May 22, 2009

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	Docket Nos. 50-282-LR
Northern States Power Co.	)	50-306-LR
	)	
(Prairie Island Nuclear Generating Plant,	)	ASLBP No. 08-871-01-LR
Units 1 and 2)	)	

NORTHERN STATES POWER COMPANY'S
MOTION TO DISMISS PIIC CONTENTION 7 AS MOOT

I. INTRODUCTION

Pursuant to 10 C.F.R. § 2.323(a), Northern States Power Company, a Minnesota corporation ("NSPM"), hereby moves this Atomic Safety and Licensing Board (the "Board") for dismissal of the Prairie Island Indian Community ("PIIC")'s Contention 7. PIIC Contention 7 relates to the adequacy of the Prairie Island Nuclear Generating Plant ("PINGP")'s program for managing the effects of aging due to embrittlement of the reactor vessel internals. NSPM moves for dismissal of PIIC Contention 7 because it has supplemented its License Renewal Application ("LRA") with the information whose omission was the basis for the Contention. PIIC Contention 7 is, therefore, moot.

II. PROCEDURAL BACKGROUND

The LRA for Operating License Nos. DPR-42 and DPR-60 for the PINGP Units 1 and 2 was submitted to the NRC by NSPM, formerly Nuclear Management Company, LLC, on April 11, 2008. On August 18, 2008, PIIC filed its "Notice of Intent to Participate and Petition to Intervene" ("PIIC Petition"). The PIIC Petition contained eleven separate contentions, including PIIC Contention 7. As proffered by the PIIC, Contention 7 alleged that the "PINGP license

Renewal Application does not include an adequate plan to monitor and manage the effects of aging due to embrittlement of the reactor pressure vessels and the associated internals." PIIC Petition at 27.

On December 5, 2008, the Board admitted seven of the PIIC's contentions, including Contention 7. Northern States Power Co. (Prairie Island Nuclear Generating Plant, Units 1 and 2), LBP-08-26, 68 N.R.C. __, (Dec. 5, 2008) ("LBP-08-26"). The Board's Order treated PIIC Contention 7 as asserting three separate claims: "(1) failure to consider embrittlement of the reactor vessel, (2) failure to consider embrittlement of the reactor vessel internals, and (3) failure to adequately describe the aging management program for the reactor vessel with regard to the Vessel Surveillance Program." LBP-08-26, slip op. at 42. The Board admitted only the second claim – regarding consideration of embrittlement of the reactor vessel internals – and expressly held the other two proffered claims inadmissible. Id. at 42-43. Thus, the modified form in which the Board admitted Contention 7 is: "The LRA does not contain an adequate plan to monitor and manage the effects of aging due to embrittlement of the reactor vessel internals." Id. at 44. As the Board expressed, PIIC Contention 7 is a "contention of omission." Id. at 42 & 43.

On May 12, 2009, NSPM filed with the NRC a supplement to its LRA; on the same date, as part of its consultation pursuant 10 C.F.R. § 2.323(b), NSPM provided a copy of the LRA supplement to the PIIC's counsel with an explanation that the information contained therein is intended to resolve the concerns expressed in PIIC Contention 7.[1] The supplement provides a

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[1] NSPM has therefore given the PIIC ten days to review the new information contained in the LRA supplement before filing this Motion. Under the Board's scheduling order, any new or amended contentions based upon new information are to be filed within thirty days of the date on which the new information becomes available. Memorandum and Order (Prehearing Conference Call Summary and Initial Scheduling Order) (Feb. 18, 2009), slip op. at 4. Any new or amended contention by the PIIC based upon the information contained in the LRA supplement would therefore be due on June 11, 2009, ten days after the date on which any answer to this motion would be due. 10 C.F.R. § 2.323(c). On May 11, 2009, this Board issued an order conditionally granting

detailed description of the new, plant-specific aging management program for PINGP's reactor vessel internals. This supplement is attached hereto as Exhibit A. The information included in the LRA supplement moots PIIC Contention 7 as admitted by the Board. Accordingly, the Contention should be dismissed.

III. A CONTENTION IS RENDERED MOOT IF AN APPLICANT CURES THE ALLEGED OMISSION FROM THE APPLICATION WHICH SERVED AS THE BASIS FOR THE CONTENTION

Where "a contention is 'superseded by the subsequent issuance of licensing-related documents'…the contention must be disposed of or modified." Duke Energy Corp. (McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station, Units 1 and 2), CLI-02-28, 56 N.R.C. 373, 382 (2002) (footnote omitted). Where "a contention alleges the omission of particular information or an issue from an application, and the information is later supplied by the applicant or considered by the Staff in a draft EIS, the contention is moot." Id. at 383 (footnote omitted).

As discussed below, PIIC Contention 7 alleged the omission of particular information from the LRA. The allegedly omitted information was supplied by NSPM in its LRA supplement filed on May 12, 2009.

IV. PIIC CONTENTION 7 SHOULD BE DISMISSED AS MOOT

A. PIIC Contention 7 Alleged The Omission Of Information From The LRA

PIIC Contention 7, as admitted, raises a dispute that the LRA does not sufficiently consider and contain an adequate plan to monitor and manage embrittlement of the reactor vessel

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NSPM's motion to dismiss PIIC Contention 8 as moot, but refrained from formally dismissing PIIC Contention 8 for twenty days from the date of the order. Order (Conditionally Granting Motion to Dismiss PIIC Contention 8 as Moot) (May 11, 2009). The Board, however, was unaware that NSPM's consistent practice has been, as part of its consultation prior to filing a motion to dismiss, to provide the relevant LRA amendment to the PIIC ten days in advance of filing any such motion. NSPM respectfully submits that in light of this practice of providing the LRA amendment to the PIIC in advance of filing a motion to dismiss, there is no need for the Board to make its ruling conditional.

internals. LBP-08-26 at 42 & 44. PIIC Contention 7 is a contention of omission, as the Board specifically observed. Id. at 42 & 43.

B. NSPM's LRA Supplement Includes The Reactor Vessel Internals Aging Management Program Whose Omission Was The Basis For Contention 7

NSPM's LRA supplement provides a detailed description of the "PWR Vessel Internals Program" which "addresses the management of aging effects in the reactor vessel internals components," and which will be implemented prior to the period of extended operation. Exhibit A, Enclosure 1 at 5 & 6. The "program is based on the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227), as approved by the NRC, and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program."[2] Id. at 6. The LRA supplement explains that the Vessel Internals Program is a new, plant-specific program and consists of the ten elements of an acceptable Aging Management Program described in NUREG-1800.[3] Id.

Specifically, the program is focused on, and credited for, managing: "crack initiation and growth due to irradiation-assisted stress corrosion cracking (IASCC), primary water stress corrosion cracking (PWSCC), stress corrosion cracking (SCC) and fatigue; reduction of fracture toughness due to radiation and thermal embrittlement and void swelling; changes in dimensions due to void swelling; loss of preload due to stress relaxation; and loss of material due to wear, in

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[2] The LRA supplement further elaborates that the "program is based upon the examination requirements for Westinghouse designed PWRs provided in MRP-227, 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0),' December 2008, along with the implementation guidance described in NEI 03-08, 'Guideline for the Management of Materials Issues,' May 2003. MRP-227 has been submitted to the NRC for review. Following NRC review and approval, MRP-227 will be revised to incorporate any necessary changes to the guidelines and reissued as MRP-227-A. The PINGP PWR Vessel Internals Program will be revised, as necessary, to incorporate the final recommendations and requirements as published in MRP-227-A." Exhibit A, Enclosure 1 at 7.

[3] Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, NUREG-1800, Rev. 1 (Sept. 2005).

reactor vessel internals components." Id. at 7; see also id. at 9-10. The LRA supplement provides, "[l]oss of fracture toughness due to radiation and thermal embrittlement is of consequence only if cracks exist and the local applied stress intensity exceeds the reduced fracture toughness. Cracking, if it occurs, is expected to initiate at the surface and is detectable by the augmented inspections performed under this program." Id. at 7.

The aging management methodologies used under the program include visual examinations, surface examinations, volumetric examinations, and physical measurements. Id. at 10 & 11. To provide timely detection of aging effects, a combination of one-time, periodic, and conditional examinations will be performed. Id. at 12. The "examination methods, coverage, and schedule of the inspection and test techniques, prescribed by the ASME Code, Section XI and MRP-227 are intended to maintain structural integrity and ensure the detection and correction of aging effects before the loss of intended function of PWR vessel internals." Id. at 12. The "acceptance criteria, against which the need for corrective actions are evaluated, ensure that the component intended functions are maintained under all current licensing basis design conditions during the period of extended operation." Id. at 13. And, after evaluation against the acceptance criteria, repairs, replacements or evaluations are performed for any flaws that fail to meet the acceptance standards. Id.

The PINGP program, though a new program, incorporates the benefits of industry operating experience because the "EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, which forms the basis of the PINGP PWR Vessel Internals Program, is based upon industry operating experience, research data, and vendor evaluations…[and] Reactor internals failures, both domestic and international, have been considered in the development of MRP-227." Id. at 15. As the LRA supplement concludes, "[i]mplementation of the PWR Vessel

Internals Program provides reasonable assurance that aging effects will be managed such that structures, systems, and components within the scope of this program will continue to perform their intended function(s) during the period of extended operation." Id. at 16.

C. The LRA Supplement Has Rendered PIIC Contention 7 Moot

PIIC Contention 7, as admitted by the Board, alleged that an adequate aging management program for the reactor vessel internals was omitted from the LRA. See LBP-08-26 at 44. NSPM has rendered the contention of omission moot by including in the May 12, 2009 LRA supplement a thorough description of its PWR Vessel Internals Program. See AmerGen Energy Co., LLC (License Renewal for Oyster Creek Nuclear Generating Station), CLI-08-28, 68 N.R.C. ___, slip op. at 25 n.72 (Nov. 6, 2008); McGuire, CLI-02-28, 56 N.R.C. at 382-83. Because it has been rendered moot, this Board should dismiss PIIC Contention 7.

V. CONCLUSION

For the reasons stated above, the Board should grant NSPM's Motion to Dismiss PIIC Contention 7 as Moot.

CERTIFICATION

As required by 10 C.F.R. § 2.323(b), counsel for NSPM certifies that he has consulted with the other parties in a sincere effort to resolve the issues raised in this motion.

Respectfully Submitted,

/Signed electronically by David R. Lewis/

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Dated: May 22, 2009

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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(Prairie Island Nuclear Generating Plant,	)	ASLBP No. 08-871-01-LR
Units 1 and 2)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "Northern States Power Company's Motion to Dismiss PIIC Contention 7 as Moot," dated May 22, 2009, was provided to the Electronic Information Exchange for service on the individuals listed below, this 22nd day of May 2009.

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/Signed electronically by David R. Lewis/

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EXHIBIT A

Xcel Energy®

May 12, 2009

L-PI-09-044
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

Supplemental 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. This letter amends the LRA to provide supplemental information addressing certain issues that have been raised as contentions in this License Renewal proceeding.

Enclosure 1 contains revisions to LRA Sections A2.32 and B2.1.32 to incorporate detailed information regarding the new plant-specific PWR Vessel Internals Program. Conforming changes are also provided for LRA Sections 3.0 and B2.0.

Enclosure 2 provides an updated version of the License Renewal Commitments list contained in the LRA transmittal letter. This updated list reflects the License Renewal commitment changes made to date in NSPM correspondence.

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. Commitment No. 25 in the list of PINGP License Renewal Commitments is revised 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.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on May 12, 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
Revisions to LRA Section 3.0 and Appendices A and B Regarding the PWR Vessel Internals Program

LRA Sections A2.32 and B2.1.32, and selected information in LRA Sections 3.0 and B2.0, are hereby revised to provide updated information regarding the PINGP PWR Vessel Internals Program. The updated information designates the program as a new plant-specific program. The LRA changes are as follows:

In LRA Section 3.1.2.1.3, Reactor Internals System, Aging Effects Requiring Management, on Page 3.1-5, the following new bullet is added:

- Cracking - Fatigue

In LRA Table 3.1.1, Summary of Aging Management Evaluations in Chapter IV of NUREG-1801 for Reactor Vessel, Internals, and Reactor Coolant System, on Page 3.1-30, Item Number 3.1.1-63 is revised to appear as follows:

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1-63	Steel reactor vessel flange, stainless steel and nickel alloy reactor vessel internals exposed to reactor coolant (e.g., upper and lower internals assembly, CEA shroud assembly, core support barrel, upper grid assembly, core support shield assembly, lower grid assembly)	Loss of material due to Wear	Inservice Inspection (IWB, IWC, and IWD)	No	Consistent with NUREG-1801. This aging effect is managed by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program or the PWR Vessel Internals Program.

In LRA Section 3.1.2.2.6, Loss of Fracture Toughness due to Neutron Irradiation Embrittlement and Void Swelling, on Page 3.1-12, the existing discussion is revised in its entirety to read as follows:

> Loss of fracture toughness due to neutron irradiation embrittlement and void swelling could occur for stainless steel and nickel alloy reactor internals components exposed to reactor coolant and neutron flux. This aging effect is managed with the PWR Vessel Internals Program. The PWR Vessel Internals Program addresses the management of aging effects in the reactor vessel internals components. The PWR Vessel Internals Program is based on the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. The program implements the examination requirements, including inspection methods, frequencies, acceptance criteria and sample sizes established in the EPRI guidelines for Westinghouse designed PWRs, to augment the ASME Section XI Inservice Inspection requirements. This program assures the intended function of affected components will be maintained during the period of extended operation.

Enclosure 1
Revisions to LRA Section 3.0 and Appendices A and B Regarding the PWR Vessel Internals Program

In LRA Section 3.1.2.2.9, Loss of Preload due to Stress Relaxation, on Page 3.1-13, the existing discussion is revised in its entirety to read as follows:

> Loss of preload due to stress relaxation could occur for nickel alloy and stainless steel reactor internals components. This aging effect is managed with the PWR Vessel Internals Program. The PWR Vessel Internals Program addresses the management of aging effects in the reactor vessel internals components. The PWR Vessel Internals Program is based on the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. The program implements the examination requirements, including inspection methods, frequencies, acceptance criteria and sample sizes established in the EPRI guidelines for Westinghouse designed PWRs, to augment the ASME Section XI Inservice Inspection requirements. This program assures the intended function of affected components will be maintained during the period of extended operation.

In LRA Section 3.1.2.2.12, Cracking due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (IASCC), on Page 3.1-14, the existing discussion is revised in its entirety to read as follows:

> Cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking could occur for stainless steel reactor internals components. This aging effect is managed with a combination of the Water Chemistry Program and the PWR Vessel Internals Program. The Water Chemistry Program includes specifications for chemical species, sampling and analysis frequencies, and corrective actions for control of water chemistry. The program controls concentrations of known detrimental chemical species such as chlorides, fluorides, sulfates and dissolved oxygen below the levels known to cause degradation. The PWR Vessel Internals Program addresses the management of aging effects in the reactor vessel internals components. The PWR Vessel Internals Program is based on the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. The program implements the examination requirements, including inspection methods, frequencies, acceptance criteria and sample sizes established in the EPRI guidelines for Westinghouse designed PWRs, to augment the ASME Section XI Inservice Inspection requirements. These programs assure the intended function of affected components will be maintained during the period of extended operation.

In LRA Section 3.1.2.2.15, Changes in Dimensions due to Void Swelling, on Page 3.1-15, the existing discussion is revised in its entirety to read as follows:

> Changes in dimensions due to void swelling could occur for stainless steel and nickel alloy reactor internals components. This aging effect is managed by the PWR Vessel Internals Program. The PWR Vessel Internals Program addresses the management of aging effects in the reactor vessel internals components. The PWR

Vessel Internals Program is based on the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. The program implements the examination requirements, including inspection methods, frequencies, acceptance criteria and sample sizes established in the EPRI guidelines for Westinghouse designed PWRs, to augment the ASME Section XI Inservice Inspection requirements. This program assures the intended function of affected components will be maintained during the period of extended operation.

In LRA Section 3.1.2.2.17, Cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking, on Pages 3.1-16 and 3.1-17, the existing discussion is revised in its entirety to read as follows:

Cracking due to stress corrosion cracking, primary water stress corrosion cracking, and irradiation-assisted stress corrosion cracking could occur for stainless steel and nickel alloy reactor internals components. This aging effect is managed with a combination of the Water Chemistry Program and the PWR Vessel Internals Program. The Water Chemistry Program includes specifications for chemical species, sampling and analysis frequencies, and corrective actions for control of water chemistry. The program controls concentrations of known detrimental chemical species such as chlorides, fluorides, sulfates and dissolved oxygen below the levels known to cause degradation. The PWR Vessel Internals Program addresses the management of aging effects in the reactor vessel internals components. The PWR Vessel Internals Program is based on the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. The program implements the examination requirements, including inspection methods, frequencies, acceptance criteria and sample sizes established in the EPRI guidelines for Westinghouse designed PWRs, to augment the ASME Section XI Inservice Inspection requirements. These programs assure the intended function of affected components will be maintained during the period of extended operation.

In LRA Table 3.1.2-3 Reactor Vessel, Internals, and Reactor Coolant System - Reactor Internals System - Summary of Aging Management Evaluation, on Pages 3.1-72 through 3.1-93, for all existing component line items being managed by the PWR Vessel Internals Program, the Notes entries are changed to E.

Enclosure 1
Revisions to LRA Section 3.0 and Appendices A and B Regarding the PWR Vessel Internals Program

In LRA Table 3.1.2-3, Reactor Vessel, Internals, and Reactor Coolant System – Reactor Internals System - Summary of Aging Management Evaluation, the following new line items are added:

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Volume 2 Line Item	Table 1 Item	Notes
Baffle and Former Plates Fasteners	Pressure Boundary, Structural Support	Stainless Steel	Treated Water (Ext)	Cracking - Fatigue	PWR Vessel Internals Program			H
BMI Columns and Flux Thimble Guides	Structural Support	Stainless Steel	Treated Water (Ext)	Cracking - Fatigue	PWR Vessel Internals Program			H
Clevis Insert Bolts	Structural Support	Nickel Alloy	Treated Water (Ext)	Loss of Material - Wear	PWR Vessel Internals Program	IV.B2-34	3.1.1-63	E
Core Barrel and Core Barrel Flange	Pressure Boundary, Shielding, Structural Support	Stainless Steel	Treated Water (Ext)	Cracking - Fatigue	PWR Vessel Internals Program			H
Core Barrel and Core Barrel Flange	Pressure Boundary, Shielding, Structural Support	Stainless Steel	Treated Water (Ext)	Loss of Material - Wear	PWR Vessel Internals Program	IV.B2-34	3.1.1-63	E
Core Barrel Outlet Nozzles	Pressure Boundary	Stainless Steel	Treated Water (Ext)	Cracking - Fatigue	PWR Vessel Internals Program			H
Lower Core Plate	Pressure Boundary, Structural Support	Stainless Steel	Treated Water (Ext)	Cracking - Fatigue	PWR Vessel Internals Program			H
Lower Core Plate	Pressure Boundary, Structural Support	Stainless Steel	Treated Water (Ext)	Loss of Material - Wear	PWR Vessel Internals Program	IV.B2-34	3.1.1-63	E
Lower Support Column Bolts	Structural Support	Stainless Steel	Treated Water (Ext)	Cracking - Fatigue	PWR Vessel Internals Program			H
Rod Cluster Control Assemblies Guide Tubes	Structural Support	Stainless Steel	Treated Water (Ext)	Cracking - Fatigue	PWR Vessel Internals Program			H
Rod Cluster Control Assemblies Guide Tubes	Structural Support	Stainless Steel	Treated Water (Ext)	Loss of Material - Wear	PWR Vessel Internals Program	IV.B2-34	3.1.1-63	E
Thermal Shields	Shielding	Stainless Steel	Treated Water (Ext)	Cracking - Fatigue	PWR Vessel Internals Program			H

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Volume 2 Line Item	Table 1 Item	Notes
Thermal Shields	Shielding	Stainless Steel	Treated Water (Ext)	Loss of Material - Wear	PWR Vessel Internals Program	IV.B2-34	3.1.1-63	E
Upper Support Plate Assembly	Structural Support	Stainless Steel	Treated Water (Ext)	Cracking - Fatigue	PWR Vessel Internals Program			H

LRA Section A2.32, on Page A-14, is revised in its entirety to read as follows:

A2.32 PWR Vessel Internals Program

The PWR Vessel Internals Program addresses the management of aging effects in the reactor vessel internals components. The PWR Vessel Internals Program is based on the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, as approved by the NRC, and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. The program implements the inspection of the reactor vessel internals components through the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program, as augmented by the examination requirements, including inspection methods, frequencies, acceptance criteria and sample sizes, established in the EPRI guidelines for Westinghouse designed PWRs.

PINGP participates in the industry programs for investigating and managing aging effects in reactor internals. The program implements applicable results of the industry programs.

This program will be implemented prior to the period of extended operation.

In LRA Section B2.0, Aging Management Programs Correlation, on Page B-8, line item XI.M16 of the NUREG-1801 program correlation table is revised to appear as follows:

NUREG-1801 ID	NUREG-1801 Program	PINGP Program	NUREG-1801 Comparison
XI.M16	PWR Vessel Internals	PWR Vessel Internals Program [Section B2.1.32]	New Plant-Specific Program, See Note 2

In LRA Section B2.0 on Page B-12, Note 2 is revised in its entirety to read:

2. A commitment is provided in Appendix B of this application [Section B2.1.32] to submit for NRC staff review and approval an inspection plan for reactor internals at least twenty-four months prior to the period of extended operation.

LRA Section B2.1.32 on Page B-66 is revised in its entirety to read as follows:

B2.1.32 PWR Vessel Internals Program

Program Description

The PWR Vessel Internals Program addresses the management of aging effects in the reactor vessel internals components. The program is based on the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227), as approved by the NRC, and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. The program implements the inspection of the reactor vessel internals components through the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program, as augmented by the examination requirements, including inspection methods, frequencies, acceptance criteria and sample sizes, established in the EPRI guidelines for Westinghouse designed PWRs.

The EPRI inspection and evaluation guidelines establish the augmented ASME Section XI Inservice Inspection requirements that will be used to monitor for the aging effects and parameters that are applicable to certain susceptible or limiting reactor vessel internals components for Westinghouse designed PWRs.

The aging management approach for PWR vessel internals consists of four major elements: (1) component categorization and aging management strategy development; (2) selection of aging management methodologies for PWR vessel internals that are both appropriate and based on an adequate level of applicable experience; (3) qualification of the recommended methodologies that is based on adequate technical justification; and (4) implementation of the recommendations based on the Industry Initiative for the Management of Materials Issues, NEI 03-08.

PINGP participates in the industry programs for investigating and managing aging effects on reactor internals. The program implements applicable results of the industry programs.

NUREG-1800 Consistency

The Prairie Island Nuclear Generating Plant PWR Vessel Internals Program is a new plant-specific program. The program consists of the ten elements of an acceptable AMP as described in NUREG-1800 Appendix A.1, Section A.1.2.3 and Table A.1-1.

Exceptions to NUREG-1800 or NUREG-1801

None

Enhancements

None

Aging Management Program Elements

The elements which are part of the PWR Vessel Internals Program are described below. The results of an evaluation of each element against NUREG-1800, Appendix A.1, Section A.1.2.3, "Aging Management Program Elements" and Table A.1-1, "Elements of an Aging Management Program for License Renewal," are also provided.

Scope of Program

The PWR Vessel Internals Program addresses the management of aging effects of the PINGP reactor vessel internals components, both non-bolted and bolted. The PINGP reactor vessel internals consist of two basic assemblies, the upper internals assembly that is removed during each refueling operation to obtain access to the reactor core, and the lower internals assembly that can be removed, if desired, following a complete core off-load. The scope does not include consumable items such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation. The scope also does not include welded attachments to the reactor vessel.

The PWR Vessel Internals Program is focused on managing crack initiation and growth due to irradiation-assisted stress corrosion cracking (IASCC), primary water stress corrosion cracking (PWSCC), stress corrosion cracking (SCC) and fatigue; reduction of fracture toughness due to radiation and thermal embrittlement and void swelling; changes in dimensions due to void swelling; loss of preload due to stress relaxation; and loss of material due to wear, in reactor vessel internals components. Loss of fracture toughness due to radiation and thermal embrittlement is of consequence only if cracks exist and the local applied stress intensity exceeds the reduced fracture toughness. Cracking, if it occurs, is expected to initiate at the surface and is detectable by the augmented inspections performed under this program.

The program is based upon the examination requirements for Westinghouse designed PWRs provided in MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)," December 2008, along with the implementation guidance described in NEI 03-08, "Guideline for the Management of Materials Issues," May 2003. MRP-227 has been submitted to the NRC for review. Following NRC review and approval, MRP-227 will be revised to incorporate any necessary changes to the guidelines and reissued as MRP-227-A. The PINGP PWR Vessel Internals Program will be revised, as necessary, to incorporate the final recommendations and requirements as published in MRP-227-A.

The program will include a plant-specific inspection plan for vessel internals based on the guidance of MRP-227 (MRP-227-A when issued). This plan will include the requirements established by the MRP for Westinghouse designed PWRs, as approved or modified by the NRC, and will define any proposed alternatives determined to be necessary. NSPM will submit the plant-specific inspection plan, including any proposed alternatives, for NRC review and approval at least 24 months prior to entry into the period of extended operation. NSPM will implement the PWR Vessel Internals Program, including the inspection plan, prior to the period of extended operation. [Note: There is a potential for program implementation to occur sooner in accordance with an industry commitment to implement the Inspection and Evaluation Guidelines within twenty-four months following issuance of the NRC-approved MRP-227-A.]

The development of the inspection and evaluation (I&E) guidelines of MRP-227 was organized around a framework and strategy for managing the effects of aging in PWR internals, together with a substantial database of material data and supporting results. Based upon the framework and strategy, and on the accumulated data, three important precursor elements to these I&E guidelines were then developed: screening criteria, categorization of PWR vessel internals, and functionality assessment of components and assemblies of components. The aging management strategy development combined the results of the functionality assessment with component accessibility, operating experience, existing evaluations, and prior examination results to determine the appropriate methodologies for maintaining the long-term functions of PWR vessel internals safely and economically. This process permitted further categorization of PWR vessel internals into functional groups. The ultimate result of the process was to assign the components into Primary, Expansion, Existing Programs, and No Additional Measures groups, with appropriate recommendations for aging management. The groups are described below.

- **Primary:** Those PWR vessel internals that are highly susceptible to the effects of at least one of the aging mechanisms were placed in the Primary group. The aging management requirements that are needed to ensure functionality of Primary components are described in MRP-227. The Primary group also includes components which have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.

- **Expansion:** Those PWR vessel internals that are highly or moderately susceptible to the effects of at least one of the aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the Expansion group. The schedule for implementation of aging management requirements for Expansion components will depend on the findings from the examinations of the Primary components at PINGP.

- **Existing Programs:** Those PWR vessel internals that are susceptible to the effects of at least one of the aging mechanisms, and for which generic and plant-specific existing aging management program requirements are capable of managing those aging effects, were placed in the Existing Programs group.

- **No Additional Measures:** Those PWR vessel internals for which the effects of all aging mechanisms are below the screening criteria were placed in the No Additional Measures group. No further action is required by MRP-227 for managing the aging of the No Additional Measures components.

The categorization process described above does not supersede the ASME Section XI Inservice Inspection requirements for reactor vessel internals components.

This AMP consists of PINGP activities that manage aging effects for components of the following systems and/or structures:

- Reactor Internals (RX) System

This element is consistent with the recommendations of NUREG-1800, Element A.1.2.3.1, Scope of Program.

Preventive Actions

The PWR Vessel Internals Program is a condition monitoring program and does not include any preventive or mitigative actions. Preventive and mitigative actions for the reactor vessel internals components are established and implemented in accordance with the PINGP Water Chemistry Program. The Water Chemistry Program manages aging effects by controlling concentrations of known detrimental chemical species, such as chlorides, fluorides, sulfates and dissolved oxygen, below the levels known to cause degradation. The program includes specifications for chemical species, sampling and analysis frequencies, and corrective actions for control of water chemistry. This program conforms to the EPRI "PWR Primary Water Chemistry Guidelines." The PINGP Water Chemistry Program is further described in LRA Section B2.1.40, Water Chemistry Program.

This element is consistent with the recommendations of NUREG-1800, Element A.1.2.3.2, Preventive Actions.

Parameters Monitored/Inspected

The PINGP PWR Vessel Internals Program monitors the effects of aging degradation mechanisms on the intended function of reactor vessel internals components through one-time, periodic, and conditional examinations, and other aging management program methodologies, as needed, in accordance with the ASME Code, Section XI and MRP-227. The program is credited for managing cracking due to IASCC, PWSCC, SCC, and fatigue; reduction of fracture toughness due to radiation and thermal embrittlement and void swelling; changes in dimensions due to void swelling; loss of preload due to stress relaxation; and loss of material due to wear in reactor internals components. The program contains elements that monitor and inspect for the parameters that govern the progress of each of these aging effects. Section 4 of MRP-227 describes the methodologies that provide the monitoring and inspection of these aging effects.

The aging management methodologies include visual examinations, surface examinations, volumetric examinations, and physical measurements. The visual (VT-3) examinations detect the general degradation conditions, whereas the visual (VT-1) and enhanced visual (EVT-1) examinations are conducted to detect discontinuities and imperfections on the surface of components. The surface examinations further characterize discontinuities on the surface of components, and the volumetric inspections indicate the presence of discontinuities or flaws throughout the volume of material. Some aging effects may involve changes in clearances, settings, and physical displacements that can be monitored by visual means, supplemented by physical measurements.

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program requires a visual VT-3 examination of the reactor vessel removable core support structures in accordance with Table IWB-2500-1, Examination Category B-N-3, once per Inservice Inspection interval.

The extent and schedule of the inspection and test techniques prescribed by the ASME Code, Section XI and augmented by MRP-227 are designed to maintain structural integrity and ensure that aging effects will be detected and corrective actions taken, before the loss of intended function.

This element is consistent with the recommendations of NUREG-1800, Element A.1.2.3.3, Parameters Monitored or Inspected.

Detection of Aging Effects

Inspection and evaluation to manage aging of reactor vessel internals consists of the following:

- selection of items for aging management;
- selection of the type of examination or other methodologies appropriate for each applicable degradation mechanism;
- specification of the required level of examination qualification;
- schedule of first examination and frequency of any subsequent examinations;
- sampling and coverage;
- expansion of scope if sufficient evidence of degradation is observed;
- examination acceptance criteria;

- methods for evaluating examination results not meeting the examination acceptance criteria;
- updating the program based on industry-wide results; and
- contingency measures to repair, replace, or mitigate.

MRP-227 describes the examination requirements for the PWR vessel internals Primary and Expansion components for Westinghouse plants. The aging management methodologies described in MRP-227 are based on either existing inservice examinations required by the ASME Code, Section XI or on well-documented and well-demonstrated examination methods with which the industry has considerable experience.

The aging management methodologies include visual examinations, surface examinations, volumetric examinations, and physical measurements, described in more detail below.

- Visual (VT-3) examinations are conducted to determine the general mechanical and structural condition of components by detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion; and by identifying conditions that could affect operational or functional adequacy of components. VT-3 examinations of internals are conducted using remote examination techniques, due to personnel radiation exposure issues.

- The visual (VT-1) examination and the enhanced visual (EVT-1) examination are utilized where a greater degree of detection capability than visual (VT-3) examination is needed to manage the aging effect. Unlike the detection of general degradation conditions by visual (VT-3) examination, visual (VT-1) and enhanced visual (EVT-1) examinations are conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion.

- Surface examination may be used to supplement either visual (VT-3) or (VT-1/EVT-1) examinations, in order to further characterize discontinuities on the surface of components. This supplemental examination may thus be used to reject or accept relevant indications. A surface examination is an examination that indicates the presence of surface discontinuities. Surface examinations are conducted using the eddy current (ET) inspection method.

- An ultrasonic examination (UT) is selected where visual or surface examination is unable to detect the effect of the age-related degradation for some PWR vessel internals. The UT inspections indicate the presence of discontinuities or flaws throughout the volume of material.

- Physical measurements, in some cases, can manage the effects of loss of material due to wear, the loss of preload or clamping force caused by thermal and irradiation-enhanced stress relaxation, and excessive distortion or deflection caused by void swelling. These aging effects may involve changes in clearances, settings, and physical displacements that can be monitored by visual means, supplemented by physical measurements that characterize the magnitude of the effects. This methodology may be used in conjunction with visual (VT-3) examination, which includes "verifying parameters, such as clearances, settings, and physical displacements."

Those components designated as having no significant aging effects will require no additional measures for future inspections other than the ASME Code inspections per Section XI, Examination Category B-N-3, for removable internal structures. The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program conducts a visual VT-3 examination of the reactor vessel removable core support structures under Table IWB-2500-1, Examination Category B-N-3, once per Inservice Inspection interval. The inspections are performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR Part 50, Appendix B.

The examination methods, coverage, and schedule of the inspection and test techniques, prescribed by the ASME Code, Section XI and MRP-227 are intended to maintain structural integrity and ensure the detection and correction of aging effects before the loss of intended function of PWR vessel internals. The examination requirements defined in MRP-227, as approved by the NRC, will be applied through use of MRP-228, "Inspection Standard for Reactor Internals," when issued.

PINGP participates in the industry programs for investigating and managing aging effects on reactor internals. The program implements applicable results of the industry programs.

This element is consistent with the recommendations of NUREG-1800, Element A.1.2.3.4, Detection of Aging Effects.

Monitoring and Trending

One-time, periodic, and conditional examinations and other aging management methodologies, scheduled in accordance with the ASME Code, Section XI and MRP-227 provide timely detection of aging effects. In addition to the Primary components, Expansion components have been defined should the scope of examination and re-examination need to be expanded beyond the Primary group due to detection of significant aging effects. Flaw indications detected during the required examinations are dispositioned in accordance with the Acceptance Criteria and Corrective Actions program elements discussed below.

This element is consistent with the recommendations of NUREG-1800, Element A.1.2.3.5, Monitoring and Trending.

Acceptance Criteria

For ASME Code Section XI inspections, indications and relevant conditions detected during examination are required to be evaluated in accordance with ASME Section XI, Article IWB-3500. In addition to the ASME Code Section XI requirements, MRP-227 provides the examination acceptance criteria for the Primary and Expansion components. Also, the expansion criteria for expanding the examinations beyond the Primary components to include the Expansion components are provided. The examination acceptance criteria include: (i) specific, descriptive relevant conditions for the visual (VT-3) examinations; (ii) requirements for recording and dispositioning surface breaking indications that are detected and sized for length by the visual (VT-1/EVT-1) examinations; and (iii) requirements for system-level assessment of bolted or pinned assemblies with volumetric (UT) examination indications that exceed specified limits. Any detected condition that does not satisfy these examination acceptance criteria must be dispositioned.

The acceptance criteria, against which the need for corrective actions are evaluated, ensure that the component intended functions are maintained under all current licensing basis design conditions during the period of extended operation.

This element is consistent with the recommendations of NUREG-1800, Element A.1.2.3.6, Acceptance Criteria.

Corrective Actions

Indications are evaluated per the acceptance criteria, which determine relevant flaw indications that are unacceptable for further service. Relevant flaw indications are corrected through implementation of appropriate repair/replacement activities.

Identified flaws are entered into the PINGP Corrective Action Program for appropriate disposition. A repair, replacement, or evaluation is performed for all flaws that exceed the acceptance standards. Additional guidance for disposition of unacceptable conditions for reactor vessel internals may be found in the ASME Code, Section XI; in MRP-227 Guidelines; and in reports referenced therein or demonstrated through an appropriate technical justification. MRP-227 provides information on methodology that can be used for the evaluation of detected conditions that exceed the examination acceptance criteria. The flaw evaluation methodology accounts for the accumulated neutron exposure and the resulting loss of fracture toughness due to radiation embrittlement in assessing the suitability of the component for continued service. Justification for flaw evaluation fracture toughness limits is provided in Section 6 of MRP-227.

Repair/replacement activities comply with ASME Section XI as invoked by 10 CFR 50.55a or approved ASME Code Cases as referenced in the latest version of NRC Regulatory Guide 1.147. Proposed alternative repair/replacement activities, if any, will be submitted to the NRC for review and approval in accordance with 10 CFR 50.55a(a)(3)(i) or 10 CFR 50.55a(a)(3)(ii).

See LRA Section B1.3 for further discussion of this element.

This element is consistent with the recommendations of NUREG-1800, Element A.1.2.3.7, Corrective Actions.

Confirmation Process

Corrective action effectiveness is part of the PINGP Corrective Action Program.

See LRA Section B1.3 for further discussion of this element.

This element is consistent with the recommendations of NUREG-1800, Element A.1.2.3.8, Confirmation Process.

Administrative Controls

See LRA Section B1.3 for the discussion of this element.

This element is consistent with the recommendations of NUREG-1800, Element A.1.2.3.9, Administrative Controls.

Operating Experience

Relatively few incidents of PWR vessel internals aging degradation have been reported in operating U.S. commercial PWR plants. However, a considerable amount of PWR vessel internals aging degradation has been observed in European PWRs, with emphasis on cracking of baffle-former bolting. For this reason, the U.S. PWR owners and operators began a program a decade ago to inspect the baffle-former bolting in order to determine whether similar problems might be expected in U.S. plants. A benefit of this decision was the experience gained with the UT examination techniques used in the inspections. In addition, the industry began substantial laboratory testing projects in order to gather the materials data necessary to support future inspections and evaluations.

PINGP baffle-former bolts are fabricated from AISI Type 347 stainless steel. This material is considered to have some limited susceptibility to cracking, due to inspection results obtained at another U.S. nuclear plant where it was concluded that 3.4% of that plant's bolts were non-functional. Out of 728 bolts inspected with UT, 55 indications were noted, but 30 of the bolts with indications performed acceptably in mechanical testing. An evaluation reported in WCAP-15036, “Determination of Acceptable Baffle-Barrel Bolting for Two-Loop Westinghouse

Domestic Plants", concluded that approximately 40-55% of the total original baffle-former bolts are required to be intact to meet the acceptance criteria for the limiting LOCA event.

Several other items with existing or suspected material degradation concerns that have been identified for PWR components are wear in thimble tubes and potentially in control guide cards, and observed cracking in some high-strength bolting and in control rod guide tube alignment (split) pins. The latter conditions have been corrected primarily through bolt replacement with less susceptible material and improved control of preload. The reactor vessel upper internals were replaced at both PINGP Units in 1986-1987. The new internals included design and material improvements to significantly reduce the susceptibility to cracking of the split pins.

The PWR Vessel Internals Program is a new program to be implemented prior to the period of extended operation and, therefore has no operating experience related to program implementation. A review of plant specific operating experience related to the PINGP reactor vessel internals identified no significant aging management issues. The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program, Table IWB-2500-1, Examination Category B-N-3 inspections of the PINGP vessel internals, have not identified any indications.

The EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, which forms the basis of the PINGP PWR Vessel Internals Program, is based upon industry operating experience, research data, and vendor evaluations. Development of the program relied upon the consensus review and inputs of the MRP Reactor Internals Core and Focus Groups, which include representatives from utilities, research scientists, and vendors. This program will continue to evolve as additional experience is gained. Reactor internals failures, both domestic and international, have been considered in the development of MRP-227.

PINGP participates in the industry programs for investigating and managing aging effects on reactor vessel internals. Through its participation in EPRI MRP activities, PINGP will continue to benefit from the reporting of reactor vessel internals inspection information, and will share its own internals inspection results with the industry, as appropriate. The PINGP PWR Vessel Internals Program implements applicable results of the industry programs.

This element is consistent with the recommendations of NUREG-1800, Element A.1.2.3.10, Operating Experience.

Conclusion

The PWR Vessel Internals Program is a new program that manages cracking due to irradiation-assisted stress corrosion cracking, primary water stress corrosion cracking, stress corrosion cracking and fatigue; reduction of fracture toughness due to radiation and thermal embrittlement and void swelling; changes in dimensions due to void swelling; loss of preload due to stress relaxation; and loss of material due to wear in reactor vessel internals components. The PWR Vessel Internals Program implements the inspection of the PINGP reactor vessel internals through the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program as augmented by the examination requirements of MRP-227.

Implementation of the PWR Vessel Internals Program provides reasonable assurance that aging effects will be managed such that structures, systems, and components within the scope of this program will continue to perform their intended function(s) during the period of extended operation. This program will be implemented prior to the period of extended operation.

In addition to the LRA changes, Commitment No. 25 in the list of PINGP License Renewal Commitments has been revised in its entirety to read as follows:

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
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. [Revised in letter dated 5/12/2009]	A. U1 - 8/9/2013 U2 - 10/29/2014 B. U1 - 8/9/2011 U2 - 10/29/2012	B2.1.32

Enclosure 2

Updated License Renewal Commitment List

15 Pages

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 U2 - 10/29/2014	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.		
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: • The program will be expanded to include eight additional	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.16

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	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
15	The Fuel Oil Chemistry Program will be enhanced as follows: • Particulate contamination testing of fuel oil in the eleven	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.19

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	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.		
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: • Program implementing procedures will be revised to ensure the components and structures subject to inspection are clearly identified.	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.23

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	• 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. [Revised in letter dated 5/12/2009]	A. U1 - 8/9/2013 U2 - 10/29/2014 B. U1 - 8/9/2011 U2 - 10/29/2012	B2.1.32
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	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.33

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	material and inspection guidance provided in NRC Regulatory Guide 1.65.		
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 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	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.35

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	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 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,	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.38

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	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. • 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

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

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	• 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 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.	April 30, 2009 Commitment closed by letter dated 4/28/09	4.3.3

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	[Added in letter dated 1/9/2009 in response to RAI 4.3.1.1-1]		
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. 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]	8/9/2013	ER 4.16.2
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.	8/9/2013	ER 4.16.2

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	[Added in ER revision submitted in letter dated 3/4/2009]		
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 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]	U1 - 8/9/2013 U2 - 10/29/2014	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	U1 - 8/9/2013 U2 - 10/29/2014	B2.1.38

Commitment Number	Commitment	Implementation Schedule	Related LRA Section Number
	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]		
43	Preventive maintenance requirements will be implemented to require periodic replacement of rubber flexible hoses in the Diesel Generators and Support System 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]	U1 - 8/9/2013 U2 - 10/29/2014	Table 3.3.2-8