion. Please explain how this effect is considered in determining the structural integrity of concrete walls and slabs.

NSPM Response to Part f

The quotation above is from Section 5.2 of Reference 1, "Degradation of Concrete Due to Chemical Attack." This section deals with concrete degradation, and does not address rebar corrosion. Rebar corrosion is addressed in the following section, Section 5.3, "Corrosion of Rebar Caused by Exposure to Borated Water." The specific question asked in Part f is addressed by the second bullet in Section 5.3 of Reference 1. This section indicates that the dissolution of calcium hydroxide from the concrete around rebar at cracks in the concrete might be assumed to develop conditions that might lead to increased rates of corrosion of the rebar. However, tests performed for the cited plants and other tests described in the open literature indicate that corrosion in such situations has been negligible, even when the low pH borated water reaching the cracks was regularly refreshed. It appears that conditions at the rebar remain sufficiently alkaline in such situations to passivate the surface, despite the presence of refreshed borated water.

Reference 5 documents tests performed using steel coupons in a refreshed boric acid solution. The relevant conclusion from the report reads as follows: "The short-term corrosion rate of reinforcing steel was observed to be about 4.2 mils/year (0.0042-inch/year) based on an exposure time of up to 56-days in an aerated boric solution with a pH in the range of 5.2 to 6.4. The corrosion rate should decrease as pH increases and oxygen content decreases under long-term conditions in the field." The tests documented in Reference 9 were two year tests of samples using refreshed boric acid in which the rebar was exposed to the boric acid at cracks. The main conclusion was as follows: "For a penetration period of 2 years, there were weakenings of the cross section of the reinforcing bars not worth being mentioned by reinforcement corrosion in separation cracks penetrated by deionised water (neutral, pH 7.0) and boric acid treatment deionised water of the pH-values 5.2 and 6.1 with crack widths of up to approximately 0.4 mm."

The tests documented in Reference 9 are considered to be the most relevant since they were long term (2 year) and used realistic cracked specimens with the rebar exposed to the boric acid in a crack. As noted in Reference 9, the corrosion of the rebar was "not worth being mentioned." The total exposure time of rebar cracks in the reinforced concrete under the refueling cavity at Prairie Island is conservatively estimated as 15 days per outage for 25 outages or 375 days or 1.03 years. This is significantly less than the two year test period of Reference 9 which resulted in no significant corrosion. Accordingly, it is concluded that no significant corrosion of rebar at cracks in the structural concrete under the refueling cavity has occurred at Prairie Island, and that effects of rebar corrosion at cracks in the concrete on structural integrity are insignificant.

Follow-up RAI Part g

g) Section 5.2.3 of Reference 1 estimated the upper bound loss of concrete depth over a 36 year period to be 0.31 inches. Please address the impact of this loss of 0.31 inch of concrete behind the stainless steel liner plate on the load carrying capacity of the stainless steel liner plate.

NSPM Response to Part g

As discussed in section 5.2.3 of the Reference 1, the upper bound loss of 0.31 inch is conservative as it assumes leakage every outage for 25 outages over a period of 36 years. There is no evidence of leakage prior to 1987 and leakage has only been observed in about half the outages due to intermittent successful leak mitigation with caulking or spray-on liner.

The impact of a potential loss of 0.31" of concrete on the load carrying capacity of the liner is minimal. The liner is effectively a membrane that is backed under the bottom and around the sides with concrete that is generally four to five feet thick. As a result, the impact on the load carrying capacity of the refueling cavity pool structure is negligible as the material loss is, at most, less than 1% of the concrete thickness. Large areas of washout or dissolution under the liner are not expected as leak rates are small (on the order of 1-2 gallons per hour) and only occur while the refueling cavity is flooded. However, even if large areas of washout or dissolution on the order of square feet and 0.31" depth were to occur, they would only be expected to result in shallow depressions in the liner and not in a failure such as tearing of the liner plate. The liner was constructed with the seams welded to stainless steel structural shapes embedded in the concrete. As such, the seams are reinforced and are not subject to loss of concrete directly under a seam. The liner plates are 3/16" and 1/4" thick Type 304 stainless steel. Stainless steel is generally very ductile and can withstand significant elongation and deformation before failure. The site has not experienced any observable depressions on either the floors or walls of the refueling cavities.

Follow-up RAI Part h

h) NSPM in a letter dated April 6, 2009, committed to visual inspections during the consecutive refueling outages of the areas where reactor cavity leakage has been observed following refueling cavity leak repairs in each unit. Which aging management program (AMP) will be used to address this issue?

NSPM Response to Part h

License Renewal Commitment Number 42 from the April 6, 2009 letter reads as follows:

> During the two consecutive refueling outages following refueling cavity leak repairs in each Unit (scheduled for refueling outages 1R26 and 2R26), visual inspections will be performed of the areas where reactor cavity leakage had been observed previously to confirm that leakage has been resolved. The inspection

results will be documented. If refueling cavity leakage is again identified, the issue will be entered into the Corrective Action Program and evaluated for identification of additional actions to mitigate leakage and monitor the condition of the containment vessel and internal structures.

The inspections of the locations which previously showed signs of refueling cavity leakage are special inspections assigned as corrective actions within the Corrective Action Program. The inspections will invoke the methodology, documentation requirements and acceptance criteria of the Structures Monitoring Program. The Structures Monitoring Program is the appropriate program since the locations where leakage has been observed are in containment interior structures.

Thereafter, as discussed in the April 6, 2009, letter, general monitoring for leakage and degradation will be performed in all areas of containment in accordance with the ASME Section XI, Subsection IWE Program and the Structures Monitoring Program. In addition, the Boric Acid Corrosion Program and the 10 CFR Part 50, Appendix J Program include inspections inside containment. These programs contain provisions to identify, evaluate, and correct degraded conditions prior to loss of function, ensuring the effects of aging for plant SSCs are adequately managed. Periodic visual inspections are performed, and inspection schedules are prescribed, ensuring timely identification of any degraded condition. The ASME Section XI, Subsection IWE Program also incorporates the requirements of 10 CFR 50.55a(b)(2)(ix) for the examination of metal containments, including the requirement in 10 CFR 50.55a(b)(2)(ix)(A) to evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in degradation to, such inaccessible areas.

During the current license period as well as the period of extended operation, degradation identified by these programs will be entered into the PINGP Corrective Action Program for evaluation and identification of corrective actions.

Follow-up RAI Part i

i) Page 52 of Reference 2 identified five recommendations to address reactor cavity leakage. Please provide NSPM's action plan and schedule for completing these five recommendations.

NSPM Response to Part i

Page 52 of the Root Cause Evaluation (RCE) recommended a repair plan that includes the following steps:

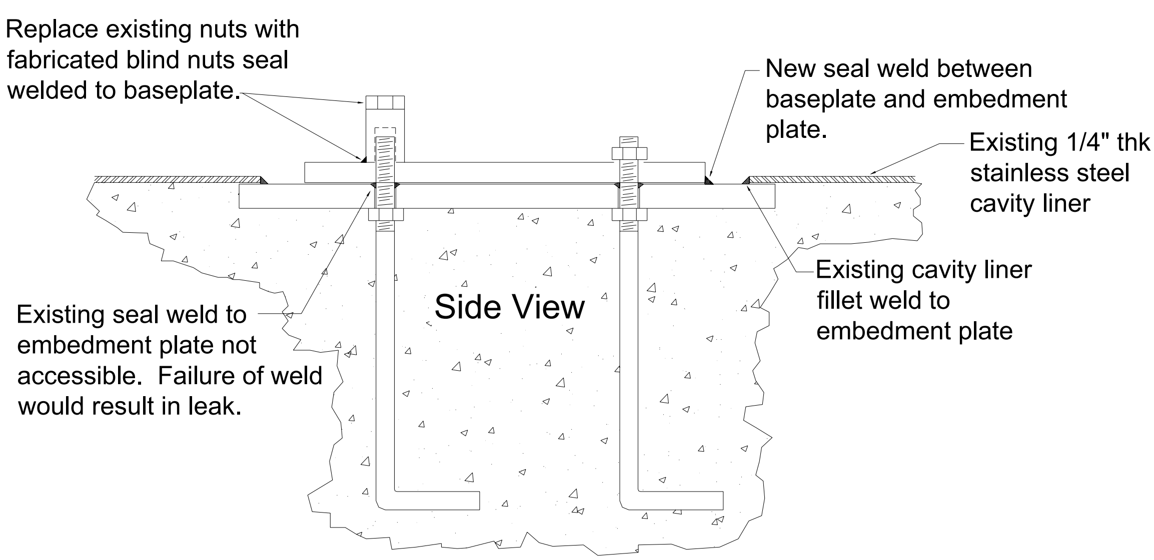
1. Unbolt and set aside all mechanically fastened fixtures (RCC Change Fixture, internals stands, and guide tube supports).
2. Vacuum Box penetrations and embedment plates to locate existing leaks. Weld repair and vacuum box completed welds
3. Preemptively seal weld and vacuum box all penetrations.

4. Vacuum Box, and/or PT weld seams and repair as needed to ensure no leakage due to stress corrosion cracking.
5. Pressure test or PT transfer tube bellows attachment welds and weld repair as needed.

The RCE also states that alternate approaches can also be considered provided the final repair plan permanently and completely mitigates future leakage.

The RCE concluded that the most likely refueling cavity leakage points are where the anchor studs for the reactor internals stands and the Rod Control Cluster (RCC) Change Fixture penetrate the associated embedment plates. The studs for these fixtures pass through holes in the embedment plates and are seal welded on the underside for the internals storage stands, and on top (then ground flush) for the RCC Change Fixture. Pinhole leaks or small cracks in one or more of the seal welds would result in a leak path through the embedment plate along the threads of the studs, allowing water under the cavity liner.

The plant management review of the Root Cause Evaluation considered the value, need and sufficiency of each recommendation. The current repair plan that emerged from these reviews is to replace the existing anchor nuts with new fabricated blind nuts that are seal welded to the existing baseplates. In addition, the existing baseplates will be seal welded to the embedment plates. The new welds will be readily accessible for examination to ensure the existing anchors and baseplates are leak tight. The following sketch depicts a typical RCC change fixture support showing the planned repairs. Similar repairs will be made to the internals stands supports.



General Arrangement of Change Fixture Supports

This repair strategy effectively accomplishes the goals of recommended steps 1 through 3, the intent of which is to permanently repair refueling cavity leakage, but in a more focused and dose-effective manner. The guide tube supports which are mounted to the refueling cavity wall were concluded to be unlikely sources of leakage and will not be sealed in a similar manner. Although the guide tube supports are of similar construction, past success at mitigating leakage by sealing the supports of the internals stands and RCC change fixture suggests that leakage occurs only at those locations.

Recommended step 4 is to vacuum box and/or dye penetrant test refueling cavity liner weld seams to ensure no leakage due to stress corrosion cracking. The site performed vacuum box and/or dye penetrant test of accessible seams of both units in 1998 and 1999 with no indications of cracking. NSPM does plan to examine a sample of accessible seams to ensure no cracking has occurred since the last inspection.

Recommended step 5 is to pressure test or dye penetrant test the transfer tube bellows attachment welds and perform weld repairs as needed. NSPM believes these welds are leak tight as past refueling cavity leakage has been completely mitigated through sealing only the internals stands and change fixture anchors. As a precaution, NSPM does plan to perform pressure testing or dye penetrant testing of accessible portions of the transfer tube bellows welds to preclude the possibility of water leaking along the transfer tube to the inside surface of the containment vessel.

All actions discussed above are planned for the next refueling outage for each unit, currently scheduled for fall 2009 for Unit 1 and spring 2010 for Unit 2.

References for NSPM Responses to Follow-up RAI B2.1.38 Parts a through i

1. Dominion Engineering, Inc. report “Evaluation of Effects of Borated Water Leaks on Concrete Reinforcing Bars and Carbon Steel Plate of the Containment Vessels at Prairie Island Units 1 and 2,” report R-4448-00-01, Rev. 0.
2. Prairie Island Nuclear Generating Station, “Refueling Cavity leakage, Event Date 1988-2008,” Report No. RCE 01160372-01, Volumes 1 and 2.
3. A. A. Sagües, et al., “Evolution of pH During In-Situ Leaching in Small Concrete Cavities,” *Cement and Concrete Research*, Vol. 27, No. 11, pp. 1747-1759, 1997.
4. D. J. Naus, *Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures - A Review of Pertinent Factors*, NUREG/CR-6927 - ORNL/TM-2006/529, Feb. 2007.
5. MPR Associates report “Boric Acid Attack of Concrete and Reinforcing Steel,” MPR-2634, Revision 2, February 2009.
6. B. Kerkhoff, “Effects of Substances on Concrete and Guide to Protective Treatments,” Portland Cement Association, Item Code: IS001, 2007.
7. N. I. Fattuhi and B. P. Hughes, “Ordinary Portland Cement Mixes with Selected Admixtures Subjected to Sulfuric Acid Attack,” *ACI Materials Journal*, Technical Paper, Title no. 85·M50, Nov. Dec. 1988, p512-518.
8. MPR Associates report “Salem Generating Station Fuel Handling Building Evaluation of Degraded Condition,” MPR-2613, Revision 3, February 2009.
9. W. Ramm and M. Biscoping, “Autogenous healing and reinforcement corrosion of water-penetrated separation cracks in reinforced concrete,” Nuclear Engineering and Design, p191–200, v179 (1998).

Enclosure 2

Prairie Island Nuclear Generating Plant License Renewal Commitments

15 Pages

Enclosure 2
Prairie Island Nuclear Generating Plant License Renewal Commitments

The following table provides the list of commitments included in the Application for Renewed Operating Licenses (LRA) for Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2, as updated in subsequent correspondence.

The commitments in this list are anticipated to be the final commitments which will be confirmed in the NRC's Safety Evaluation Report (SER) for the renewed operating licenses. These commitments, as confirmed in the SER, will become effective upon NRC issuance of the renewed licenses. In addition, as stated in the LRA, the final commitments will be incorporated into the Updated Safety Analysis Report (USAR).

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
1	Each year, following the submittal of the PINGP License Renewal Application and at least three months before the scheduled completion of the NRC review, NMC will submit amendments to the PINGP application pursuant to 10 CFR 54.21(b). These revisions will identify any changes to the Current Licensing Basis that materially affect the contents of the License Renewal Application, including the USAR supplements.	12 months after LRA submittal date and at least 3 months before completion of NRC review Annual Update was submitted by letter dated 4/13/09	1.4
2	Following the issuance of the renewed operating license, the summary descriptions of aging management programs and TLAAs provided in Appendix A, and the final list of License Renewal commitments, will be incorporated into the PINGP USAR as part of a periodic USAR update in accordance with 10 CFR 50.71(e). Other changes to specific sections of the PINGP USAR necessary to reflect a renewed operating license will also be addressed at that time.	First USAR update in accordance with 10 CFR 50.71(e) following issuance of renewed operating licenses	A1.0
3	An Aboveground Steel Tanks Program will be implemented. Program features will be as described in LRA Section B2.1.2.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.2
4	Procedures for the conduct of inspections in the External Surfaces Monitoring Program, Structures Monitoring Program,	U1 - 8/9/2013	B2.1.6

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Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	Buried Piping and Tanks Inspection Program, and the RG 1.127 Inspection of Water-Control Structures Associated with Nuclear Power Plants Program will be enhanced to include guidance for visual inspections of installed bolting.	U2 - 10/29/2014	
5	A Buried Piping and Tanks Inspection Program will be implemented. Program features will be as described in LRA Section B2.1.8.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.8
6	The Closed-Cycle Cooling Water System Program will be enhanced to include periodic inspection of accessible surfaces of components serviced by closed-cycle cooling water when the systems or components are opened during scheduled maintenance or surveillance activities. Inspections are performed to identify the presence of aging effects and to confirm the effectiveness of the chemistry controls. Visual inspection of component internals will be used to detect loss of material and heat transfer degradation. Enhanced visual or volumetric examination techniques will be used to detect cracking. [Revised in letter dated 1/20/2009 in response to RAI 3.3.2-13-01]	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.9
7	The Compressed Air Monitoring Program will be enhanced as follows: • Station and Instrument Air System air quality will be monitored and maintained in accordance with the instrument air quality guidance provided in ISA S7.0.01-1996. Particulate testing will be revised to use a particle size methodology as specified in ISA S7.0.01.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.10

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Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	• The program will incorporate on-line dew point monitoring. [Revised in letter dated 2/6/2009 in response to Region III License Renewal Inspection]		
8	An Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will be completed. Program features will be as described in LRA Section B2.1.11.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.11
9	An Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will be implemented. Program features will be as described in LRA Section B2.1.12.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.12
10	An Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program will be implemented. Program features will be as described in LRA Section B2.1.13.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.13
11	The External Surfaces Monitoring Program will be enhanced as follows: • The scope of the program will be expanded as necessary to include all metallic and non-metallic components within the scope of License Renewal that require aging management in accordance with this program. • The program will ensure that surfaces that are inaccessible or not readily visible during plant operations will be inspected during refueling outages. • The program will ensure that surfaces that are	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.14

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Prairie Island Nuclear Generating Plant License Renewal Commitments

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	inaccessible or not readily visible during both plant operations and refueling outages will be inspected at intervals that provide reasonable assurance that aging effects are managed such that the applicable components will perform their intended function during the period of extended operation. • The program will apply physical manipulation techniques, in addition to visual inspection, to detect aging effects in elastomers and plastics. • The program will include acceptance criteria (e.g., threshold values for identified aging effects) to ensure that the need for corrective actions will be identified before a loss of intended functions. • The program will ensure that program documentation such as walkdown records, inspection results, and other records of monitoring and trending activities are auditable and retrievable. [Revised in letter dated 2/6/2009 in response to RAI B2.1.14-1 Follow Up question]		
12	The Fire Protection Program will be enhanced to require periodic visual inspection of the fire barrier walls, ceilings, and floors to be performed during walkdowns at least once every refueling cycle. [Revised in letter dated 12/5/2008 in response to RAI B2.1.15-3]	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.15
13	The Fire Water System Program will be enhanced as follows:	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.16

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	• The program will be expanded to include eight additional yard fire hydrants in the scope of the annual visual inspection and flushing activities. • The program will require that sprinkler heads that have been in place for 50 years will be replaced or a representative sample of sprinkler heads will be tested using the guidance of NFPA 25, "Inspection, Testing and Maintenance of Water-Based Fire Protection Systems" (2002 Edition, Section 5.3.1.1.1). Sample testing, if performed, will continue at a 10-year interval following the initial testing.		
14	The Flux Thimble Tube Inspection Program will be enhanced as follows: • The program will require that the interval between inspections be established such that no flux thimble tube is predicted to incur wear that exceeds the established acceptance criteria before the next inspection. • The program will require that re-baselining of the examination frequency be justified using plant-specific wear rate data unless prior plant-specific NRC acceptance for the re-baselining was received. If design changes are made to use more wear-resistant thimble tube materials, sufficient inspections will be conducted at an adequate inspection frequency for the new materials. • The program will require that flux thimble tubes that cannot be inspected must be removed from service.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.18

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
15	The Fuel Oil Chemistry Program will be enhanced as follows: • Particulate contamination testing of fuel oil in the eleven fuel oil storage tanks in scope of License Renewal will be performed, in accordance with ASTM D 6217, on an annual basis. • One-time ultrasonic thickness measurements will be performed at selected tank bottom and piping locations prior to the period of extended operation.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.19
16	A Fuse Holders Program will be implemented. Program features will be as described in LRA Section B2.1.20.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.20
17	An Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will be implemented. Program features will be as described in LRA Section B2.1.21	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.21
18	An Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program will be implemented. Program features will be as described in LRA section B2.1.22. Inspections for stress corrosion cracking will be performed by visual examination with a magnified resolution as described in 10 CFR 50.55a(b)(2)(xxi)(A) or with ultrasonic methods. [Revised in letter dated 2/6/2009 in response to RAI B2.1.22-1 Follow Up question]	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.22
19	The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program will be enhanced as follows:	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.23

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Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	• Program implementing procedures will be revised to ensure the components and structures subject to inspection are clearly identified. • Program inspection procedures will be enhanced to include the parameters corrosion and wear where omitted.		
20	A Metal-Enclosed Bus Program will be implemented. Program features will be as described in LRA Section B2.1.26.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.26
21	Number Not Used [Revised in letter dated 3/27/2009]		
22	Number Not Used [Revised in letter dated 4/13/2009]		
23	A One-Time Inspection Program will be completed. Program features will be as described in LRA Section B2.1.29.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.29
24	A One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program will be completed. Program features will be as described in LRA Section B2.1.30.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.30
25	A. A PWR Vessel Internals Program will be implemented. Program features will be as described in LRA Section B2.1.32. B. An inspection plan for reactor internals will be submitted for NRC review and approval at least twenty-four months prior to the period of extended operation. In addition, the submittal will include any necessary revisions to the PINGP PWR Vessel	A. U1 - 8/9/2013 U2 - 10/29/2014 B. U1 - 8/9/2011 U2 - 10/29/2012	B2.1.32

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	Internals Program, as well as any related changes to the PINGP scoping, screening and aging management review results for reactor internals, to conform to the NRC-approved Inspection and Evaluation Guidelines. [Revised in letter dated 5/12/2009] [Revised in letter dated 6/24/09 in response to Follow-up RAI B2.1.38]		
26	The Reactor Head Closure Studs Program will be enhanced to incorporate controls that ensure that any future procurement of reactor head closure studs will be in accordance with the material and inspection guidance provided in NRC Regulatory Guide 1.65.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.33
27	The Reactor Vessel Surveillance Program will be enhanced as follows: • A requirement will be added to ensure that all withdrawn and tested surveillance capsules, not discarded as of August 31, 2000, are placed in storage for possible future reconstitution and use. • A requirement will be added to ensure that in the event spare capsules are withdrawn, the untested capsules are placed in storage and maintained for future insertion.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.34
28	The RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Program will be enhanced as follows: • The program will include inspections of concrete and steel components that are below the water line at the Screenhouse and Intake Canal. The scope will also	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.35

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	require inspections of the Approach Canal, Intake Canal, Emergency Cooling Water Intake, and Screenhouse immediately following extreme environmental conditions or natural phenomena including an earthquake, flood, tornado, severe thunderstorm, or high winds. • The program parameters to be inspected will include an inspection of water-control concrete components that are below the water line for cavitation and erosion degradation. • The program will visually inspect for damage such as cracking, settlement, movement, broken bolted and welded connections, buckling, and other degraded conditions following extreme environmental conditions or natural phenomena.		
29	A Selective Leaching of Materials Program will be completed. Program features will be as described in LRA B2.1.36.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.36
30	The Structures Monitoring Program will be enhanced as follows: • The following structures, components, and component supports will be added to the scope of the inspections: o Approach Canal o Fuel Oil Transfer House o Old Administration Building and Administration Building Addition o Component supports for cable tray, conduit, cable, tubing tray, tubing, non-ASME vessels, exchangers, pumps, valves, piping, mirror	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.38

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	insulation, non-ASME valves, cabinets, panels, racks, equipment enclosures, junction boxes, bus ducts, breakers, transformers, instruments, diesel equipment, housings for HVAC fans, louvers, and dampers, HVAC ducts, vibration isolation elements for diesel equipment, and miscellaneous electrical and mechanical equipment items o Miscellaneous electrical equipment and instrumentation enclosures including cable tray, conduit, wireway, tube tray, cabinets, panels, racks, equipment enclosures, junction boxes, breaker housings, transformer housings, lighting fixtures, and metal bus enclosure assemblies o Miscellaneous mechanical equipment enclosures including housings for HVAC fans, louvers, and dampers o SBO Yard Structures and components including SBO cable vault and bus duct enclosures. o Fire Protection System hydrant houses o Caulking, sealant and elastomer materials o Non-safety related masonry walls that support equipment relied upon to perform a function that demonstrates compliance with a regulated event(s). • The program will be enhanced to include additional inspection parameters. • The program will require an inspection frequency of once every five (5) years for structures and structural		

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Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	components within the scope of the program. The frequency of inspections can be adjusted, if necessary, to allow for early detection and timely correction of negative trends. • The program will require periodic sampling of groundwater and river water chemistries to ensure they remain non-aggressive.		
31	A Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program will be implemented. Program features will be as described in LRA Section B2.1.39.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.39
32	The Water Chemistry Program will be enhanced as follows: • The program will require increased sampling to be performed as needed to confirm the effectiveness of corrective actions taken to address an abnormal chemistry condition. • The program will require Reactor Coolant System dissolved oxygen Action Level limits to be consistent with the limits established in the EPRI PWR Primary Water Chemistry Guidelines." [Revised in letter dated 12/5/2008 in response to RAI B2.1.40-3]	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.40
33	The Metal Fatigue of Reactor Coolant Pressure Boundary Program will be enhanced as follows: • The program will monitor the six component locations identified in NUREG/CR-6260 for older vintage Westinghouse plants, either by tracking the cumulative number of imposed stress cycles using cycle counting, or	U1 - 8/9/2013 U2 - 10/29/2014	B3.2

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Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	by tracking the cumulative fatigue usage, including the effects of coolant environment. The following locations will be monitored: o Reactor Vessel Inlet and Outlet Nozzles o Reactor Pressure Vessel Shell to Lower Head o RCS Hot Leg Surge Line Nozzle o RCS Cold Leg Charging Nozzle o RCS Cold Leg Safety Injection Accumulator Nozzle o RHR-to-Accumulator Piping Tee • Program acceptance criteria will be clarified to require corrective action to be taken before a cumulative fatigue usage factor exceeds 1.0 or a design basis transient cycle limit is exceeded. [Revised in letter dated 1/9/2009 in response to RAI 4.3.1.1-1]		
34	Reactor internals baffle bolt fatigue transient limits of 1835 cycles of plant loading at 5% per minute and 1835 cycles of plant unloading at 5% per minute will be incorporated into the Metal Fatigue of Reactor Coolant Pressure Boundary Program and USAR Table 4.1-8.	U1 - 8/9/2013 U2 - 10/29/2014	B3.2
35	NSPM will perform an ASME Section III fatigue evaluation of the lower head of the pressurizer to account for effects of insurge/outsurge transients. The evaluation will determine the cumulative fatigue usage of limiting pressurizer component(s) through the period of extended operation. The analyses will account for periods of both "Water Solid" and "Standard Steam Bubble" operating strategies. Analysis results will be incorporated, as applicable, into the Metal Fatigue of Reactor	U1 - 8/9/2013 U2 - 10/29/2014	4.3.1.3

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Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	Coolant Pressure Boundary Program. [Revised in letter dated 1/9/2009 in response to RAI 4.3.1.1-1]		
36	NSPM will complete fatigue calculations for the pressurizer surge line hot leg nozzle and the charging nozzle using the methodology of the ASME Code (Subsection NB) and will report the revised CUFs and CUFs adjusted for environmental effects at these locations as an amendment to the PINGP LRA. Conforming changes to LRA Section 4.3.3, "PINGP EAF Results," will also be included in that amendment to reflect analysis results and remove references to stress-based fatigue monitoring. [Added in letter dated 1/9/2009 in response to RAI 4.3.1.1-1]	April 30, 2009 Commitment closed by letter dated 4/28/09	4.3.3
37	NSPM will revise procedures for excavation and trenching controls and archaeological, cultural and historic resource protection to identify sensitive areas and provide guidance for ground-disturbing activities. The procedures will be revised to include drawings and illustrations to assist users in identifying culturally sensitive areas, and pictures of artifacts that are prevalent in the area of the Plant site. The revised procedures will also require training of the Site Environmental Coordinator and other personnel responsible for proper execution of excavation or other ground-disturbing activities. [Added in ER revision submitted in letter dated 3/4/2009]	8/9/2013	ER 4.16.1
38	NSPM will conduct a Phase I Reconnaissance Field Survey of the disturbed areas within the Plant's boundaries. In addition, NSPM will conduct Phase I field surveys of areas of known archaeological sites to precisely determine their boundaries.	8/9/2013	ER 4.16.2

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Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	NSPM will use the results of these surveys to designate areas for archaeological protection. [Added in ER revision submitted in letter dated 3/4/2009]		
39	NSPM will prepare, maintain and implement a Cultural Resources Management Plan (CRMP) to protect significant historical, archaeological, and cultural resources that may currently exist on the Plant site. In connection with the preparation of the CRMP, NSPM will conduct botanical surveys to identify culturally and medicinally important species on the Plant site, and incorporate provisions to protect such plants into the CRMP. [Added in ER revision submitted in letter dated 3/4/2009]	8/9/2013	ER 4.16.2
40	NSPM will consult with a qualified archaeologist prior to conducting any ground-disturbing activity in any area designated as undisturbed and in any disturbed area that is described as potentially containing archaeological resources (as determined by the Phase I Reconnaissance Field Survey discussed in Commitment Number 38). [Added in ER revision submitted in letter dated 3/4/2009]	8/9/2013	ER 4.16.2
41	During the first refueling outage following refueling cavity leak repairs in each Unit (scheduled for refueling outages 1R26 and 2R26), concrete will be removed from the sump C pit to expose an area of the containment vessel bottom head. Visual examination and ultrasonic thickness measurement will be performed on the portions of the containment vessels exposed by the excavations. An assessment of the condition of exposed	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.38

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Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	concrete and rebar will also be performed. Degradation observed in the exposed containment vessel, concrete or rebar will be entered into the Corrective Action Program and evaluated for impact on structural integrity and identification of additional actions that may be warranted. [Added in letter dated 4/6/09 in response to Follow Up RAI B2.1.38]		
42	During the two consecutive refueling outages following refueling cavity leak repairs in each Unit (scheduled for refueling outages 1R26 and 2R26), visual inspections will be performed of the areas where reactor cavity leakage had been observed previously to confirm that leakage has been resolved. The inspection results will be documented. If refueling cavity leakage is again identified, the issue will be entered into the Corrective Action Program and evaluated for identification of additional actions to mitigate leakage and monitor the condition of the containment vessel and internal structures. [Added in letter dated 4/6/09 in response to Follow Up RAI B2.1.38]	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.38
43	Preventive maintenance requirements will be implemented to require periodic replacement of rubber flexible hoses in the Diesel Generators and Support System and in the 122 Diesel Driven Fire Pump that are exposed to fuel oil or lubricating oil internal environments. [Added in letter dated 4/6/09 in response to RAI 3.3.2-8-1] [Revised in letter dated 6/5/09]	U1 - 8/9/2013 U2 - 10/29/2014	Table 3.3.2-8

June 24, 2009

L-PI-09-082
10 CFR 54

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Response to NRC Request for Additional Information Regarding Application for Renewed Operating Licenses

By letter dated April 11, 2008, Northern States Power Company, a Minnesota Corporation, (NSPM) submitted an Application for Renewed Operating Licenses (LRA) for the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2. During the Aging Management Audit, the NRC was briefed on water seepage from the refueling cavity into the containment sumps that had been detected during refueling outages. In a letter dated November 5, 2008, the NRC issued RAI AMP-B2.1.38-2 regarding that seepage, and PINGP responded on December 5, 2008. The matter was discussed in a public meeting on March 2, 2009. An additional Follow-up RAI was issued on March 31, 2009, and the PINGP response was provided on April 6, 2009. On May 28, 2009 the NRC visited the PINGP site to review documents related to refueling cavity leakage. On June 4, 2009, the NRC issued the Safety Evaluation Report With Open Items Related to the License Renewal of the Prairie Island Nuclear Generating Plant Units 1 and 2 (SER). The SER identified the refueling cavity seepage as Open Item 3.0.3.2.17-1, pending completion of the NRC review of the April 6, 2009, Follow-up RAI response. Subsequently, in a letter dated June 10, 2009, an additional Follow-up RAI was issued regarding that seepage. The PINGP response to that Follow-up RAI is provided in Enclosure 1.

As a separate matter, in a conference call on June 10, 2009, the PINGP PWR Vessel Internals Program was discussed. During that call, PINGP agreed to clarify Part B of the associated License Renewal Commitment No. 25 to indicate that the vessel internals inspection plan submittal would also include any LRA changes to the scoping, screening, and AMR results, and the description of the PWR Vessel Internals Program, that are necessary to reflect the final NRC-approved Inspection and Evaluation guidance. Accordingly, License Renewal Commitment No. 25 is being revised as noted below to incorporate this clarification. A complete listing of PINGP License Renewal Commitments, updated to reflect NSPM correspondence to date, is provided in Enclosure 2.

If there are any questions or if additional information is needed, please contact Mr. Eugene Eckholt, License Renewal Project Manager.

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Summary of Commitments

This letter contains no new commitments. License Renewal Commitment No. 25 is revised to read as follows:

A. A PWR Vessel Internals Program will be implemented. Program features will be as described in LRA Section B2.1.32. The program will be implemented prior to the period of extended operation.
B. An inspection plan for reactor internals will be submitted for NRC review and approval at least twenty-four months prior to the period of extended operation. In addition, the submittal will include any necessary revisions to the PINGP PWR Vessel Internals Program, as well as any related changes to the PINGP scoping, screening and aging management review results for reactor internals, to conform to the NRC-approved Inspection and Evaluation Guidelines.

The implementation schedules for Parts A and B of this commitment are unchanged: 8/9/2013 (Unit 1) and 10/29/2014 (Unit 2) for Part A, and 8/9/2011 (Unit 1) and 10/29/2012 (Unit 2) for Part B.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on June 24, 2009.

/S/ Michael D. Wadley

Michael D. Wadley
Site Vice President, Prairie Island Nuclear Generating Plant Units 1 and 2
Northern States Power Company - Minnesota

Enclosures (2)

cc:

Administrator, Region III, USNRC
License Renewal Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
Prairie Island Indian Community ATTN: Phil Mahowald
Minnesota Department of Commerce

Enclosure 1
Response to Follow-up RAI B2.1.38

NRC Follow-up RAI B2.1.38

In letter L-PI-08-098, dated December 5, 2008, "Responses to NRC Requests for Additional Information Dated November 5, 2008 Regarding Application for Renewed Operating Licenses," the applicant submitted responses to the staff's RAIs. In addition, the applicant provided information during a public meeting on March 2, 2009, and a previous follow-up RAI in letter L-PI-09-047, April 6, 2009.

On May 28, 2009, the NRC staff performed an audit at the Prairie Island Nuclear Generating Station (PINGP), Units 1 and 2, related to the supporting documentation for the reactor cavity leakage. The staff reviewed the following documents:

References:
1. Dominion Energy Incorporated, "Evaluation of Effects of Borated Water Leaks on Concrete Reinforcing Bars and Carbon Steel Plate of the Containment Vessels at Prairie Island Units 1 and 2," Report No. R-4448-00-01, Rev. 0.
2. Prairie Island Nuclear Generating Station, "Refueling Cavity leakage, Event Date 1988-2008," Report No. RCE 01160372-01, Volumes 1 and 2.

In order to complete our review of issues related to the PINGP reactor cavity leakage discussed in the above referenced reports, the staff requests the following additional information:

[Note: To assist the reader, the NSPM response to each part (a through i) of the RAI is provided immediately after the statement of each part. References applicable to all responses are listed after the response to part i.]

Follow-up RAI Part a

a) Section 2.2 of Reference 1 recommends that a test be performed to determine if there is high assurance that the pH present in the water between the containment steel plates and concrete is more than 12.5. Please provide a schedule for performing this test.

NSPM Response to Part a

It is well established that boric acid will be neutralized by contact with concrete. Tests of the effects of boric acid in contact with concrete have shown a rapid rise in pH of the boric acid solution. (Page 3-2 of Reference 5) It has also been shown by calculations using the EPRI MULTEQ program that the equilibrium pH will be about 12.5 with excess quantities of calcium present, as would be the case with the large amounts of calcium hydroxide in concrete. Evidence that neutralization of boric acid has occurred in concrete at Prairie Island is provided by the pH values of 7 and 7.8 that have been measured in the active leakage. (Page 4-2 of Reference 1) To provide additional confirmation of this behavior, a simple laboratory test has been performed, as discussed below.

Enclosure 1
Response to Follow-up RAI B2.1.38

To provide background for the process which results in boric acid being neutralized when in contact with concrete, the materials that form concrete and their chemical properties are summarized below.

Concrete is a composite material consisting of a binder (cement paste) and a filler of fine and/or course aggregate particles that combine to form a synthetic conglomerate. The cement is a mixture of compounds made by grinding crushed limestone, clay, sand, and iron ore together to form a homogeneous powder that is then heated at very high temperatures to form a clinker. After the clinker cools, it is ground and mixed with a small amount of gypsum to regulate setting and facilitate placement. This produces the general-purpose portland cement that is mixed with water to produce cement paste that binds the aggregate particles together. Portland cements are composed primarily of four chemical compounds: tricalcium silicate, dicalcium silicate, tricalcium aluminate, and tetracalcium aluminoferrite. The calcium silicate hydrates constitute about 75% of the mass. (Section 3, Reference 4)

The hardened cement paste consists mainly of calcium silicate hydrates, calcium hydroxide, and lower proportions of calcium sulphoaluminate hydrate. About 20% of the hardened cement paste volume is calcium hydroxide. The pore solution is normally a saturated solution of calcium hydroxide within which high concentrations of potassium and sodium hydroxides are present. (Section 3, Reference 4)

Hardening of concrete occurs as a result of hydration, which is a chemical reaction in which the major compounds in the cement form chemical bonds with water molecules and become hydrates. Since cement is the most expensive ingredient in concrete, it is desirable to utilize the minimum amount necessary to produce the desired properties and characteristics. Aggregate typically occupies 60 to 75% of the volume of concrete, with the balance of the concrete mix generally consisting of 10 to 15% cement, 15 to 20% water, and 5 to 8% air, if entrained. (Section 3, Reference 4)

Tests reported in Reference 3 indicate that water of neutral pH placed in small holes drilled into concrete reaches an equilibrium pH of about 12.8 to 13.3 after one to two weeks. This is consistent with general industry information that indicates that pore water in concrete generally has a pH of about 12.5 or higher. This pH is fully protective of rebar (Section 4.3.2 of Reference 4). These results indicate that normal fresh water in contact with concrete will reach an equilibrium pH of 12.5 or more and be protective of steel (i.e., result in insignificant corrosion). However, these results do not address the possible effects of the boron in the water on the pH. This is discussed further in the following paragraphs.

In order to more firmly establish the pH that will develop in small volumes of borated water in contact with large amounts of concrete, a simple laboratory test was performed. The test was performed by adding chemicals representative of those in concrete to an open beaker with a volume of one liter of deionized water at room temperature. After each chemical addition, the solution was stirred and the pH was measured. The steps of the test and the pH values measured are shown in Table 1. Comments regarding the solutions tested and the results are as follows:

- Solution 1 involved the addition of calcium oxide alone, and represents the case of water in contact with the calcium hydroxide that is present in large quantities in concrete. Calcium oxide forms calcium hydroxide when dissolved in water. The measured pH of 12.05 was slightly below the normally reported equilibrium value of 12.5 for a saturated solution of calcium hydroxide, which is expected to contain approximately 1,000 ppm calcium ions (as calcium hydroxide). The slightly lower-than-expected pH is attributed to the fact that full equilibrium had not been reached when the pH measurement was made (i.e., the pH was measured before the calcium oxide was completely dissolved).

- Solution 2 reflects the addition of sodium, albeit at lower concentrations than calcium. The concentration of sodium added to the test solution was consistent with that anticipated based on information reported in Reference 3 and is consistent with Reference 4 which notes that high concentrations are present in pore water. As shown in Table 1, the sodium increased the measured pH by a small amount.

- Solution 3 reflects the addition of the equivalent of 3000 ppm boric acid, which is representative of the concentration present during refueling. The measured pH of 8.85 is in the expected range indicated by calculations for a case where some boron remains in solution.

- Solutions 4 through 8 reflect the addition of increasing amounts of calcium oxide. This simulates exposure over time to the excess amounts of calcium hydroxide that are present in the concrete. Initially, the excess boric acid present in solution buffers the solution pH, and calcium oxide additions have only a small effect on the resulting pH. However, the pH increases rapidly to the 12.3 range after the concentration of calcium oxide exceeds the stoichiometric amount needed to react with the boric acid that was initially present.

Table 1 pH of Simulated Boric Acid Leakage in Contact with Chemicals From Concrete

Solution	Mass (g)				pH
	CaO	ppm Ca	NaOH	H_3BO_3	
1	1.3992	1,000			12.05
2	1.3992	1,000	3.3061		12.39
3	1.3992	1,000	3.3061	17.1585	8.85
4	6.3992	4,573	3.3061	17.1585	8.94
5	11.4016	8,149	3.3061	17.1585	9.01
6	24.7441	17,684	3.3061	17.1585	9.74
7	29.7456	21,259	3.3061	17.1585	12.28
8	34.7469	24,833	3.3061	17.1585	12.30

The primary conclusion that can be drawn from the test results summarized in Table 1 is that a high protective pH is reached by borated water that is in contact with excess amounts of calcium oxide. This indicates that a similar high pH will develop in borated water trapped between the steel containment vessel and the concrete since there are excess amounts of calcium hydroxide in the concrete.

The test discussed above was performed in accordance with written instructions, and the results were documented following normal laboratory practice. However, it was not performed in accordance with formal nuclear QA requirements. Nevertheless, the results are considered to be consistent with theoretical values and provide strong supporting evidence that high protective pH values will be reached in borated water trapped between the steel containment vessel and the concrete.

Follow-up RAI Part b

b) Section 2.2 of Reference 1 recommends removal of concrete inside the containment at the following locations:

   i. Sump C
   ii. Through the wall at elevation 695 closer to the transfer tube

   However, Northern States Power Company, Minnesota (NSPM) in a letter dated April 6, 2009, committed to remove concrete from Sump C only. Please clarify. In addition, please explain why removal of concrete from Sump C is not planned during the next scheduled outages at PINGP, Units 1 and 2.

NSPM Response to Part b

In reviewing the recommendations from Reference 1 to determine the appropriate Corrective Actions to be assigned within the Corrective Action Program, engineering management considered the value, need and sufficiency of each recommendation provided. The review concluded that there is limited value in removing concrete from the inside diameter of the containment vessel at the 697' floor elevation, as it is not known whether this area is wetted and has a potential for corrosion to exist. The site has instead removed the grout in the RHR suction sump of both units as these areas are lower in containment elevation and consistently show wetting when refueling cavity leakage occurs. It is also believed the RHR sumps would be more likely than the 697' elevation to show any corrosion due to repeated wetting and close proximity to ambient oxygen. In addition, much of the area between the transfer tube at the 715' elevation and annulus floor at the 706' elevation can be monitored by ultrasonic thickness measurement from the exterior of the containment vessel, further diminishing the value of removing concrete from the interior wall. Therefore, the Corrective Action assignments did not include removal of concrete at elevation 697'.

Removal of the concrete in sump C (under the reactor vessel) will be performed in the next refueling outages following the outages during which the refueling cavity liners are repaired. This is primarily for logistical reasons. The estimated thickness of the concrete at the thinnest location in the floor of sump C is 16 inches with reinforcing bar both near the top of the proposed excavation and near the containment vessel inside diameter. The work area is relatively small and in close proximity to the reactor vessel thimble tubes which provide an ASME Class 1 pressure boundary. As such, performing the excavation safely requires considerable planning and specialized tooling. The site will use the upcoming refueling outage in each Unit to survey the excavation sites, and the time between outages to plan the excavations and secure the appropriate tools.

Follow-up RAI Part c

c) Section 4.2 of Reference 1 has identified an upper bound loss of 0.25 inch in the 1.50 inch steel containment due to borated water corrosion over a 36 year period. Please advise if the stresses in the steel containment remain within the American Society of Mechanical Engineers Code allowable values for this loss of 0.25 inches. According to Section 4.1 of Reference 1, minimum thickness required for the steel containment for all loading conditions is 1.4908 inches. In addition, please clarify if NSPM has considered the potential of continued reactor cavity leakage over the life extension period of 60 years.

NSPM Response to Part c

The evaluation estimates the likely corrosion of the containment vessel to date at no more than 0.010". This estimate accounts for the neutralization of borated water in concrete. Recent ultrasonic thickness measurements, including measurements of known wetted areas in the RHR suction sump, showed no corrosion with all thickness readings above the nominal plate thickness. If any significant loss were identified, an ASME code evaluation would be required.

As discussed in section 4.2 of Reference 1, the 0.25" value of corrosion assumes continuous wetting with aerated, concentrated, boric acid over a period of 36 years. This value does not consider the buffering effect of the concrete or the consumption of oxygen dissolved in the water. Therefore, the long term environment that could lead to this level of corrosion would not exist. The report's reference to 0.25" as an upper bound does not clearly convey its meaning. This value was provided for comparison purposes only to provide further support for the low corrosion rates expected, and does not represent an expected condition in a PINGP containment vessel. Therefore, an ASME Code analysis of the containment vessel which assumes loss of 0.25" of vessel wall has not been performed. Reference 1 does not suggest that a wall loss of this magnitude would leave the vessel capable of meeting code allowables. Indeed, the report states that any observed wall loss that reduced the vessel below the nominal 1.5" thickness would have to be evaluated in accordance with ASME Section XI. Reference 1 only provides the judgment that even with a wall loss of 0.25", the containment vessel would still be able to withstand accident pressure without a loss of containment integrity.

The areas of the steel containment vessel that are potentially subject to borated water exposure from refueling cavity leakage are the bottom head and sections of the shell behind concrete at the end of the refueling cavity and transfer pit (referred to in the

USAR as the “cold spot”). Both the shell and bottom head are fabricated from SA-516-70 material with a minimum tensile strength of 70 ksi. The Pioneer Service & Engineering Company containment vessel stress report shows the shell and bottom head were designed in accordance with ASME section VIII with a design pressure of 41.4 psig and a corresponding required thickness of 1.5 inch.

In accordance with ASME section VIII the design allowable membrane stress is limited to 25% of the material minimum tensile strength, or 17.5 ksi. Allowable stresses during a Design Basis Accident (DBA) are considerably higher. As indicated in USAR table 12.2-22, total stresses under a DBA with Design Basis Earthquake range from 24.48 ksi to 27.86 ksi. These stresses are approximately one half the DBA allowable stress of 52.5 ksi. Stresses are generally proportional to thickness. As such, even with thinning of 0.25 inch of the 1.5 inch shell thickness, total stresses would still be well below the stress limits for the load combinations that combine stresses from a DBA with those from a Design Basis Earthquake.

The site fully expects that leakage will be stopped during the next refueling outage of each unit. During the outage following the outage of repair, any water observed in Sump C will be evacuated. However, any residual water behind concrete that may not be able to be evacuated would have a very small stagnant volume with its pH elevated by the alkalinity in concrete. Any potential corrosion in such regions would be similar in magnitude to (or lower than) the 0.010" conservatively estimated for 36 years to date. With this level of corrosion, the overall conclusion remains valid that containment vessel integrity would be unaffected.

The site will continue to monitor the containment vessel and internal structures through the ASME section XI, IWE program and the Structures Monitoring program. If any new leakage is identified that indicates the refueling cavity leakage has recurred, the issue will be entered into the Corrective Action Program for evaluation and identification of corrective actions.

Follow-up RAI Part d

d) In Section 5.2.3 of Reference 1, the rate of degradation estimated for PINGP concrete is two times that used previously for Salem/Connecticut Yankee plants to account for the difference in the type of concrete aggregates at PINGP. Please advise if NSPM has performed or intends to perform any tests to confirm the use of this assumption for the degradation rate.

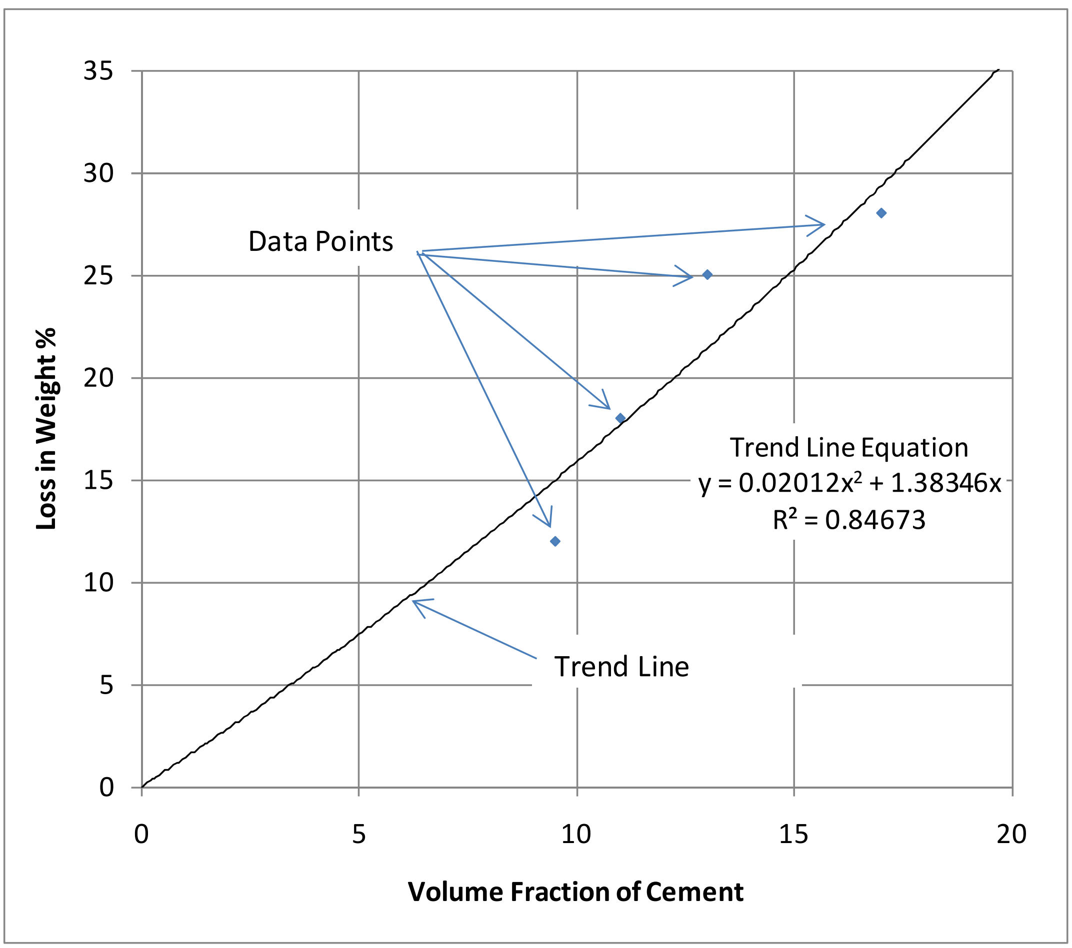
NSPM Response to Part d

The acceleration factor of 2.0 for the increased rate of degradation resulting from the presence of about 5% carbonate-based aggregate at Prairie Island relative to the 0% carbonate-based aggregate at the cited plants was based on engineering judgment.

Tests of the effects of acids on reinforced concrete show that acids weaken concrete by dissolving cement and carbonate-based aggregate (Page 29 of Reference 4 and page 4 of Reference 6). Tests discussed in Reference 7 indicate that the weight loss experienced by cubes of concrete immersed in acid increases as the volume fraction of cement in the concrete increases. These tests were performed on concrete that had non-soluble aggregate (i.e., only siliceous aggregate and no calcium carbonate-based aggregate). This increased weight loss is attributed to the fact that, as the volume fraction of cement increases, the volume fraction of the concrete that is soluble in acids increases. Based on this result, it is reasonable to assume that the effect of boric acid on concrete containing soluble aggregates will follow a similar pattern; i.e., the degradation of concrete due to boric acid will increase as the total fraction of material in the concrete that is soluble in acids increases, whether that soluble material is cement or aggregate.

Quantitative data in Reference 7 for weight loss of concrete cubes when exposed to an acid concentration of 3% is shown in Figure 1. A trend line is shown on the figure that provides an equation for quantifying this dissolution behavior. As can be seen, the weight loss increases somewhat more strongly than linearly as the volume fraction of cement, (i.e., of soluble material), increases.

Figure 1 Weight Loss in Acid vs. Volume Fraction of Cement (based on data from Reference 7)



A footnote on page 10 of Reference 4 indicates that concrete normally contains 10 to 15% of cement, or an average of about 12.5%. As noted on page 2-2 of Reference 1, the amount of carbonaceous aggregate in the concrete used at Prairie Island was about 5%. Adding this 5% to the average 12.5% cement results in a total of 17.5% soluble material in the concrete at Prairie Island as compared to the 12.5% in a concrete with all igneous (no carbonate) based aggregate such as was used at the cited plants. (Section 8.1.3 of Reference 8) Using the trend line equation shown in Figure 1, increasing the fraction of soluble material (cement in the figure, cement plus soluble aggregate in this case) from 12.5% to 17.5% increases the weight loss by a factor of 1.49. This indicates that the assumed factor of 2.0 increase in severity of acid attack to account for the presence of 5% carbonate-based aggregate is conservative by a significant margin (2 vs. 1.49).

Based on this result, it is considered that there is no need for additional tests to evaluate the effects of the carbonate-based aggregate used at Prairie Island.

Follow-up RAI Part e

e) Section 5.2.6 of Reference 1 states, "…concrete is not relied upon for tensile strength (tensile strength provided by rebar)." Please explain how formation of large thru thickness cracks will affect the transverse shear capacity of concrete slabs and walls. Shear strength of concrete is directly related to the tensile strength.

NSPM Response to Part e

The nominal shear strength of reinforced concrete is based on the combination of the nominal shear strength provided by the concrete and the nominal strength provided by shear reinforcement (steel rebar). If a reinforced concrete member were postulated to have a large idealized crack along the entire shear plane, the reinforcement would need to carry the shear force. However, there is no indication that such a crack might exist at PINGP.

The existence of a wide crack in the refueling cavity floor or walls that has traveled through the entire thickness of a concrete section along a single shear plane is highly unlikely. Plant operating experience has not identified any large cracks in either the Unit 1 or Unit 2 reactor containment concrete structures. The cracks that have been identified are often characterized as hairline or normal shrinkage cracks. Large through-member cracks are not reasonable to postulate at PINGP based on the plant's adherence to good design and construction practices. The refueling pool floor slab, at its minimum, is 4'-0" thick (3'-8" of reinforced concrete and 4" of leveling grout) and contains both top and bottom reinforcement in each direction. Where water seepage from cracks has occurred during flood up of the refueling cavity, the cracks have never been described as large or wide, which is consistent with the small amount of water that seeps from the cracks, estimated at no more than 1-2 gallons per hour. Additionally, no evidence of significant washout of material has been identified, a condition that could be indicative of the development of a widening crack.

Follow-up RAI Part f

f) Section 5.2.6 of Reference 1 states, "Degradation of concrete by exposure to borated water can also occur at cracks in the concrete. This could lead to loss of strength of concrete in a narrow band through thickness of material." A crack across thickness of material may expose rebars to corrosion. Please explain how this effect is considered in determining the structural integrity of concrete walls and slabs.

NSPM Response to Part f

The quotation above is from Section 5.2 of Reference 1, "Degradation of Concrete Due to Chemical Attack." This section deals with concrete degradation, and does not address rebar corrosion. Rebar corrosion is addressed in the following section, Section 5.3, "Corrosion of Rebar Caused by Exposure to Borated Water." The specific question asked in Part f is addressed by the second bullet in Section 5.3 of Reference 1. This section indicates that the dissolution of calcium hydroxide from the concrete around rebar at cracks in the concrete might be assumed to develop conditions that might lead to increased rates of corrosion of the rebar. However, tests performed for the cited plants and other tests described in the open literature indicate that corrosion in such situations has been negligible, even when the low pH borated water reaching the cracks was regularly refreshed. It appears that conditions at the rebar remain sufficiently alkaline in such situations to passivate the surface, despite the presence of refreshed borated water.

Reference 5 documents tests performed using steel coupons in a refreshed boric acid solution. The relevant conclusion from the report reads as follows: "The short-term corrosion rate of reinforcing steel was observed to be about 4.2 mils/year (0.0042-inch/year) based on an exposure time of up to 56-days in an aerated boric solution with a pH in the range of 5.2 to 6.4. The corrosion rate should decrease as pH increases and oxygen content decreases under long-term conditions in the field." The tests documented in Reference 9 were two year tests of samples using refreshed boric acid in which the rebar was exposed to the boric acid at cracks. The main conclusion was as follows: "For a penetration period of 2 years, there were weakenings of the cross section of the reinforcing bars not worth being mentioned by reinforcement corrosion in separation cracks penetrated by deionised water (neutral, pH 7.0) and boric acid treatment deionised water of the pH-values 5.2 and 6.1 with crack widths of up to approximately 0.4 mm."

The tests documented in Reference 9 are considered to be the most relevant since they were long term (2 year) and used realistic cracked specimens with the rebar exposed to the boric acid in a crack. As noted in Reference 9, the corrosion of the rebar was "not worth being mentioned." The total exposure time of rebar cracks in the reinforced concrete under the refueling cavity at Prairie Island is conservatively estimated as 15 days per outage for 25 outages or 375 days or 1.03 years. This is significantly less than the two year test period of Reference 9 which resulted in no significant corrosion. Accordingly, it is concluded that no significant corrosion of rebar at cracks in the structural concrete under the refueling cavity has occurred at Prairie Island, and that effects of rebar corrosion at cracks in the concrete on structural integrity are insignificant.

Follow-up RAI Part g

g) Section 5.2.3 of Reference 1 estimated the upper bound loss of concrete depth over a 36 year period to be 0.31 inches. Please address the impact of this loss of 0.31 inch of concrete behind the stainless steel liner plate on the load carrying capacity of the stainless steel liner plate.

NSPM Response to Part g

As discussed in section 5.2.3 of the Reference 1, the upper bound loss of 0.31 inch is conservative as it assumes leakage every outage for 25 outages over a period of 36 years. There is no evidence of leakage prior to 1987 and leakage has only been observed in about half the outages due to intermittent successful leak mitigation with caulking or spray-on liner.

The impact of a potential loss of 0.31" of concrete on the load carrying capacity of the liner is minimal. The liner is effectively a membrane that is backed under the bottom and around the sides with concrete that is generally four to five feet thick. As a result, the impact on the load carrying capacity of the refueling cavity pool structure is negligible as the material loss is, at most, less than 1% of the concrete thickness. Large areas of washout or dissolution under the liner are not expected as leak rates are small (on the order of 1-2 gallons per hour) and only occur while the refueling cavity is flooded. However, even if large areas of washout or dissolution on the order of square feet and 0.31" depth were to occur, they would only be expected to result in shallow depressions in the liner and not in a failure such as tearing of the liner plate. The liner was constructed with the seams welded to stainless steel structural shapes embedded in the concrete. As such, the seams are reinforced and are not subject to loss of concrete directly under a seam. The liner plates are 3/16" and 1/4" thick Type 304 stainless steel. Stainless steel is generally very ductile and can withstand significant elongation and deformation before failure. The site has not experienced any observable depressions on either the floors or walls of the refueling cavities.

Follow-up RAI Part h

h) NSPM in a letter dated April 6, 2009, committed to visual inspections during the consecutive refueling outages of the areas where reactor cavity leakage has been observed following refueling cavity leak repairs in each unit. Which aging management program (AMP) will be used to address this issue?

NSPM Response to Part h

License Renewal Commitment Number 42 from the April 6, 2009 letter reads as follows:

> During the two consecutive refueling outages following refueling cavity leak repairs in each Unit (scheduled for refueling outages 1R26 and 2R26), visual inspections will be performed of the areas where reactor cavity leakage had been observed previously to confirm that leakage has been resolved. The inspection

results will be documented. If refueling cavity leakage is again identified, the issue will be entered into the Corrective Action Program and evaluated for identification of additional actions to mitigate leakage and monitor the condition of the containment vessel and internal structures.

The inspections of the locations which previously showed signs of refueling cavity leakage are special inspections assigned as corrective actions within the Corrective Action Program. The inspections will invoke the methodology, documentation requirements and acceptance criteria of the Structures Monitoring Program. The Structures Monitoring Program is the appropriate program since the locations where leakage has been observed are in containment interior structures.

Thereafter, as discussed in the April 6, 2009, letter, general monitoring for leakage and degradation will be performed in all areas of containment in accordance with the ASME Section XI, Subsection IWE Program and the Structures Monitoring Program. In addition, the Boric Acid Corrosion Program and the 10 CFR Part 50, Appendix J Program include inspections inside containment. These programs contain provisions to identify, evaluate, and correct degraded conditions prior to loss of function, ensuring the effects of aging for plant SSCs are adequately managed. Periodic visual inspections are performed, and inspection schedules are prescribed, ensuring timely identification of any degraded condition. The ASME Section XI, Subsection IWE Program also incorporates the requirements of 10 CFR 50.55a(b)(2)(ix) for the examination of metal containments, including the requirement in 10 CFR 50.55a(b)(2)(ix)(A) to evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in degradation to, such inaccessible areas.

During the current license period as well as the period of extended operation, degradation identified by these programs will be entered into the PINGP Corrective Action Program for evaluation and identification of corrective actions.

Follow-up RAI Part i

i) Page 52 of Reference 2 identified five recommendations to address reactor cavity leakage. Please provide NSPM’s action plan and schedule for completing these five recommendations.

NSPM Response to Part i

Page 52 of the Root Cause Evaluation (RCE) recommended a repair plan that includes the following steps:

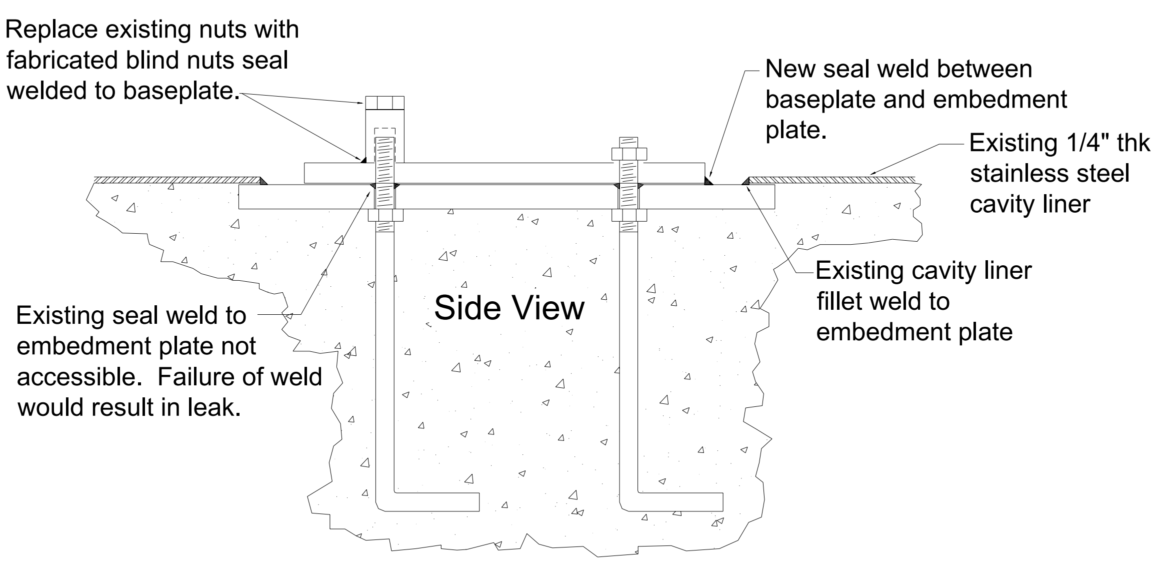
1. Unbolt and set aside all mechanically fastened fixtures (RCC Change Fixture, internals stands, and guide tube supports).
2. Vacuum Box penetrations and embedment plates to locate existing leaks. Weld repair and vacuum box completed welds
3. Preemptively seal weld and vacuum box all penetrations.

4. Vacuum Box, and/or PT weld seams and repair as needed to ensure no leakage due to stress corrosion cracking.
5. Pressure test or PT transfer tube bellows attachment welds and weld repair as needed.

The RCE also states that alternate approaches can also be considered provided the final repair plan permanently and completely mitigates future leakage.

The RCE concluded that the most likely refueling cavity leakage points are where the anchor studs for the reactor internals stands and the Rod Control Cluster (RCC) Change Fixture penetrate the associated embedment plates. The studs for these fixtures pass through holes in the embedment plates and are seal welded on the underside for the internals storage stands, and on top (then ground flush) for the RCC Change Fixture. Pinhole leaks or small cracks in one or more of the seal welds would result in a leak path through the embedment plate along the threads of the studs, allowing water under the cavity liner.

The plant management review of the Root Cause Evaluation considered the value, need and sufficiency of each recommendation. The current repair plan that emerged from these reviews is to replace the existing anchor nuts with new fabricated blind nuts that are seal welded to the existing baseplates. In addition, the existing baseplates will be seal welded to the embedment plates. The new welds will be readily accessible for examination to ensure the existing anchors and baseplates are leak tight. The following sketch depicts a typical RCC change fixture support showing the planned repairs. Similar repairs will be made to the internals stands supports.



General Arrangement of Change Fixture Supports

This repair strategy effectively accomplishes the goals of recommended steps 1 through 3, the intent of which is to permanently repair refueling cavity leakage, but in a more focused and dose-effective manner. The guide tube supports which are mounted to the refueling cavity wall were concluded to be unlikely sources of leakage and will not be sealed in a similar manner. Although the guide tube supports are of similar construction, past success at mitigating leakage by sealing the supports of the internals stands and RCC change fixture suggests that leakage occurs only at those locations.

Recommended step 4 is to vacuum box and/or dye penetrant test refueling cavity liner weld seams to ensure no leakage due to stress corrosion cracking. The site performed vacuum box and/or dye penetrant test of accessible seams of both units in 1998 and 1999 with no indications of cracking. NSPM does plan to examine a sample of accessible seams to ensure no cracking has occurred since the last inspection.

Recommended step 5 is to pressure test or dye penetrant test the transfer tube bellows attachment welds and perform weld repairs as needed. NSPM believes these welds are leak tight as past refueling cavity leakage has been completely mitigated through sealing only the internals stands and change fixture anchors. As a precaution, NSPM does plan to perform pressure testing or dye penetrant testing of accessible portions of the transfer tube bellows welds to preclude the possibility of water leaking along the transfer tube to the inside surface of the containment vessel.

All actions discussed above are planned for the next refueling outage for each unit, currently scheduled for fall 2009 for Unit 1 and spring 2010 for Unit 2.

References for NSPM Responses to Follow-up RAI B2.1.38 Parts a through i

1. Dominion Engineering, Inc. report “Evaluation of Effects of Borated Water Leaks on Concrete Reinforcing Bars and Carbon Steel Plate of the Containment Vessels at Prairie Island Units 1 and 2,” report R-4448-00-01, Rev. 0.
2. Prairie Island Nuclear Generating Station, “Refueling Cavity leakage, Event Date 1988-2008,” Report No. RCE 01160372-01, Volumes 1 and 2.
3. A. A. Sagües, et al., “Evolution of pH During In-Situ Leaching in Small Concrete Cavities,” *Cement and Concrete Research*, Vol. 27, No. 11, pp. 1747-1759, 1997.
4. D. J. Naus, *Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures - A Review of Pertinent Factors*, NUREG/CR-6927 - ORNL/TM-2006/529, Feb. 2007.
5. MPR Associates report “Boric Acid Attack of Concrete and Reinforcing Steel,” MPR-2634, Revision 2, February 2009.
6. B. Kerkhoff, “Effects of Substances on Concrete and Guide to Protective Treatments,” Portland Cement Association, Item Code: IS001, 2007.
7. N. I. Fattuhi and B. P. Hughes, “Ordinary Portland Cement Mixes with Selected Admixtures Subjected to Sulfuric Acid Attack,” *ACI Materials Journal*, Technical Paper, Title no. 85·M50, Nov. Dec. 1988, p512-518.
8. MPR Associates report “Salem Generating Station Fuel Handling Building Evaluation of Degraded Condition,” MPR-2613, Revision 3, February 2009.
9. W. Ramm and M. Biscoping, “Autogenous healing and reinforcement corrosion of water-penetrated separation cracks in reinforced concrete,” Nuclear Engineering and Design, p191–200, v179 (1998).

Enclosure 2

Prairie Island Nuclear Generating Plant License Renewal Commitments

15 Pages

Enclosure 2
Prairie Island Nuclear Generating Plant License Renewal Commitments

The following table provides the list of commitments included in the Application for Renewed Operating Licenses (LRA) for Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2, as updated in subsequent correspondence.

The commitments in this list are anticipated to be the final commitments which will be confirmed in the NRC's Safety Evaluation Report (SER) for the renewed operating licenses. These commitments, as confirmed in the SER, will become effective upon NRC issuance of the renewed licenses. In addition, as stated in the LRA, the final commitments will be incorporated into the Updated Safety Analysis Report (USAR).

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
1	Each year, following the submittal of the PINGP License Renewal Application and at least three months before the scheduled completion of the NRC review, NMC will submit amendments to the PINGP application pursuant to 10 CFR 54.21(b). These revisions will identify any changes to the Current Licensing Basis that materially affect the contents of the License Renewal Application, including the USAR supplements.	12 months after LRA submittal date and at least 3 months before completion of NRC review Annual Update was submitted by letter dated 4/13/09	1.4
2	Following the issuance of the renewed operating license, the summary descriptions of aging management programs and TLAAs provided in Appendix A, and the final list of License Renewal commitments, will be incorporated into the PINGP USAR as part of a periodic USAR update in accordance with 10 CFR 50.71(e). Other changes to specific sections of the PINGP USAR necessary to reflect a renewed operating license will also be addressed at that time.	First USAR update in accordance with 10 CFR 50.71(e) following issuance of renewed operating licenses	A1.0
3	An Aboveground Steel Tanks Program will be implemented. Program features will be as described in LRA Section B2.1.2.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.2
4	Procedures for the conduct of inspections in the External Surfaces Monitoring Program, Structures Monitoring Program,	U1 - 8/9/2013	B2.1.6

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	Buried Piping and Tanks Inspection Program, and the RG 1.127 Inspection of Water-Control Structures Associated with Nuclear Power Plants Program will be enhanced to include guidance for visual inspections of installed bolting.	U2 - 10/29/2014	
5	A Buried Piping and Tanks Inspection Program will be implemented. Program features will be as described in LRA Section B2.1.8.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.8
6	The Closed-Cycle Cooling Water System Program will be enhanced to include periodic inspection of accessible surfaces of components serviced by closed-cycle cooling water when the systems or components are opened during scheduled maintenance or surveillance activities. Inspections are performed to identify the presence of aging effects and to confirm the effectiveness of the chemistry controls. Visual inspection of component internals will be used to detect loss of material and heat transfer degradation. Enhanced visual or volumetric examination techniques will be used to detect cracking. [Revised in letter dated 1/20/2009 in response to RAI 3.3.2-13-01]	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.9
7	The Compressed Air Monitoring Program will be enhanced as follows: • Station and Instrument Air System air quality will be monitored and maintained in accordance with the instrument air quality guidance provided in ISA S7.0.01-1996. Particulate testing will be revised to use a particle size methodology as specified in ISA S7.0.01.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.10

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	• The program will incorporate on-line dew point monitoring. [Revised in letter dated 2/6/2009 in response to Region III License Renewal Inspection]		
8	An Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will be completed. Program features will be as described in LRA Section B2.1.11.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.11
9	An Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will be implemented. Program features will be as described in LRA Section B2.1.12.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.12
10	An Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program will be implemented. Program features will be as described in LRA Section B2.1.13.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.13
11	The External Surfaces Monitoring Program will be enhanced as follows: • The scope of the program will be expanded as necessary to include all metallic and non-metallic components within the scope of License Renewal that require aging management in accordance with this program. • The program will ensure that surfaces that are inaccessible or not readily visible during plant operations will be inspected during refueling outages. • The program will ensure that surfaces that are	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.14

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	inaccessible or not readily visible during both plant operations and refueling outages will be inspected at intervals that provide reasonable assurance that aging effects are managed such that the applicable components will perform their intended function during the period of extended operation. • The program will apply physical manipulation techniques, in addition to visual inspection, to detect aging effects in elastomers and plastics. • The program will include acceptance criteria (e.g., threshold values for identified aging effects) to ensure that the need for corrective actions will be identified before a loss of intended functions. • The program will ensure that program documentation such as walkdown records, inspection results, and other records of monitoring and trending activities are auditable and retrievable. [Revised in letter dated 2/6/2009 in response to RAI B2.1.14-1 Follow Up question]		
12	The Fire Protection Program will be enhanced to require periodic visual inspection of the fire barrier walls, ceilings, and floors to be performed during walkdowns at least once every refueling cycle. [Revised in letter dated 12/5/2008 in response to RAI B2.1.15-3]	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.15
13	The Fire Water System Program will be enhanced as follows:	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.16

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	• The program will be expanded to include eight additional yard fire hydrants in the scope of the annual visual inspection and flushing activities. • The program will require that sprinkler heads that have been in place for 50 years will be replaced or a representative sample of sprinkler heads will be tested using the guidance of NFPA 25, "Inspection, Testing and Maintenance of Water-Based Fire Protection Systems" (2002 Edition, Section 5.3.1.1.1). Sample testing, if performed, will continue at a 10-year interval following the initial testing.		
14	The Flux Thimble Tube Inspection Program will be enhanced as follows: • The program will require that the interval between inspections be established such that no flux thimble tube is predicted to incur wear that exceeds the established acceptance criteria before the next inspection. • The program will require that re-baselining of the examination frequency be justified using plant-specific wear rate data unless prior plant-specific NRC acceptance for the re-baselining was received. If design changes are made to use more wear-resistant thimble tube materials, sufficient inspections will be conducted at an adequate inspection frequency for the new materials. • The program will require that flux thimble tubes that cannot be inspected must be removed from service.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.18

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
15	The Fuel Oil Chemistry Program will be enhanced as follows: • Particulate contamination testing of fuel oil in the eleven fuel oil storage tanks in scope of License Renewal will be performed, in accordance with ASTM D 6217, on an annual basis. • One-time ultrasonic thickness measurements will be performed at selected tank bottom and piping locations prior to the period of extended operation.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.19
16	A Fuse Holders Program will be implemented. Program features will be as described in LRA Section B2.1.20.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.20
17	An Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will be implemented. Program features will be as described in LRA Section B2.1.21	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.21
18	An Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program will be implemented. Program features will be as described in LRA section B2.1.22. Inspections for stress corrosion cracking will be performed by visual examination with a magnified resolution as described in 10 CFR 50.55a(b)(2)(xxi)(A) or with ultrasonic methods. [Revised in letter dated 2/6/2009 in response to RAI B2.1.22-1 Follow Up question]	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.22
19	The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program will be enhanced as follows:	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.23

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	• Program implementing procedures will be revised to ensure the components and structures subject to inspection are clearly identified. • Program inspection procedures will be enhanced to include the parameters corrosion and wear where omitted.		
20	A Metal-Enclosed Bus Program will be implemented. Program features will be as described in LRA Section B2.1.26.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.26
21	Number Not Used [Revised in letter dated 3/27/2009]		
22	Number Not Used [Revised in letter dated 4/13/2009]		
23	A One-Time Inspection Program will be completed. Program features will be as described in LRA Section B2.1.29.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.29
24	A One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program will be completed. Program features will be as described in LRA Section B2.1.30.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.30
25	A. A PWR Vessel Internals Program will be implemented. Program features will be as described in LRA Section B2.1.32. B. An inspection plan for reactor internals will be submitted for NRC review and approval at least twenty-four months prior to the period of extended operation. In addition, the submittal will include any necessary revisions to the PINGP PWR Vessel	A. U1 - 8/9/2013 U2 - 10/29/2014 B. U1 - 8/9/2011 U2 - 10/29/2012	B2.1.32

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	Internals Program, as well as any related changes to the PINGP scoping, screening and aging management review results for reactor internals, to conform to the NRC-approved Inspection and Evaluation Guidelines. [Revised in letter dated 5/12/2009] [Revised in letter dated 6/24/09 in response to Follow-up RAI B2.1.38]		
26	The Reactor Head Closure Studs Program will be enhanced to incorporate controls that ensure that any future procurement of reactor head closure studs will be in accordance with the material and inspection guidance provided in NRC Regulatory Guide 1.65.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.33
27	The Reactor Vessel Surveillance Program will be enhanced as follows: • A requirement will be added to ensure that all withdrawn and tested surveillance capsules, not discarded as of August 31, 2000, are placed in storage for possible future reconstitution and use. • A requirement will be added to ensure that in the event spare capsules are withdrawn, the untested capsules are placed in storage and maintained for future insertion.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.34
28	The RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Program will be enhanced as follows: • The program will include inspections of concrete and steel components that are below the water line at the Screenhouse and Intake Canal. The scope will also	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.35

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	require inspections of the Approach Canal, Intake Canal, Emergency Cooling Water Intake, and Screenhouse immediately following extreme environmental conditions or natural phenomena including an earthquake, flood, tornado, severe thunderstorm, or high winds. • The program parameters to be inspected will include an inspection of water-control concrete components that are below the water line for cavitation and erosion degradation. • The program will visually inspect for damage such as cracking, settlement, movement, broken bolted and welded connections, buckling, and other degraded conditions following extreme environmental conditions or natural phenomena.		
29	A Selective Leaching of Materials Program will be completed. Program features will be as described in LRA B2.1.36.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.36
30	The Structures Monitoring Program will be enhanced as follows: • The following structures, components, and component supports will be added to the scope of the inspections: ○ Approach Canal ○ Fuel Oil Transfer House ○ Old Administration Building and Administration Building Addition ○ Component supports for cable tray, conduit, cable, tubing tray, tubing, non-ASME vessels, exchangers, pumps, valves, piping, mirror	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.38

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	insulation, non-ASME valves, cabinets, panels, racks, equipment enclosures, junction boxes, bus ducts, breakers, transformers, instruments, diesel equipment, housings for HVAC fans, louvers, and dampers, HVAC ducts, vibration isolation elements for diesel equipment, and miscellaneous electrical and mechanical equipment items o Miscellaneous electrical equipment and instrumentation enclosures including cable tray, conduit, wireway, tube tray, cabinets, panels, racks, equipment enclosures, junction boxes, breaker housings, transformer housings, lighting fixtures, and metal bus enclosure assemblies o Miscellaneous mechanical equipment enclosures including housings for HVAC fans, louvers, and dampers o SBO Yard Structures and components including SBO cable vault and bus duct enclosures. o Fire Protection System hydrant houses o Caulking, sealant and elastomer materials o Non-safety related masonry walls that support equipment relied upon to perform a function that demonstrates compliance with a regulated event(s). • The program will be enhanced to include additional inspection parameters. • The program will require an inspection frequency of once every five (5) years for structures and structural		

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	components within the scope of the program. The frequency of inspections can be adjusted, if necessary, to allow for early detection and timely correction of negative trends. • The program will require periodic sampling of groundwater and river water chemistries to ensure they remain non-aggressive.		
31	A Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program will be implemented. Program features will be as described in LRA Section B2.1.39.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.39
32	The Water Chemistry Program will be enhanced as follows: • The program will require increased sampling to be performed as needed to confirm the effectiveness of corrective actions taken to address an abnormal chemistry condition. • The program will require Reactor Coolant System dissolved oxygen Action Level limits to be consistent with the limits established in the EPRI PWR Primary Water Chemistry Guidelines." [Revised in letter dated 12/5/2008 in response to RAI B2.1.40-3]	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.40
33	The Metal Fatigue of Reactor Coolant Pressure Boundary Program will be enhanced as follows: • The program will monitor the six component locations identified in NUREG/CR-6260 for older vintage Westinghouse plants, either by tracking the cumulative number of imposed stress cycles using cycle counting, or	U1 - 8/9/2013 U2 - 10/29/2014	B3.2

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	by tracking the cumulative fatigue usage, including the effects of coolant environment. The following locations will be monitored: o Reactor Vessel Inlet and Outlet Nozzles o Reactor Pressure Vessel Shell to Lower Head o RCS Hot Leg Surge Line Nozzle o RCS Cold Leg Charging Nozzle o RCS Cold Leg Safety Injection Accumulator Nozzle o RHR-to-Accumulator Piping Tee • Program acceptance criteria will be clarified to require corrective action to be taken before a cumulative fatigue usage factor exceeds 1.0 or a design basis transient cycle limit is exceeded. [Revised in letter dated 1/9/2009 in response to RAI 4.3.1.1-1]		
34	Reactor internals baffle bolt fatigue transient limits of 1835 cycles of plant loading at 5% per minute and 1835 cycles of plant unloading at 5% per minute will be incorporated into the Metal Fatigue of Reactor Coolant Pressure Boundary Program and USAR Table 4.1-8.	U1 - 8/9/2013 U2 - 10/29/2014	B3.2
35	NSPM will perform an ASME Section III fatigue evaluation of the lower head of the pressurizer to account for effects of insurge/outsurge transients. The evaluation will determine the cumulative fatigue usage of limiting pressurizer component(s) through the period of extended operation. The analyses will account for periods of both "Water Solid" and "Standard Steam Bubble" operating strategies. Analysis results will be incorporated, as applicable, into the Metal Fatigue of Reactor	U1 - 8/9/2013 U2 - 10/29/2014	4.3.1.3

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	Coolant Pressure Boundary Program. [Revised in letter dated 1/9/2009 in response to RAI 4.3.1.1-1]		
36	NSPM will complete fatigue calculations for the pressurizer surge line hot leg nozzle and the charging nozzle using the methodology of the ASME Code (Subsection NB) and will report the revised CUFs and CUFs adjusted for environmental effects at these locations as an amendment to the PINGP LRA. Conforming changes to LRA Section 4.3.3, "PINGP EAF Results," will also be included in that amendment to reflect analysis results and remove references to stress-based fatigue monitoring. [Added in letter dated 1/9/2009 in response to RAI 4.3.1.1-1]	April 30, 2009 Commitment closed by letter dated 4/28/09	4.3.3
37	NSPM will revise procedures for excavation and trenching controls and archaeological, cultural and historic resource protection to identify sensitive areas and provide guidance for ground-disturbing activities. The procedures will be revised to include drawings and illustrations to assist users in identifying culturally sensitive areas, and pictures of artifacts that are prevalent in the area of the Plant site. The revised procedures will also require training of the Site Environmental Coordinator and other personnel responsible for proper execution of excavation or other ground-disturbing activities. [Added in ER revision submitted in letter dated 3/4/2009]	8/9/2013	ER 4.16.1
38	NSPM will conduct a Phase I Reconnaissance Field Survey of the disturbed areas within the Plant's boundaries. In addition, NSPM will conduct Phase I field surveys of areas of known archaeological sites to precisely determine their boundaries.	8/9/2013	ER 4.16.2

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	NSPM will use the results of these surveys to designate areas for archaeological protection. [Added in ER revision submitted in letter dated 3/4/2009]		
39	NSPM will prepare, maintain and implement a Cultural Resources Management Plan (CRMP) to protect significant historical, archaeological, and cultural resources that may currently exist on the Plant site. In connection with the preparation of the CRMP, NSPM will conduct botanical surveys to identify culturally and medicinally important species on the Plant site, and incorporate provisions to protect such plants into the CRMP. [Added in ER revision submitted in letter dated 3/4/2009]	8/9/2013	ER 4.16.2
40	NSPM will consult with a qualified archaeologist prior to conducting any ground-disturbing activity in any area designated as undisturbed and in any disturbed area that is described as potentially containing archaeological resources (as determined by the Phase I Reconnaissance Field Survey discussed in Commitment Number 38). [Added in ER revision submitted in letter dated 3/4/2009]	8/9/2013	ER 4.16.2
41	During the first refueling outage following refueling cavity leak repairs in each Unit (scheduled for refueling outages 1R26 and 2R26), concrete will be removed from the sump C pit to expose an area of the containment vessel bottom head. Visual examination and ultrasonic thickness measurement will be performed on the portions of the containment vessels exposed by the excavations. An assessment of the condition of exposed	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.38

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	concrete and rebar will also be performed. Degradation observed in the exposed containment vessel, concrete or rebar will be entered into the Corrective Action Program and evaluated for impact on structural integrity and identification of additional actions that may be warranted. [Added in letter dated 4/6/09 in response to Follow Up RAI B2.1.38]		
42	During the two consecutive refueling outages following refueling cavity leak repairs in each Unit (scheduled for refueling outages 1R26 and 2R26), visual inspections will be performed of the areas where reactor cavity leakage had been observed previously to confirm that leakage has been resolved. The inspection results will be documented. If refueling cavity leakage is again identified, the issue will be entered into the Corrective Action Program and evaluated for identification of additional actions to mitigate leakage and monitor the condition of the containment vessel and internal structures. [Added in letter dated 4/6/09 in response to Follow Up RAI B2.1.38]	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.38
43	Preventive maintenance requirements will be implemented to require periodic replacement of rubber flexible hoses in the Diesel Generators and Support System and in the 122 Diesel Driven Fire Pump that are exposed to fuel oil or lubricating oil internal environments. [Added in letter dated 4/6/09 in response to RAI 3.3.2-8-1] [Revised in letter dated 6/5/09]	U1 - 8/9/2013 U2 - 10/29/2014	Table 3.3.2-8