PrairieIslandNPEm Resource

From: RidsNrrPMPrairieIsland Resource
Sent: Monday, January 12, 2009 10:39 AM
To: Richard Plasse
Subject: FW: EIE Submittal
Attachments: E090109t183001_LPI09004 Responses to RAIs dated 12102008.pdf

Rick - I believe these are license renewal docs…

Tom

From: Kenny Nguyen
Sent: Monday, January 12, 2009 8:50 AM
To: marlys.davis@xenuclear.com; RidsNrrPMPrairieIsland Resource; James McKnight; Kenny Nguyen; Michael Collins; Mike King (OIS)
Subject: EIE Submittal

Good morning,

The att'd EIE submittals were not submitted in accordance with specifications contained in NRC's electronic submission rulemaking (and it's associated guidance document; effective January 1, 2004) which can be accessed at the following webpage:

http://www.nrc.gov/site-help/e-submittals/guid-elec-submission.pdf

Specifically the **PDF files are not Fast Web View enabled (issue a "File->Save As" (can overwrite the same file) will fix this).**

Please fix the PDF and re-submit (including enclosures) in accordance with NRC Submission Guidance. Please include in the comment field that this is a re-submittal.

If you have any questions, please contact me at 301-415-2046 or e-mail kdn@nrc.gov.

We appreciate your participation in the NRC EIE program.

PS. If you're using Adobe 7.0 Pro, you might want to download and install the latest revision. As we found out, if you're using 7.0.0-7.0.3, you will get a wrong result, such as Fast Web View is enabled while in fact it is NOT enabled.

Thanks,
Kenny Nguyen
IT Specialist/Project Officer
301-415-2046

Hearing Identifier: Prairie_Island_NonPublic
Email Number: 310

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From: RidsNrrPMPrairieIsland Resource

Created By: RidsNrrPMPrairieIsland.Resource@nrc.gov

Recipients:
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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

Responses to NRC Requests for Additional Information Dated December 10, 2008 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. In a letter dated December 10, 2008, the NRC transmitted Requests for Additional Information (RAIs) regarding that application. This letter provides responses to those RAIs.

Enclosure 1 provides the text of each RAI followed by the NSPM response.

Enclosure 2 provides an updated version of the Preliminary License Renewal Commitment List contained in the LRA transmittal letter. This updated list reflects changes made to date in the various NSPM letters responding to NRC RAIs.

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 adds one new commitment and revises two commitments in the list of Preliminary License Renewal Commitments contained in the LRA transmittal letter dated April 11, 2008.

Preliminary License Renewal Commitment No. 36 is added as follows:

> 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.

This amendment will be submitted by April 30, 2009.

Preliminary License Renewal Commitment No. 33 is revised as follows:

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 by tracking the cumulative fatigue usage, including the effects of coolant environment. The following locations will be monitored:
  - Reactor Vessel Inlet and Outlet Nozzles
  - Reactor Pressure Vessel Shell to Lower Head
  - RCS Hot Leg Surge Line Nozzle
  - RCS Cold Leg Charging Nozzle
  - RCS Cold Leg Safety Injection Accumulator Nozzle
  - 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.

These enhancements will be implemented prior to the period of extended operation.

Preliminary License Renewal Commitment No. 35 is revised as follows:

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 Coolant Pressure Boundary Program.

This analysis and any associated program enhancements will be completed prior to the period of extended operation.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on January 9, 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
NSPM Responses to NRC Requests for Additional Information
Dated December 10, 2008

RAI 4.3.1.1-1

Some of the reported CUF results were generated by FatiguePro software. However, FatiguePro is not endorsed by NRC staff, since FatiguePro does not produce all six individual components of a transient stress tensor (Sxx, Syy, Szz, Sxy, Syz, Szx) to support the American Society of Mechanical Engineers (ASME) Section III fatigue analysis method. FatiguePro produces only one stress component and uses that single stress component for fatigue evaluation.

Please identify from Tables 4.3-2 through 4.3-8, if any, those items whose CUF values were calculated without using every individual component of the transient stress tensors. The items that are identified must be reevaluated in accordance with the ASME Section III guidelines so that the fatigue results are valid and applicable to the period of extended operation. In addition, the staff noticed multiple occurrences of the terminology "stress-based monitoring" in the body of the LRA. If the plant does not have appropriate stress monitoring capability, use of such a terminology would be misleading. Please make appropriate corrections.

NSPM Response to RAI 4.3.1.1-1

LRA Section 4.3.1 (Page 4.3-2), "Cumulative Usage Factors" states that the cumulative usage factors reported in Section 4.3.1, except as otherwise noted, are the design basis values and do not consider the effects of reactor water environment on fatigue life. The CUFs reported in Tables 4.3-2, 4.3-3, 4.3-4, 4.3-5, 4.3-6, and 4.3-7 are all calculated in accordance with ASME Section III using the values of six stress components versus time for the complete stress cycle, taking into account both the gross and local structural discontinuities and the thermal effects which vary during the cycle.

The calculations of Environmentally-Assisted Fatigue in Section 4.3.3 of the LRA reported in Table 4.3-8 are based on ASME Section III CUFs with the exception of the pressurizer surge line hot leg nozzle safe end and the charging system nozzle for each Unit. The unadjusted CUFs at both locations are based on FatiguePro stress-based fatigue analyses. NSPM is in the process of performing ASME Code (Subsection NB) compliant fatigue calculations for the pressurizer surge line hot leg nozzle and the charging nozzle 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. This amendment will be submitted by April 30, 2009. This action is identified as Preliminary License Renewal Commitment No. 36.

In addition, to clarify NSPM's intention in regard to evaluating the effects of insurge/outsurge transients on the lower head of the pressurizer (and to eliminate a reference to stress-based fatigue monitoring), the following LRA changes are made:

In LRA Section 4.3.1.3, the last paragraph on Page 4.3-10 and the first paragraph on Page 4.3-11 are deleted and replaced in their entirety with the following:

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 Coolant Pressure Boundary Program. This analysis and any associated program enhancements will be completed prior to the period of extended operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program will manage metal fatigue of the pressurizer due to insurge/outsurge transients in accordance with 10 CFR 54.21(c)(1)(iii).

The following additional LRA changes are also being made to eliminate references to stress-based fatigue monitoring:

In LRA Section A3.2, on Page A-17, the third sentence of the first paragraph is deleted and replaced in its entirety with the following: "The program also tracks fatigue usage in critical high-usage components."

In LRA Section B3.2, Program Description, on page B-84, the third sentence of the first paragraph is deleted and replaced in its entirety with the following: "The program also tracks fatigue usage in critical high-usage components."

In LRA Section B3.2, Enhancements, on page B-85, the first bulleted enhancement is deleted and replaced in its entirety with the following:

- Scope of Program, Preventive Actions, Parameters Monitored/Inspected, Detection of Aging Effects, Monitoring and Trending

  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 by tracking the cumulative fatigue usage, including the effects of coolant environment. The following locations will be monitored:

NUREG/CR-6260 Location
Reactor Pressure Vessel Inlet and Outlet Nozzles
Reactor Pressure Vessel Shell to Lower Head
RCS Hot Leg Surge Line Nozzle
RCS Cold Leg Charging Nozzle
RCS Cold Leg Safety Injection Accumulator Nozzle
RHR-to-Accumulator Piping Tee

To reflect the changes discussed above, two Preliminary License Renewal Commitments included in the LRA transmittal letter dated April 11, 2008, are also revised.

Preliminary License Renewal Commitment No. 33 is revised to read as follows:

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
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 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.	U1 - 8/9/2013 U2 - 10/29/2014	B3.2

Preliminary License Renewal Commitment No. 35 is revised to read as follows:

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
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 Coolant Pressure Boundary Program.	U1 - 8/9/2013 U2 - 10/29/2014	4.3.1.3

RAI 4.3.1.2-1

The design cycles and the projected 60-year cycles for Prairie Island Nuclear Generating Plant's (PINGP's) design basis transients are shown in Table 4.3-1 of the LRA. The staff has noted that the paragraph under LRA Table 4.3-4 of the LRA, which discusses the plant loading and unloading transient cycles, is vague. In the targeted paragraph, an unrelated/unrecognized number, 1835, which is not shown in Table 4.3-1, is suddenly brought up. Clarify and indicate the source of this 1835 number and clarify whether this value represents the new design basis cycle values for the baffle bolts.

NSPM Response to RAI 4.3.1.2-1

As part of the planned Measurement Uncertainty Recapture-Power Uprate (MUR-PU) project and fuel transition, NSPM has recently completed a fatigue evaluation of the Reactor Vessel (RV) lower internals. Fatigue analyses of the RV lower internals are not part of the PINGP CLB but were conservatively included to evaluate the proposed MUR-PU and fuel transition. Fatigue usage at all RV lower internal locations, with the exception of the baffle bolts, was determined using the full set of design transients defined in Table 4.3-1 of the PINGP LRA. For the baffle bolts, the number of plant loading and unloading transients at 5% of full power per minute was decreased from 18,300 to 1,835 to obtain a usage factor below 1.0. Therefore, the baffle bolts are limited to 1,835 loading and unloading cycles at 5% of full power per minute, which is a new design basis cycle value for the baffle bolts. As discussed in PINGP LRA Section B3.2, the Metal Fatigue of Reactor Coolant Pressure Boundary Program and PINGP USAR Table 4.1-8 will be enhanced to include this additional cyclic limit for baffle bolt fatigue.

RAI 4.3.1.3-1

To project the cycles and cumulative usage factor (CUF) to the period of extended operation, it is necessary to have the base line information. Before the dates of issuance of NRC Bulletins 88-08 and 88-11, there wasn't any data collected regarding stratification and insurge/outsurge events for pressurizer surge lines.

Please discuss how you reconstructed the cycles that occurred prior to December 20, 1988 (the date of issuance for Bulletin 88-11), for the pressurizer surge line stratification events and insurge/outsurge events. Please provide the date when the events tracking began and discuss how the tracked data was used to achieve the 60-year cycle projection.

NSPM Response to RAI 4.3.1.3-1

As described in Section 4.3.1.6 of the PINGP LRA (Page 4.3-16), the site-specific evaluations of the pressurizer surge line (i.e., WCAP-12839 for Unit 1 and WCAP-12639 for Unit 2) are considered TLAAs since the evaluations use time-limited assumptions such as thermal and pressure transients, and operating cycles. The dominant event cycles that contribute to fatigue in the surge line analyses are the 200 heatups and cooldowns that include stratification and striping in the pressurizer surge line. The surge line temperature transients during heatup and cooldown are characterized in WCAP-12839 and WCAP-12639 by maximum system differential temperatures between the pressurizer water and RCS hot leg that occur over five RCS temperature ranges. The system differential temperature ranges were used to define the stratification events and insurge/outsurge events for the purpose of the analyses.

In order to ensure that the pressurizer surge line analyses were bounding for PINGP, the actual system differential temperatures (maximum temperature differential between pressurizer and RCS hot leg) were obtained for each Unit heatup and cooldown since initial plant operation. Unit 1 differential temperature data was recorded beginning in 1973, and Unit 2 differential temperature data was recorded beginning in 1974. Following completion of the pressurizer surge line thermal stratification analyses in the early 1990s, PINGP continues to monitor temperature differentials between the pressurizer water and RCS hot leg as required by the Metal Fatigue of Reactor Coolant Pressure Boundary Program to ensure that plant operation is within the bounds of the pressurizer surge line transient definitions contained in WCAP-12839 and WCAP-12639.

As stated previously, the dominant event cycles that contribute to surge line stratification and insurge/outsurge events are the plant heatups and cooldowns, which are limited by design to 200. As described in Section 4.3.1 and Table 4.3-1 of the PINGP LRA, the number of heatups and cooldowns projected to be accumulated after 60 years of operation (i.e., approximately 125) is less than the number of cycles used to calculate fatigue of the surge line in WCAP-12839 for Unit 1 and WCAP-12639 for Unit 2.

RAI 4.3.1.4-1

Table 4.3-6 of the LRA shows the CUF values at certain components in the steam generators. It is understood that the CUF values for Unit 1, reflect steam generator replacement (occurred in November 2004), therefore the CUF at the corresponding components/locations of Units 1 and 2, are expected to be somewhat different. However, for the primary inlet nozzle and primary outlet nozzle, the differences between Unit 1 and Unit 2, are significant. Specifically, at these two nozzles, the CUF is as high as 0.880 for the original steam generator (Unit 2) but it is dropped to near zero, 0.007, for the replaced one (Unit 1). Explain how this is possible. Specify the analytical tool used along with a discussion concerning differences in the input data.

NSPM Response to RAI 4.3.1.4-1

As noted in Section 4.3.1.4 of the PINGP LRA (Page 4.3-11), the primary and secondary sides of the steam generators are Class 1 and were designed in accordance with ASME Section III, 1995 Edition through 1996 Addenda for Unit 1 and 1965 Edition through Winter 1966 Addenda for Unit 2. The cumulative usage factors for the steam generator primary inlet and outlet nozzles were calculated by AREVA NP for the Unit 1 replacement steam generators (RSGs) in 2004, and by Westinghouse Electric Corporation for the Unit 2 original steam generators (OSGs). The differences in reported cumulative usage factors (CUFs) in the nozzle base metal may be attributed to one or a combination of the following: (1) differences between the Unit 1 and Unit 2 nozzle geometry and materials of construction, (2) differences in design transients and external nozzle loads and moments used in the fatigue evaluations, or (3) differences in methodology of CUFs. Each potential difference is discussed below.

The Unit 1 primary inlet and outlet nozzles are forged with the primary head and are fabricated from low alloy steel (SA 508 Grade 3 Class 2). The Unit 2 primary inlet and outlet nozzles were cast with the primary head and are fabricated from carbon steel (SA 216 Grade WCC). The thicknesses of the Unit 1 nozzles are less than the Unit 2 nozzles due to differences in materials of construction. The geometry of the Unit 1 and Unit 2 nozzles are both similar to ASME Section III, Figure NB-3338.2(a)-2, Figure (c), with the only significant difference being the nozzle thicknesses. Due to differences in materials of construction and wall thickness the Unit 1 nozzles will respond quicker to design temperature transients, through-wall gradients will be reduced, and thermal stresses are expected to be lower than the Unit 2 nozzles when exposed to equivalent thermal transients.

The design transients used in the fatigue evaluation of the Unit 1 and Unit 2 inlet and outlet nozzles are based on the Westinghouse standard NSSS design transients and are consistent between Units 1 and 2. These design transients are listed in Table 4.3-1 of the PINGP LRA. External loads and moments applied to the nozzles are in accordance with design specifications for both units. For the Unit 1 RSGs it was necessary to perform a new reactor coolant loop analysis with the RSGs explicitly modeled. The resultant loads generated for the vessel nozzles were compared to those provided for the original steam generator. Bounding loads were selected and used in the stress analysis of the Unit 1 RSG inlet and outlet nozzles. Therefore, the loads and moments applied to the Unit 1 nozzles are equivalent to or bound the loads and moments used for the OSG Unit 2 stress analysis of the nozzles.

The calculation of the primary nozzle CUFs performed in 2004 for Unit 1 is based on finite element analysis to determine temperature fields in thermal calculations, and stress and displacement fields in mechanical or thermo-mechanical calculations. The finite element analysis results are interpreted using post-processing programs to permit stress analyses in accordance with ASME Section III, Subsection NB. Usage factors were calculated in accordance with NB-3222.4 (e)(5) using loadset pairs based on the NSSS design transients. The maximum calculated CUF of 0.007 for the primary inlet nozzle was calculated at the inside surface of the primary inlet nozzle base metal.

The calculation of the primary nozzle CUFs performed in 1969 for Unit 2 is based on photoelastic and thermal transient analyses considering the worst case normal and upset loading conditions. The maximum peak stress intensity considering worst case normal and upset conditions was conservatively applied for the total number of design cyclic applications (i.e., 24,000 cyclic applications for all transients the unit must undergo, which bounds the applicable design transients in Table 4.3-1 of the PINGP LRA). This resulted in a very conservative cumulative usage factor of 0.88 for the inside surface of the primary inlet nozzle base metal.

In summary, the differences between the usage factors for the Unit 1 and Unit 2 steam generator primary inlet and outlet nozzles are attributed to differences in materials of construction, wall thickness, and calculation methods used to perform the CUF calculations. The Unit 2 evaluation was performed using a peak stress intensity for the worst case transient applied to the NSSS design transients, whereas the Unit 1 calculation considered individual usage contributions due to loadset pairs for the NSSS design transients. In addition, the Unit 2 evaluation was performed prior to the development of detailed finite element stress analysis computer codes and was very conservative.

RAI 4.3.1.4-2

In LRA Section 4.3.1.4, the applicant discusses fracture mechanics analyses for the steam generator feedwater nozzle and performed fracture mechanics analyses to justify leaving the eroded thermal sleeves in the as-found condition. Clarify why the fracture mechanics evaluation/flaw evaluation for the steam generator feedwater nozzles are considered to be a time-limited aging analysis (TLAA) that meets the definition for TLAAs in 10 CFR 54.3.

NSPM Response to RAI 4.3.1.4-2

The Steam Generator Fatigue and Fracture Mechanics Evaluation of Feedwater Inlet Nozzle discussion was conservatively included in LRA Section 4.3.1.4 even though the analysis did not meet all six criteria in 10 CFR 54.3(a) for defining a TLAA. In particular, the analysis did not meet criterion (3) in that it did not "Involve time-limited assumptions defined by the current operating term, for example, 40 years." The crack growth analysis does not provide a basis for demonstrating that a known flaw is acceptable for continued operation for the life of the plant. The analysis simply defined an appropriate examination frequency (nominally each 100 days of operation in Modes 2 or 3) that is based on a postulated flaw of a certain size. There were no known flaws in the steam generator feedwater nozzle region of interest. Therefore, since the crack growth analysis is not managing an actual (existing) crack and the analysis was not performed for the service life of the component (i.e., 40 years), this evaluation is not a TLAA.

NSPM monitors the Unit 2 feedwater nozzle to pipe transition forging welds for evidence of cracking using ultrasonic inspection through owner elected examinations maintained within the PINGP ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. (Refer to "Submittal of the 4th Interval Inservice Inspection Plan – Units 1 & 2", June 21, 2004, L-PI-04-076, ADAMS Accession Number ML041800468.) The periodic examinations ensure that the feedwater nozzle region remains free of cracking.

NSPM has determined that this discussion should be removed from the LRA. Accordingly, the following changes are made:

In LRA Table 4.1-2 on Page 4.1-8, the line item for inservice flaw growth analyses that demonstrate structural stability for 40 years is hereby revised to appear as follows:

NUREG-1800 Generic TLAA Examples	Applicability to PINGP	LRA Section
Inservice flaw growth analyses that demonstrate structural stability for 40 years	No - Did not meet TLAA criteria	-

In LRA Section 4.3.1.4 on Pages 4.3-12 and 4.3-13, the section entitled "Fatigue and Fracture Mechanics Evaluation of Feedwater Inlet Nozzle" is hereby deleted in its entirety.

RAI 4.3.1.5-1

In Section 4.3.1.5 of the LRA, the applicant states that "Exemption from fatigue evaluation was justified for the casing feet, casing nozzle, and upper and lower seal housings and bolts," without supporting this statement with a regulatory basis. Provide your technical and regulatory basis for exempting the reactor coolant pump casing feet, casing nozzle, and upper and lower seal housings and bolts from being within the scope of an ASME Section III CUF analysis.

NSPM Response to RAI 4.3.1.5-1

As described in Section 4.3.1.5 of the PINGP LRA, the reactor coolant pumps (RCPs) were designed in accordance with Article 4 of ASME Section III through the 1970 Winter Addendum. Exemption from fatigue evaluation was justified in accordance with ASME Section III, Article 415.1, for the casing feet, casing nozzle, and upper and lower seal housings and bolts. The applicable RCP calculations to show compliance with Article N-415.1, (a) through (f) are based on design transients that bound those presented in LRA Table 4.3-1, and are intended to represent 40 years of operation. The numbers of analyzed design transients used in the RCP exemption from fatigue analyses required by Article N-415.1 for the casing feet, casing nozzle, and upper and lower seal housings and bolts, will not be exceeded in 60 years of operation, and these TLAAs will remain valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

RAI 4.3.3-1

It is known that some of the reactor coolant system piping charging nozzles are designed and manufactured with a thermal sleeve and some are not.

Please confirm whether or not a thermal sleeve exists in the charging nozzle, and confirm that the stress and fatigue analyses were performed based on the actual geometric conditions of the charging nozzle in use, so that the fatigue results are valid and applicable to the period of extended operation.

NSPM Response to RAI 4.3.3-1

Each Unit at PINGP has one 2.0 inch stainless steel charging line that connects to a 27.5 inch I.D. stainless steel reactor coolant cold leg pipe. Each charging line to RCS pipe connection is made using a stainless steel half coupling (i.e., charging nozzle) that contains a thermal sleeve fabricated from Type 316 stainless steel. Both charging line to RCS cold leg pipe connections at PINGP contain a thermal sleeve.

These connections were originally designed in accordance with B31.1.0 and did not include cumulative usage factor evaluations. In order to address environmentally-assisted fatigue for license renewal, PINGP performed stress-based fatigue analyses of the charging nozzles using FatiguePro as described in Section 4.3.3 of the PINGP LRA (Page 4.3-24). The charging nozzle finite element model used in the FatiguePro evaluation was developed using the appropriate geometry of the PINGP charging nozzles and thermal sleeves, as detailed on plant fabrication drawings. As discussed in the response to RAI 4.3.1.1-1 above, however, NSPM has elected to perform an ASME Section III, NB-3200, fatigue analysis of the charging nozzles. The updated usage factor will be used to evaluate the effects of reactor water environment on the fatigue life of the charging nozzles. The finite element model associated with the ASME Section III analyses will be based upon the charging nozzle geometry as defined by PINGP drawings.

RAI 4.3.3-2

LRA Section 4.3.3 discusses environmentally assisted fatigue (GSI-190) and provides values of the environmentally assisted fatigue correction factors, F_{en}, applicable to PINGP. In the calculation of F_{en} for the low alloy steels, the applicant assumes that the dissolved oxygen level for pressurized-water reactor plants is below 0.05 ppm at temperatures above 150 ^{o}C. This assumption in effect takes away the dependency of F_{en} on strain rate and sulfur content and also significantly weakens the dependency of F_{en} on temperature. Provide your basis for using this assumption.

NSPM Response to RAI 4.3.3-2

At PINGP, RCS temperatures above 150 ^{o}C (302 ^{o}F) correspond to operation in Mode 4 (Hot Shutdown, 350 ^{o}F > Tavg > 200 ^{o}F), Mode 3 (Hot Standby, Tavg $\geq$ 350 ^{o}F), Mode 2 (Startup), or Mode 1 (Power Operation). PINGP controls oxygen in the Reactor Coolant System (RCS) by maintaining a hydrogen overpressure in the Volume Control Tank. Minimum primary water hydrogen levels are maintained which are effective in mitigating oxidizing conditions due to radiolysis or oxygen ingress into the reactor coolant.

The control parameters for RCS dissolved oxygen are in accordance with the PINGP Water Chemistry Program and the PINGP Technical Requirements Manual (TRM) which specify a dissolved oxygen action level limit of less than 100 ppb in Modes 1, 2, 3, and Mode 4 with RCS Temperature $\geq$ 250 °F. Per the TRM, if the limit is exceeded, the dissolved oxygen shall be restored to less than 100 ppb within a 24 hour period or the Unit shall be shutdown. At PINGP, dissolved oxygen in the RCS is typically less than 5 ppb prior to criticality (Modes 2, 3, and 4) and 0 ppb during power operation (Mode 1). In addition, a review of PINGP RCS water chemistry data from 1999-2008 was performed. The review confirmed that the dissolved oxygen content in the RCS never exceeded 40 ppb (0.04 ppm) when the RCS temperature was greater than 300 °F. Therefore, given the past plant operating history and adherence to action level limits, it is reasonable to assume that the dissolved oxygen level in the RCS at PINGP has been and will continue to be maintained below 0.05 ppm (50 ppb) at RCS temperatures above 150 °C (302 °F).

RAI 4.7.4-1

License renewal application (LRA) Section 4.7.4 indicates that the polar cranes, auxiliary building cranes, the turbine building cranes, and spent fuel cranes will be used less than 20,000 lifting cycles over 60 years.

Provide your estimate for the number of lifting cycles that have occurred in each of the stated cranes and your 60-year lifting cycle projections for these cranes.

NSPM Response to RAI 4.7.4-1

The NUREG-0612 cranes with fatigue TLAAs include the polar cranes, auxiliary building crane, the turbine building cranes, and spent fuel crane. These cranes are described in Section 12.2.12 of the PINGP USAR.

<u>Polar Crane, Auxiliary Building Crane, Turbine Building Crane</u>

In accordance with the Northern States Power Company (NSP) letter to the NRC dated November 8, 1982, titled "Control of Heavy Loads (Response to Staff Concerns on the Six Month Submittal)", in the Response to Inquiry 4 on page 8, the evaluation of fatigue for the polar crane, auxiliary building crane, and turbine building crane assumed 800 design loading cycles for the heaviest load over 40 years. This allowed for two outages per year and ten lifts of the heaviest load per outage over 40 years. Multiplying 800 heavy load cycles by 1.5 to accommodate 60 years of operation yields a projection of 1200 heavy load cycles through the period of extended operation, providing significant margin to the design load cycle limit of 20,000 cycles.

At present (January 2009), both PINGP Units have completed 25 refueling outages. Multiplying 10 lifts of the heaviest load per outage by 25 refueling outages yields an estimate of 250 heavy load cycles to date for the polar cranes and turbine building cranes, and an estimate of 500 heavy load cycles to date for the auxiliary building crane since it services both Units. Based upon the estimated number of heavy lifting cycles accrued to date, it is not expected that these cranes will attain the projected number of lifts (i.e., 1200) after 60 years of operation.

Spent Fuel Crane

In accordance with the NSP letter to the NRC dated November 8, 1982, titled "Control of Heavy Loads (Response to Staff Concerns on the Six Month Submittal)", Attachment 1, in the Response to Inquiry 4 on page 1-4, the evaluation of fatigue for the spent fuel crane assumed 800 design loading cycles over 40 years. The weight of the largest actual load, 3700 lbs., is substantially less than the rated crane capacity of 6000 lbs. The use of 800 design load cycles allowed for two outages per year and ten lifts of the heaviest load per outage over 40 years. Multiplying 800 heavy load cycles by 1.5 to accommodate 60 years of operation yields a projection of 1200 heavy load cycles through the period of extended operation, providing significant margin to the design load cycle limit of 20,000 cycles.

At present (January 2009), both PINGP Units have completed 25 refueling outages. Since the single spent fuel crane services both Units, it is expected to have been utilized for 50 refueling outages to date. Multiplying 10 lifts of the heaviest load per outage by 50 refueling outages yields an estimate of 500 heavy load cycles to date for the spent fuel crane. Based upon the estimated number of heavy lifting cycles accrued to date, it is not expected that the spent fuel crane will attain the projected number of lifts (i.e., 1200) after 60 years of operation.

RAI AMP B2.1.3-2

During review of LRA Section B2.1.3, "Operating Experience," the staff noted that there were two instances of cracking due to intergranular stress corrosion cracking (IGSCC) of Safety Injection Accumulator Tanks. Although these cracks were repaired and the In-service Inspection Program was augmented to include additional inspections, such as ultrasonic testing and liquid penetrant inspection, the applicant did not indicate if new activities were implemented to improve and monitor the environment to preclude IGSCC. Please provide details of any preventive activities to preclude IGSCC.

NSPM Response to RAI AMP B2.1.3-2

The leak detected in January 1988 in the Unit 2 Safety Injection 22 Accumulator Tank water level sensing line was a result of intergranular stress corrosion cracking (IGSCC) that initiated as a consequence of high stresses caused by the improper fit-up of the pipe to the nozzle in preparation for welding. The leak was repaired. Evaluation of the nozzle leakage resulted in the periodic inspection (ultrasonic and dye penetrant examinations) of the Unit 1 and 2 Safety Injection Accumulator nozzles on a 10-year frequency.

In May 2005, cracks were found in the level transmitter nozzles on the 21 Accumulator Tank in Unit 2. The affected nozzles were replaced. Metallurgical analysis indicated the cracking mechanism to be IGSCC initiated by sensitization of the stainless steel components during post-weld heat treatment of the tanks. The apparent cause evaluation concluded that the best method for the management of nozzle cracking was periodic monitoring, including internal visual inspection and dye penetrant examination, in addition to the external ultrasonic examination. The current examination frequency is once every third refueling outage.

The Safety Injection Accumulator nozzle examinations are owner elected examinations maintained within the PINGP ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. The nozzles are also periodically visually inspected (VT-2) for leakage as required by the IWB, IWC, and IWD Program.

The primary pressure boundary piping of PWRs has generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry. However, the potential for SCC exists due to inadvertent introduction of contaminants into the primary coolant system. The PINGP Water Chemistry Program periodically monitors water chemistry and controls detrimental contaminants (such as chlorides, fluorides, dissolved oxygen, and sulfate) to levels below those known to result in cracking. Plant procedures require water chemistry control in accordance with Revision 5 of the "PWR Primary Water Chemistry Guidelines", EPRI TR-1002884, for primary and auxiliary water systems. EPRI water chemistry guidelines are based on substantial industry experience and laboratory analysis. The program has been effective in controlling plant chemistry and taking required actions to address out-of-specification values.

In the examples described above, conditions were identified and corrected prior to causing any significant impact to safe operation or loss of intended functions. No additional preventive activities were implemented beyond continuing with the existing mitigative features of the Water Chemistry Program. The PINGP ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program and the Water Chemistry Program effectively manage the condition of the pressure retaining components within the license renewal boundary and ensure the associated aging effects such as cracking are adequately managed. Appropriate guidance is contained in PINGP procedures for indications of degradation requiring reevaluation, repair, or replacement.

Enclosure 2

Updated Preliminary License Renewal Commitment List

13 Pages

Preliminary License Renewal Commitments

The following table provides the list of preliminary commitments included in the Application for Renewed Operating Licenses (LRA) for Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2. These commitments reflect the contents of the LRA as submitted, and any updates provided in subsequent correspondence, but are considered preliminary in that the specific wording of some commitments may change, and additional commitments may be made, during the NRC review of the LRA.

The final commitments as submitted by NSPM, and accepted by NRC, are expected to be confirmed in the NRC's Safety Evaluation Report (SER) for the renewed operating licenses. The final commitments, as confirmed in the SER, will become effective upon NRC issuance of the renewed operating 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	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 U2 - 10/29/2014	B2.1.6

Preliminary License Renewal Commitments

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.		
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 an internal visual examination of accessible surfaces of components serviced by closed-cycle cooling water when the systems or components are opened during scheduled maintenance or surveillance activities.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.9
7	The Compressed Air Monitoring Program will be enhanced to require that Station and Instrument Air System air quality 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
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

Preliminary License Renewal Commitments

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
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 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.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.14
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	B2.1.16

Preliminary License Renewal Commitments

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.	U2 - 10/29/2014	
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
15	The Fuel Oil Chemistry Program will be enhanced as follows:	U1 - 8/9/2013	B2.1.19

Preliminary License Renewal Commitments

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	• 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.	U2 - 10/29/2014	
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.	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: • 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.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.23

Preliminary License Renewal Commitments

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
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	For the Nickel-Alloy Nozzles and Penetrations Program, PINGP commits to the following activities for managing the aging of nickel-alloy components susceptible to primary water stress corrosion cracking: • Comply with applicable NRC orders, and • Implement applicable NRC Bulletins, Generic Letters, and staff-accepted industry guidelines.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.27
22	The Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program will be enhanced as follows: • The program will require that any deviations from implementing the appropriate required inspection methods of the NRC First Revised Order EA-03-009, "Issue of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," dated February 20, 2004 (Order), as amended, will be submitted for NRC review and approval in accordance with the Order, as amended. • The program will require that any deviations from implementing the required inspection frequencies mandated by the Order, as amended, will be submitted for NRC review and approval in accordance with the Order, as amended. • The program will require that relevant flaw indications detected during the augmented inspections of the upper	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.28

Preliminary License Renewal Commitments

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	vessel head penetration nozzles will be evaluated in accordance with the criteria provided in the letter from Mr. Richard Barrett, NRC, Office of Nuclear Reactor Regulation (NRR), Division of Engineering to Alex Marion, Nuclear Energy Institute (NEI), dated April 11, 2003, or in accordance with NRC-approved Code Cases that incorporate the flaw evaluation procedures and criteria of the NRC's April 11, 2003, letter to NEI. • The program will require that, if leakage or evidence of cracking in the vessel head penetration nozzles (including associated J-groove welds) is detected while ranked in the “Low,” “Moderate,” or “Replaced” susceptibility category, the nozzles are to be immediately reclassified to the “High” susceptibility category and the required augmented inspections for the “High” susceptibility category are to be implemented during the same outage the leakage or cracking is detected.		
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	For the PWR Vessel Internals Program, PINGP commits to the following activities for managing the aging of reactor vessel internals components: • Participate in the industry programs for investigating and managing aging effects on reactor internals;	U1 - 8/9/2011 U2 - 10/29/2012	B2.1.32

Preliminary License Renewal Commitments

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	• Evaluate and implement the results of the industry programs as applicable to the reactor internals; and • Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
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	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.35

Preliminary License Renewal Commitments

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	steel components that are below the water line at the Screenhouse and Intake Canal. The scope will also 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	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.38

Preliminary License Renewal Commitments

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	○ Component supports for cable tray, conduit, cable, tubing tray, tubing, non-ASME vessels, exchangers, pumps, valves, piping, mirror 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 ○ 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 ○ Miscellaneous mechanical equipment enclosures including housings for HVAC fans, louvers, and dampers ○ SBO Yard Structures and components including SBO cable vault and bus duct enclosures. ○ Fire Protection System hydrant houses ○ Caulking, sealant and elastomer materials ○ 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.		

Preliminary License Renewal Commitments

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	• The program will require an inspection frequency of once every five (5) years for structures and structural 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	U1 - 8/9/2013 U2 - 10/29/2014	B3.2

Preliminary License Renewal Commitments

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	Westinghouse plants, either by tracking the cumulative number of imposed stress cycles using cycle counting, or 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

Preliminary License Renewal Commitments

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
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 Coolant Pressure Boundary Program. [Revised in letter dated 1/9/2009 in response to RAI 4.3.1.1-1]	U1 - 8/9/2013 U2 - 10/29/2014	4.3.1.3
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	4.3.3