REVISED 09/03/10

July 30, 2010

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)	)	

NSPM'S INITIAL STATEMENT OF POSITION ON SAFETY CULTURE CONTENTION

Pursuant to 10 C.F.R. § 2.1207(a)(1) and the Atomic Safety and Licensing Board's ("Board") Memorandum and Order (Summarizing Prehearing Conference Call and Amending Hearing Schedule) (April 20, 2010) Northern States Power Company, a Minnesota corporation ("NSPM") hereby submits its Initial Statement of Position ("Statement") on the Prairie Island Indian Community's ("PIIC") Safety Culture Contention ("Safety Culture Contention"). This Statement is supported by the "Testimony of Steven C. Skoyen on Safety Culture Contention" ("Skoyen Dir.") and exhibits thereto; and the "Joint Testimony of Scott Northard, Kurt W. Petersen and Ed M. Peterson II on Safety Culture Contention" ("Northard Dir.") and exhibits thereto.

I. INTRODUCTION

By application dated April 11, 2008 and supplemented May 16, 2008, Nuclear Management Company, LLC requested renewal of Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2 ("PINGP") (the

"Application").[1] On June 17, 2008, the Nuclear Regulatory Commission ("NRC" or "Commission") published a Notice of Opportunity for Hearing ("Notice") regarding the Application. 73 Fed. Reg. 34,335 (June 17, 2008). The Notice permitted any person whose interest may be affected to file a request for hearing and petition for leave to intervene within 60 days of the Notice. Id.

On August 18, 2008, the PIIC petitioned to intervene and submitted eleven contentions. On December 5, 2008, the Board granted PIIC's petition and admitted seven of the contentions. Northern States Power Co. (Prairie Island Nuclear Generating Plant, Units 1 and 2), LBP-08-26, 68 N.R.C. 905 (2008). All of these contentions have since been resolved, either by settlement between the parties or by rulings by the Board granting NSPM's motions to dismiss.

Thereafter, the Board issued a Scheduling Order reflecting the agreement of the parties that any new or amended contentions on new data or conclusions in the draft Safety Evaluation Report ("SER") issued by the NRC Staff ("Staff") could be filed within 30 days after issuance of the document. Licensing Board Memorandum and Order (Prehearing Conference Call Summary and Initial Scheduling Order) (Feb. 18, 2009) at 4.

On June 4, 2009, the Staff issued the Safety Evaluation Report with Open Items Related to the License Renewal of Prairie Island Nuclear Generating Plant Units 1 and 2 ("SER with Open Items") (ADAMS Accession No. ML091550014). The SER resolved all safety issues relevant to the Application with the exception of three open items identified in the report. Id. at iii, 1-7 to 1-9. One of the open items related to assessing and managing any effects of leakage that has occurred in the refueling cavity of each unit. Id. at 1-8 to 1-9, 3-142 to 3-143.

[1] On September 22, 2008, the NRC amended the operating licenses for the PINGP, Units 1 and 2, to transfer the operating authority from Nuclear Management Company, LLC to NSPM. As a result, NSPM is now the sole licensee for these units (licensed to both own and operate the units), and the applicant for renewal of the PINGP operating licenses. Letter from D. Lewis to ASLB (Sept. 29, 2008).

The Advisory Committee on Reactor Safeguards ("ACRS"), Subcommittee on License Renewal, met on July 7, 2009, to review NSPM's application, the Staff's SER with Open Items, and associated documents. Transcript, ACRS Subcommittee on License Renewal for the Prairie Island Generating Station (July 7, 2009) ("ACRS Tr.") (ADAMS Accession No. ML092180127). The subcommittee discussed extensively the open item related to the refueling cavity leakage, the actions that NSPM had previously taken to manage this leakage, NSPM's root cause evaluation, the results of related inspections showing no degradation, NSPM's plans for permanent repairs, the results of a conservative evaluation to bound the effect of any degradation that might have occurred, and the commitments that NSPM had made in response to NRC follow-up requests for additional information ("RAIs"). ACRS Tr. at 47-81.

On October 16, 2009, the Staff issued the final Safety Evaluation Report Related to the License Renewal of Prairie Island Nuclear Generating Plant Units 1 and 2 ("Final SER") (ADAMS Accession No. ML092890209). The Final SER summarized additional information on the refueling cavity leakage issue that had been provided by NSPM on June 24, 2009 (ADAMS Accession No. ML091800018) and August 7, 2009 (ADAMS Accession No. ML092360408) and NSPM's responses to the follow-up RAIs, and closed the refueling cavity leakage open item on the basis of this information. Final SER at 1-8 to 1-9, 3-142 to 3-149.

On November 23, 2009, the PIIC submitted a new contention that cited the Final SER. The PIIC's proposed contention alleged:

> Contrary to the conclusion in the Safety Evaluation Report (SER), the Community does not believe that 'the requirements of 10 CFR 54.29(a) have been met.' Due to recent significant non-compliances with NRC regulations, as well as the applicant's failure to address a known potentially serious safety problem identified in the SER, the Community does not believe that there is any justification for a reasonable assurance determination by the NRC that the applicant will '. . . manag[e] the effects of aging during the period of extended operation on the functionality of structure and components' as required by 10 CFR 54.29(a)(1).

Prairie Island Indian Community's Submission of a New Contention on the NRC's Safety Evaluation Report (Nov. 23, 2009) at 4. NSPM and the NRC Staff each filed answers opposing the PIIC contention on several substantive and procedural grounds.

On January 28, 2010, the Board issued an Order admitting the following reformulated contention:

> PINGP's safety culture is not adequate to provide the reasonable assurance required by 10 C.F.R § 54.29(a)(1) that PINGP can manage the effects of aging during the requested period of extended operation.

Order (Narrowing and Admitting PIIC's Safety Culture Contention) (Jan. 28, 2010) at 14. Both NSPM and the Staff have filed petitions seeking interlocutory Commission review of the Board's decision to admit the Safety Culture Contention. The Commission has not yet ruled on those petitions.

The facts, testimony and evidence relating to the Safety Culture Contention are described below.

II. APPLICABLE LEGAL STANDARDS

10 C.F.R. § 54.29(a) authorizes issuance of a renewed licensing if the Commission finds that actions have been identified and have been or will be taken with respect to managing the effects of aging and time-limited aging analyses, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis ("CLB"). In admitting the Safety Culture contention for litigation, the Board explained that the contention is based on this provision. Id. As the Board further explained, "PIIC contends that a weak safety culture exists at PINGP and that, as a result, such 'reasonable assurance' cannot be found. Id.

As the Commission has recently explained,

> In issuing the license renewal regulations, the Commission recognized that "not all reactors are in full compliance with their respective CLBs on a continuous basis" and that "[t]he NRC conducts its inspection and enforcement activities under the presumption that non-compliances will occur." But "all aspects of a plant's CLB . . . and the NRC's regulatory

> process," including inspection and oversight activities, "carry forward into the renewal period" to maintain the CLB. "[L]imits on the scope of [the] renewal review and hearing" were based "on careful review of the sufficiency of the NRC regulatory process to resolve issues not considered in renewal." The regulatory process continuously reassesses whether there is a need for additional oversight or regulations to protect public health and safety.

Entergy Nuclear Generation Co. (Pilgrim Nuclear Power Station), CLI-10-14, __ N.R.C. __, slip op. at 17-18 (June 17, 2010) (footnotes omitted).

Thus, license renewal is not based on a standard of perfection or error-free performance (though that is what every licensee continually strives for). Rather, the standard for this demonstration is one of "reasonable assurance." "To issue a renewed license, the NRC must find 'reasonable assurance' that the licensee will manage the effects of aging on the functionality of SSCs identified to require an aging management review." Id. at 20. See also Nuclear Power Plant License Renewal Final Rule, 60 Fed. Reg. 22,461, 22,479 (1995) (". . . the [license renewal] process is not intended to demonstrate absolute assurance that structures or components will not fail, but rather that there is reasonable assurance that they will perform such that the intended functions . . . are maintained consistent with the CLB").

Reasonable assurance is not a standard that is quantified to any particular confidence level, but is based on sound technical judgment of the particulars of a case, including compliance with the applicable regulations. Thus, "reasonable assurance" is not reduced to a mechanical formula or set of objective standards, but may be "given content through case-by-case applications of [the Commission's] technical judgment" in light of all relevant information. Pilgrim, CLI-10-14, slip op. at 21 (footnotes omitted).

III. APPLICANTS' STATEMENT OF POSITION ON FACTUAL ISSUES

A. NSPM's Witnesses on Safety Culture Contention

NSPM's testimony on the Safety Culture Contention will be presented by an individual and a panel of three witnesses. All the NSPM witnesses have extensive experience and expertise in the respective areas addressed in their testimony. They are as follows:

- Mr. Steven C. Skoyen, who will testify on the refueling cavity leakage aspect of the Safety Culture Contention, is employed by NSPM as Engineering Programs Manager for PINGP. He has approximately twenty years of professional experience in the identification of, and response to, equipment problems in nuclear power plants. He is currently responsible for the technical oversight, strategic planning and improvement of ASME, NRC and INPO required programs with respect to plant equipment at PINGP. He is also responsible for the identification and scheduling of component maintenance, testing and inspection activities and for the coordination of the engineering response to equipment problems. He has performed similar duties in his previous employment with, among others, Westinghouse Electric Corporation.

  Mr. Skoyen has been involved with the PINGP refueling cavity leakage issue, with direct responsibility for its resolution, since December 2006. He remains involved in overseeing the causal investigations and the corrective actions taken towards eliminating the leakage.

- Messrs. Scott D. Northard, Kurt W. Petersen and Ed M. Peterson II will testify as a panel on all other aspects of the Safety Culture Contention.

  - Mr. Northard is employed by NSPM as Recovery Manager for PINGP. He has more than thirty years of experience in the nuclear power plant industry, including positions as Plant Manager, Regulatory Affairs Manager, Nuclear Safety Assurance Manager, Business Support

Manager, Site Engineering Director, Director Asset Management, and Manager Nuclear Projects. His areas of concentration have included the development and implementation of plans for improving nuclear power plant safety culture and operational performance.

Mr. Northard became involved with safety culture issues at PINGP when he was named head of the Performance Recovery Plan in March 2009. He has remained the plant official principally responsible for safety culture improvement initiatives since that time.

- Mr. Petersen, also employed by NSPM, is the Business Support Manager responsible for the corrective action program at PINGP. He has substantial experience in the management of corrective action programs at commercial nuclear power stations. This experience includes implementation of 10 CFR 50 Appendix B compliance programs, Human Performance Improvement Programs, and related plant performance assessment and improvement programs, having performed management level work in these areas at the Turkey Point Nuclear Generating Units 3 and 4 and at PINGP. He has held various other positions in the nuclear industry, including Lead Production Supervisor and several positions of increasing responsibility in the maintenance area (Work Week Manager, Maintenance Supervisor, Operations Command Center Maintenance Manager, and head of Maintenance Training).

  Mr. Petersen was the Performance Assessment Supervisor responsible for the administration of the Corrective Action Program (CAP) at PINGP from May 2006 until August 2009. Since that time, he has remained responsible for the management oversight of the CAP at PINGP.

- Mr. Peterson is employed as Ombudsman by the Wolf Creek Nuclear Operating Company. He has have thirty-three years of experience in quality assurance ("QA") related oversight of both the construction and operation phases of nuclear power plants. This experience includes 26 years of work for Wolf Creek Nuclear Operating Company as Ombudsman, Quality Administrator, Operations QA Supervisor, Operations QA Auditor, and Quality Control Supervisor at the Wolf Creek Generating Station. He was previously employed by Daniel International Corporation as Senior Quality Engineer Supervisor at Wolf Creek; by Bechtel Power Corporation as Quality Control Engineer at the South Texas Project; by Brown and Root Inc. as Mechanical Quality Engineer – Documents Supervisor, also at the South Texas Project; and by Daniel International Corporation as Mechanical and Civil Quality Control Inspector at the Wolf Creek Generating Station.

Mr. Peterson has been actively involved in thirteen safety culture assessments since October 2007. Of those, he was site host for two assessments at Wolf Creek, participated as a team member on five assessments, and was team leader on six assessments. In his current capacity as Alternate Team Leader for the Utilities Service Alliance, Nuclear Safety Culture Assessment ("NSCA") II Project Team, Mr. Peterson oversees safety culture assessments and supports the project team with coordination, maintenance, and improvements to the NSCA process. He has been an active participant in two of the three completed NEI 09-07 pilot assessments on further development of the NSCA procedures, serving as team lead on one of such assessments.

Mr. Peterson led a nuclear safety culture assessment conducted at PINGP on June 21-25, 2010 under the auspices of, and in accordance with, the process established by the Utilities Service Alliance ("USA") (a consortium of nuclear power generating stations). The NSCA was performed by a team of independent industry experts and PINGP personnel.

B. Introduction to PINGP's Safety Culture Programs

1. Definitions

1. Safety Culture is defined by the NRC, in a proposed safety culture policy statement ("Draft Safety Culture Policy Statement: Request for Public Comments"), 74 Fed. Reg. 57,525, 57,526 (November 6, 2009) as "that assembly of characteristics, attitudes, and behaviors in organizations and individuals, which establishes that as an overriding priority, nuclear safety and security issues receive the attention warranted by their significance." The Institute of Nuclear Power Operations ("INPO") has a similar definition of safety culture: "An organization's values and behaviors – modeled by its leaders and internalized by its members – that serve to make nuclear safety an overriding priority." Northard Dir. at A13.

2. In addition to the NRC's draft policy statement, NRC Inspection Manual Chapter ("IMC") 0305, which governs the Reactor Oversight Process, identifies a number of safety culture components:

Problem Identification & Resolution (PI&R)
P1. Corrective Action Program
P2. Operating experience
P3. Self- and Independent Assessments

Human Performance
H1. Decision-Making
H2. Resources
H3. Work Control
H4. Work Practices

Safety Conscious Work Environment
S1. Environment for Raising Concerns
S2. Preventing, Detecting, and Mitigating Perceptions of Retaliation

Other Safety Culture Components
D1. Accountability
D2. Continuous learning environment
D3. Organizational change management
D4. Safety policies.

Northard Dir. at A14 and Northard Exhibit 20 (NSP000038).

3. INPO also has developed a set of standards on behalf of the nuclear industry called the "Principles for a Strong Nuclear Safety Culture" ("INPO Principles") which are used throughout the industry to perform independent assessments of safety culture at operating reactors in the United States. Although worded somewhat differently, the INPO Principles have a close correlation with the Safety Culture Components defined by the NRC. The INPO Principles are:

> Principle 1. Everyone is personally responsible for nuclear safety.
>
> Principle 2. Leaders demonstrate commitment to safety.
>
> Principle 3. Trust permeates the organization.
>
> Principle 4. Decision-making reflects safety first.
>
> Principle 5. Nuclear technology is recognized as special and unique.
>
> Principle 6. A questioning attitude is cultivated.
>
> Principle 7. Organizational learning is embraced.
>
> Principle 8. Nuclear safety undergoes constant examination.

Northard Dir. at A14 and Northard Exhibit 4 (NSP000022).

2. Safety Culture Programs at PINGP

4. Safety is a core value of PINGP, to which NSPM is absolutely committed. This fundamental commitment is reflected in the company's nuclear organization's statement of Vision, Mission and Values: "Vision: Work together to provide safe, reliable and cost effective nuclear energy for the communities we serve. Mission: Foster a learning environment that promotes safe operations, continually enhances operational performance, promotes accountability for strong financial stewardship and demonstrates leadership within the nuclear industry and the communities we serve. Values: Maintain a defense in depth strategy to protect employees, the public, and the environment from the inherent nuclear, radiological, environmental and industrials safety risks associated with operations. Be honest, ethical, and accountable, treating people with respect as we work toward our common goals." Northard Dir. at A16 and Northard Exhibit 5 (NSP000023).

5. In every one of these principles, safety is first. NSPM's commitment to safety is also reflected and demonstrated in its sustained performance. NSPM has 112 reactor years of safe reactor operating experience. Northard Dir. at A16.

6. NSPM instills its commitment to safety in its employees and at its nuclear plants through its policies, programs, and training at every level. At the highest level, Corporate Policy CP 0017 – Nuclear Safety Culture and Risk Management Principles identifies the essential attributes of a healthy nuclear safety culture with the goal of creating a framework for open discussion and continuing evolution of safety culture. This policy adopts and incorporates the Principles established by the INPO. Northard Dir. at A17 and Northard Exhibit 6 (NSP000024).

7. This same Corporate Policy establishes core risk management principles:

   Nothing is routine

   Take the time to challenge uncertainty

   Risk significant activities will be made visible

   Risk activities will be planned, challenged, and controlled

   No risk option – first choice

   Prioritization to minimize operational challenges. Id.

8. The Corporate Policy also sets forth NSPM's expectations for a safety conscious work environment ("SCWE"), which are that:

   Workers at NSPM have the responsibility to ensure that they promptly raise nuclear safety concerns

   NSPM has the obligation to provide : a work environment that encourages workers to raise concerns without a fear of retaliation; efficient methods and options for raising concerns; and appropriate safety conscious work environment information to workers

   NSPM will not tolerate acts of harassment, intimidation, retaliation, or discrimination toward workers that raise concerns.

   Id.

9. NSPM Officers and Site Vice Presidents are responsible for promoting, cultivating, and assessing the nuclear safety culture at their sites and within NSPM. NSPM nuclear employees are expected to follow the risk management principles, to demonstrate risk management behaviors, to promptly raise nuclear safety concerns, and to treat nuclear safety as their primary responsibility. Northard Dir. at A18.

10. There are numerous programs that NSPM utilizes to instill the commitment to safety. One such program, established by Corporate Directive 3.4, is the Picture of Xcellence. The Picture of Xcellence is a model for changing and sustaining workforce behaviors through a union of management structure, procedures, and process that result in continuous performance improvement. It uses integrated plans that drive workforce behaviors and structured meetings to monitor performance and provide practical feedback, recognizing that individual behaviors can have an impact on organizational success. Id. at A19; Northard Exhibit 7 (NSP000025).

11. The Picture of Xcellence is based on the premise that performance of any organization is the result of the behaviors exhibited by the individuals who make up the organization. Sustained good performance requires daily good behaviors. The Picture of Xcellence provides the structure to develop and sustain a work environment that provides repeatable behaviors resulting in excellence. This work environment requires establishing and maintaining the following four principles: (1) Select and retain the Right People in the Right Jobs; (2) Communicate and Enforce the Right Picture; (3) Verify effective implementation of the Right Processes; and (4) Provide the Right Management Coaching and ensure effective Employee Engagement. Id.

12. The Right Picture is an accurate perspective of excellence with regard to the standards applied to conduct NSPM's business. Having that perspective on the high standard of excellence gives an individual and the organization a benchmark from which to measure their own performance. Getting the Right Picture is achieved by:

Clearly and credibly communicating the right expectations

Modeling the right behaviors

Understanding and demonstrating the right performance

Understanding and aligning with the right vision, goals, strategy and plan
Demonstrating the right passion
Providing timely and effective performance feedback.
Northard Dir. at A20 and Northard Exhibit 7 (NSP000025).

13. The Picture of Xcellence is organized by pillars: Nuclear Xcellence, Organizational Xcellence, Training Xcellence, and Equipment Xcellence. Each pillar is described in terms of attributes which characterize the pillar and the behaviors which define those attributes. In addition, each pillar contains objective performance measures, developed based upon benchmarking nuclear industry leaders, which are used to measure progress or the need for improvement. Northard Dir. at A21 and Northard Exhibit 7 (NSP000025).

14. The Picture of Xcellence program includes a number of "forcing functions" to apply the Picture of Xcellence to work activities, making it the responsibility of every employee to "coach and engage" other employees toward improving site performance. For example, meetings are conducted at the fleet, site, and individual level to ensure alignment with the Picture of Xcellence, to reinforce expectations, and to ensure appropriate resource allocations. One such type of meeting is the daily "D-15" meetings that are held between the members of each department and their front line supervisor to review identified focus areas, discuss the results of the Picture of Xcellence, and reinforce appropriate behaviors. Northard Dir. at A23 and Northard Exhibits 7 (NSP000025) and 8 (NSP000026).

15. The Picture of Xcellence also focuses on individual Xcellence, identifying a set of performance measures, actions that are necessary as enablers of excellence, a Human Performance Observation Program, and a set of Human Performance Tools. Collectively, these activities comprise NSPM's Human Performance Program. Northard Dir. at A23.

16. The Human Performance Program, governed by Fleet Procedure FP-PA-HU-01, encompasses a number of activities, which include 1) regular meetings of a human performance improvement team ("HUIT") consisting of members from key

departments to discuss human performance trends and corrective actions and monitor performance efforts; 2) a human performance improvement plan, prepared by the HUIT; 3) requirements for each organization to promote appropriate culture and create a learning organization; 4) training and coaching to promote error prevention; 5) a human performance event identification and investigation process; 6) communication of human performance information; and 7) recognition of good human performance practices, including individual recognition and lessons learned. Northard Dir. at A24 and Northard Exhibit 9 (NSP000027).

17. An important element of the Human Performance Program is NSPM's Human Performance Observation Program, governed by Fleet Procedure FP-PA-HU-03. The purpose of this program is to promote a leadership presence in the field on a regular basis to demonstrate the high level of commitment toward improving human performance by establishing, communicating and reinforcing clear expectations for behavior, continuous improvement, appropriate policies, efficient and effective processes, and common values. Under this program, leadership teams conduct observations of selected activities in the field on a monthly basis, followed by timely coaching and feedback to promote continuous performance improvement and reinforce use of the Human Performance Tools. The program is designed to contribute to a robust safety culture, enhance direct management involvement in site activities, improve management awareness of strengths and areas for improvement, and allow for reinforcement of expectations and standards. Northard Dir. at A24 and Northard Exhibit 10 (NSP000028).

18. NSPM's Human Performance Tools, contained in Fleet Procedure FP-PA-HU-02, establish a specific set of practices for individuals and a specific set of practices for supervisors that are intended to reduce errors. The basic purpose of these tools is to help the individual worker maintain positive control of a work situation – that is, "Do the job right the first time." A pocket sized summary of the information from this procedure is provided as the Human Performance Handbook for use in the field. Northard Dir. at A25 and Northard Exhibit 11 (NSP000029).

19. There are a number of other programs in place that assure that strong nuclear safety culture is maintained at PINGP. These programs include the Employee Concerns Program, the Differing Professional Opinions program and the PEACH process, the Corrective Action Program, and the use of periodic Safety Culture Surveys and other assessment tools. Northard Dir. at A26 and Northard Exhibits 6 (NSP000024), 7 (NSP000025) and 12 (NSP000030).

20. The Employee Concerns Program ("ECP") is a program that supports a Safety Conscious Work Environment by providing site workers with an alternative and independent avenue for raising nuclear safety concerns (as well as workplace concerns), which they may do anonymously if they wish. The ECP program ensures that the issues are addressed in a timely, effective, respectful, objective and technically intrusive manner regardless of the source of that issue, and the members of the ECP organization serve as advocates for issue resolution. If an employee is dissatisfied with the resolution of a nuclear safety concern, or feels that an unresolved nuclear safety concern exists, he or she may appeal the findings of the ECP investigation to NSPM's Chief Nuclear Officer. The Differing Professional Opinions program and the PEACH process are additional means whereby employees may bring issues forward for resolution. Of course, any PINGP employee is free to bring any concern to the NRC's attention. Northard Dir. at A26.

21. In addition, these programs have been augmented since December 2009 with initiatives that address a substantive cross-cutting issue in the area of human performance identified by the NRC last year. Id.

22. Taken together, these programs and tools promote a strong safety culture in all aspects of PINGP's operations. Id.

23. All employees are provided training on maintaining a strong safety culture at PINGP. General Access Training, which is provided to all new employees and is conducted annually for all current employees, includes safety culture and the various programs that serve to enhance it as one of the training topics. Classroom training is used for the new employee training, and computer-based training is utilized for the annual requalification. Each course requires that the employee take

a knowledge test on the topic, with a minimum 80% of the answers needing to be answered correctly in order to pass the test. Finally, Safety Conscious Work Environment training is provided annually to the plant employees; this training summarizes all the safety culture programs. Id. at A27.

24. Information on the safety culture programs is posted on bulletin boards throughout the plant, and is included on the Prairie Island home page, creating easy and continued access to them. Numerous posters are hung in hallways, conference rooms, training rooms, and other common spaces, emphasizing safety culture and human performance tools. These posters include, for example, the Risk Management Principles, the Risk Management Behaviors, and the Principles for a Strong Nuclear Safety Culture. These posters also emphasize the various methods that can be used to report workplace concerns. Newsletters, such as the Team Notes covering the daily D-15 meetings, continually reinforce appropriate safety culture and behavior. Id. at A28.

C. Response to Safety Culture Contention Claims

In its Safety Culture Contention, the PIIC raises several issues as evidence of the existence of a weak safety culture at PINGP. Those are (1) the presence of leakage from the refueling cavity in both units; (2) the issuance of "White" Findings by the NRC against PINGP with respect to radioactive material shipment deficiencies (both PINGP units), improper valve positioning (Unit 1), and design of the component cooling water system (Unit 2); (3) the identification by the NRC of "substantive crosscutting" issues in the area of Human Performance; and (4) the existence of concerns with the Corrective Action Program at PINGP. Each of these issues is demonstrated in NSPM's testimony not to be indicative of a deficient safety culture; moreover, the testimony shows that the safety culture at the plant is strong and more than adequate to provide reasonable assurance that the effects of aging of the plant's structures, systems and components subject to 10 C.F.R. Part 54 will be adequately managed.

1. Refueling Cavity Leakage

The evidence provided by NSPM witness Mr. Skoyen demonstrates that there is no support for PIIC's claim that the history of the refueling cavity leakage evidences a weak safety culture at PIGNP. NSPM's performance has not been deficient nor has there been dereliction of its obligations as licensee. NSPM has been proactive in pursuing multiple avenues to resolve the leakage issue and conducting a detailed root cause evaluation when previous corrective measures proved less than completely effective. NSPM has also commissioned several independent engineering evaluations, which have shown that the leakage has no safety significance. NSPM has repaired, or has scheduled repair of, the components identified as the source of the leakage, and will continue to do so if new leakage sources are identified. Overall, NSPM's handling of the refueling cavity leakage issue is indicative of a strong, rather than weak, safety culture at PINGP.

a. History of Refueling Cavity Leakage Issue

25. The PINGP refueling cavity is an area within the Reactor Containment Vessel ("RCV") surrounding the reactor pressure vessel ("RPV"). It contains the fuel transfer area and the reactor internals storage areas, and is located between the RPV and the spent fuel pool. The cavity's function is to allow for refueling by transferring spent fuel from the RPV through the fuel transfer tube into the spent fuel pool in the auxiliary building, followed by transfer of fresh fuel assemblies from the auxiliary building back into the RPV. Skoyen Dir. at A7.

26. The refueling cavity is composed of concrete walls, inside of which is a stainless steel liner. It contains the refueling cavity water during refueling operations. In the upper region of the cavity, the reactor vessel flange seals the RPV from the refueling cavity through a segmented rubber seal at the head of the RPV. In the lower region of the cavity, the upper and lower reactor internals stands provide storage for the internals, and the transfer area contains components including the fuel transfer tube, the rod control cluster ("RCC") change fixture, and the RCC change fixture guide tube along the cavity wall. Id. and Skoyen Exhibit 2 (NSP000002).

27. The refueling cavity is filled with borated water during the refueling process, so that spent fuel may be transferred underwater from the RPV to the spent fuel pool, followed by transfer of new fuel back into the RPV, and then it is drained after refueling is completed. Skoyen Dir. at A7.

28. Typically, the refueling cavity remains flooded for 10-14 days during refueling. It is kept dry during the rest of the unit's approximately 18-month operating cycle. It cannot be accessed during regular plant operation because of the heat and radiation levels in the area of the RPV. Id.

29. PINGP operating personnel first observed indications of refueling cavity leakage in the containments of both PINGP units in 1987 and 1988. Leakage typically would begin two to four days after the refueling cavity was flooded and would end about three days after the cavity was drained. The leaking substance was determined to be borated water, such as is used to flood the refueling cavity. Over time, leaking borated water was observed in sump B (the residual heat removal ("RHR") system loss-of-coolant accident recirculation sump), sump C (the sump under the RPV at the bottom of the containment), the regenerative heat exchanger room (located underneath the refueling cavity), the reactor coolant drain tank ("RCDT") space, and various walls, floors, and vaults. Id. at A8.

30. In 1988, following the initial indications of refueling cavity leakage, PINGP completed weld repairs on the Unit 1 cavity at the suspected locations of the leakage. Since 1988, pumping of sump B and sump C in both units has been conducted, as needed, during refueling outages to remove any water that has reached the sumps. Id. and Skoyen Exhibit 4 (NSP000004).

31. During the 1998 Unit 2 outage, leakage was observed in sump B. A non-conformance report ("NCR") was prepared and issued to address this problem. NSPM personnel promptly initiated a series of efforts during this outage to pinpoint the source of the leakage, including vacuum box testing of accessible seams and fasteners as well as dye penetrant testing of suspect areas that could not be vacuum box tested. Skoyen Dir. at A8.

32. During the 1998 Unit 2 outage, NSPM evaluated the condition of the containment vessel wall by partially removing the grout (a thin mortar used for structural shaping purposes) around the penetration. The vessel showed no signs of degradation. Id.

33. PINGP employed various methods to prevent the leakage, including installation of a strippable liner to the refueling cavity, starting during the 2000 Unit 2 outage and continuing through the 2003 Unit 2 outage. These applications had inconsistent results – sometimes they succeeded in preventing leakage from occurring, while other times the leakage occurred despite the coating application. Skoyen Dir. at A8 and Skoyen Exhibit 4 (NSP000004).

34. Starting in 2004, PINGP began caulking suspected leak paths at the baseplates and fasteners of the internals stands and the RCC change fixture. After similar caulking steps were performed during the 2005 Unit 2 outage, the 2006 Unit 1 outage, and the 2008 Unit 1 outage, no leakage from the refueling cavity was observed. However, during the 2006 Unit 2 outage, leakage was observed through the grout in sump B. In the next Unit 2 outage in 2008, despite caulking, leakage was reported in the ceiling of the regenerative heat exchanger room, the 22 vault (where steam generators and reactor coolant pumps are located), and sump B. Id.

35. Because of the limited success of the measures implemented until then to address the leakage, a corrective action document identifying the problem, CAP 1160372, was initiated in the fall of 2008 recommending performance of a Root Cause Evaluation ("RCE") to evaluate the leakage problem and develop a permanent solution to it. After a comprehensive evaluation was made, a Root Cause Evaluation Report, RCE 01160372-01, *Refueling Cavity Leakage, Event Date 1988-2008* was prepared and issued in 2009. Id. and Skoyen Exhibit 4 (NSP000004).

36. The RCE and two subsequent engineering evaluations (EC 14139 and EC 15044) were commissioned to identify the sources of the leakage and assess the potential effects of borated water leakage on the containment vessel, concrete, and concrete reinforcing bar. The evaluations involved a wide-ranging analysis of the history of observed leakage; the likely leakage source(s); an analysis of relevant operating experience; and recommendations for corrective actions, including effectiveness

reviews. The RCE, completed in April 2009 and updated in February 2010 to reflect results of fall 2009 Unit 1 repairs, determined that the most likely sources of lower cavity leakage were the floor embedment plates for the reactor vessel internals stands and the RCC assembly change fixture, as well as the RCC change fixture guide tube supports located on the cavity wall. Skoyen Dir. at A8 and Skoyen Exhibit 4 (NSP000004).

b. Difficulties in Identifying Sources of Leakage

37. It is not possible to look for leakage from the cavity save immediately before the refueling cavity is flooded (approximately three days), or after the refueling operation is completed and the cavity is drained, but before plant operations resume (approximately seven days). These periods occur only once every approximately 18 months. Skoyen Dir. at A7.

38. Leaking occurs only (if at all) for a few days every eighteen months, during the refueling process, and the areas where the leakage occurs are partially or totally inaccessible. None of the suspected leakage points are accessible at the time they are leaking. One cannot access the refueling cavity at all while the reactor is at power. The cavity is similarly inaccessible while flooded, making it impossible to observe precisely which points are responsible for leakage. It is accessible only during outages, in the few days before and after cavity flooding. The space in the upper cavity located directly underneath the sandplug covers is accessible only by removing the sandplugs, an operation that is scheduled to be performed only once every ten years. Sump C is not accessible while the cavity is flooded due to high radiation levels and can only be accessed for a period of approximately seven days after the upper cavity is drained. Sump B remains accessible either directly or through the inspection opening for the majority of the outage. Id. at A9.

39. As a result of the inability to access the leak source points during the refueling process, one must look for evidence of where leakage collects and identify probable sources based on inspecting and repairing possible leak points, followed by testing the effectiveness of those repairs by observing any leakage during the next flooding. Id.

40. Due to the limited accessibility of relevant components, the refueling outage allows only limited opportunities for making repairs and performing inspections. There is only a few-day period at the beginning and end of each outage in which these actions may be taken. During the last two outages, PINGP personnel have allocated as much time as possible at the beginning of the outages in order to carry out repairs. Id.

41. NSPM has determined that there are two leakage sources, corresponding to locations in the lower cavity and the upper cavity. The lower cavity leakage, which accounts for leakage entering sump B and the regenerative heat exchanger room, occurs where the internals stands and RCC change fixture anchor studs penetrate the associated embedment plates on the refueling cavity floor, and where anchor studs penetrate the RCC guide tube supports on the cavity wall. Skoyen Dir. at A10 and Skoyen Exhibit 4 (NSP000004).

42. The anchor studs are secured to the embedment plates by being set in through-holes in the plates and then seal-welded to the plates. These components have welds between the studs and embedment plates that are designed to form a watertight seal. Although inspection of the condition of the seal welds is difficult due to their inaccessibility, it appears that they have developed pin-hole-sized leaks or cracks. Those cracks, if existing, create a leak path along the threads of the studs, which then allows water to flow under the cavity liner into cracks in the concrete and down, emerging in the ceiling and walls of the regenerative heat exchanger room and eventually through the inner wall of the containment vessel. Once at the containment vessel, the water travels down and horizontally, potentially filling any voids between the containment vessel and concrete down to the low point of the bottom head of the containment vessel. As the water then rises, it starts to leak through various construction joints, cracks, and the grout in sump B. Id.

43. The upper cavity leakage, which accounts for the leaked refueling water entering sump C and the leakage entering sump B, results from faulty seals at the sandplug covers, allowing refueling water to reach sump C at the bottom of the reactor

containment. This leakage from the upper cavity likely enters sump B indirectly through one of the construction joints. Id. and Skoyen Exhibit 9 (NSP000009).

c. Recent Repairs

44. In the fall of 2009, NSPM completed permanent repairs to the Unit 1 floor embedment plates for RCC change fixture supports and internals stands supports by removing existing nuts, replacing them with blind nuts that were seal-welded to the baseplate, and applying a seal weld between the baseplate and the embedment plate. During refueling cavity flooding following these repairs, minor leakage was found in the regenerative heat exchanger room. The guide tube supports are the likely leakage source due to the similarity of their design to that of the internals stand supports. Skoyen Dir. at A8 and Skoyen Exhibit 4 (NSP000004).

45. Prior to cavity flooding during the spring 2010 Unit 2 outage, NSPM similarly completed a series of repairs to the Unit 2 reactor vessel internals stands supports and RCC assembly change fixture supports, in addition to the RCC assembly guide tube supports. Skoyen Dir. at A8 and Skoyen Exhibits 9 (NSP000009) and 10 (NSP000010).

46. The fall 2009 repair to the Unit 1 floor embedment plates eliminated 95 to 97.5 percent of the leakage historically experienced from the lower cavity. Following these repairs in Unit 1, only minor leakage occurred on the ceiling of the regenerative heat exchanger room, after the cavity had been flooded for over 14 days, in the amount of 0.05 gph, or roughly seven drops per minute. No leakage was observed in sump B. In response to this minor continued leakage in the regenerative heat exchanger room, NSPM performed expanded inspections, including examination of the liner plate to floor embedment plate fillet welds and transfer tube welds. Skoyen Dir. at A12 and Skoyen Exhibit 4 (NSP000004).

47. Following the fall 2009 Unit 1 repairs, vacuum box testing of the seam welds of the liner plate in the lower cavity revealed no leakage. Examination of the transfer tube welds, including both dye penetrant and visual inspection, showed no indications of leakage. Inspection of the lower cavity presented no depressions or soft areas

indicative of water having eroded the surface. There is no evidence of leakage having reached the containment vessel or the steel pressure vessel. Id.

48. Following the 2010 Unit 2 repairs, leakage in sump B was observed almost immediately after the cavity was flooded. The leak initially occurred at a rate of 0.8 gph which decreased gradually over a few days to 0.02 gph. By the time the refueling cavity was drained, leakage had diminished almost entirely in sump B. NSPM believes that this leakage originated in the upper refueling cavity, from faulty seals at the sandplug covers, and migrated into sump B either before or after reaching sump C. The spring 2010 Unit 2 repairs appear to have resolved the leakage from the lower cavity, eliminating over 97.5 percent of the total leakage historically experienced. During the spring 2010 Unit 2 outage, less than 0.01 gph of leakage, or roughly 1 drop per minute, was observed coming from the mezzanine adjacent to the regenerative heat exchanger room. Leakage in sump B started at an initial rate of 0.8 gph, which gradually decreased to essentially no leakage by the end of refueling once the cavity was drained below the upper cavity. Skoyen Dir. at A8 and A12 and Skoyen Exhibits 9 (NSP000009) and 10 (NSP000010).

d. Safety Significance of Leakage

49. PINGP has commissioned several studies to assess the potential safety significance of the refueling cavity leakage, to ensure that leakage has not adversely affected the integrity of the containment structures. In 1998, NSPM commissioned Automated Engineering Services ("AES") to perform an evaluation of the effects of borated water on concrete, the reinforcing steel bars (rebar), and the RCV steel plate. This evaluation concluded that the effects of the leakage on the containment structures would be minimal and would have no safety significance. Skoyen Dir. at A8 and Skoyen Exhibit 5 (NSP000005).

50. In 2006, PINGP requested that AES re-review the cause and potential safety significance of the leakage relative to its earlier evaluation performed in 1998. The resulting evaluation confirmed that the basis and conclusions of the 1998 report remained valid, and that the integrity of the concrete, rebar, and containment shell had not been compromised. Skoyen Dir. at A8 and Skoyen Exhibit 6 (NSP000006).

51. PINGP commissioned yet another outside expert study from Dominion Engineering, Inc. ("DEI"), which was completed in February 2009. The DEI evaluation, R-4448-00-01, *Evaluation of Effect of Borated Water Leaks on Concrete, Reinforcing Bars, and Carbon Steel Plate of the Containment Vessels at Prairie Island Units 1 and 2*, concluded that there was no evidence of significant corrosion of the steel containment vessel, that there was no evidence of significant degradation of concrete, and that any corrosion of rebar in the concrete would be minor. Skoyen Dir. at A8 and Skoyen Exhibit 8 (NSP000008).

52. The exposure of concrete to borated water is estimated to have affected the concrete under the cavity liner to a depth of no more than 0.31 inches. There has been no observed thinning or corrosion of the concrete. Although this is an insignificant amount of thinning in most areas, NSPM will continue to evaluate it, especially in areas where concrete in contact with the liner is thin at the wall near the transfer tubes. This thin area is insignificant for structural integrity purposes because it does not support any other components. Skoyen Dir. at A16 and Skoyen Exhibits 4 (NSP000004), 8 (NSP000008) and 12 (NSP000012).

53. The NRC Staff has agreed with NSPM's conclusion that the leakage lacks safety significance. In its October 2009 SER, the Staff found that "any loss of load carrying capacity of the concrete would be negligible since the concrete sections are four to five feet thick." Skoyen Dir. at A16 and Skoyen Exhibit 14 (NSP000014).

54. NSPM and its outside engineering consultants have performed multiple examinations of the RCV walls. In all instances, wall thickness measurements have been shown to be at or above ASTM specifications, and no corrosion or pitting of the containment vessel have been observed. Skoyen Dir. at A17 and Skoyen Exhibits 4 (NSP000004), 8 (NSP000008) and 14 (NSP000014).

55. With respect to the rebar, the 2009 DEI evaluation determined that there were no detected signs of reinforcing bar corrosion, and that any areas in which the rebar cover depth is the minimum allowed would have experienced only insignificant corrosion of 0.016 inch. Even if the lower region of the containment had been

wetted by borated water for much of the plant's life, no corrosion of this rebar is expected to have occurred. Skoyen Dir. at A18 and Skoyen Exhibit 8 (NSP000008).

56. During the 2009 Unit 1 outage, rebar was uncovered during excavation of grout in sump B to access the containment vessel for inspection. Sump B was chosen for inspection because it is located at a lower elevation and consistently shows wetting when refueling cavity leakage occurs. The visible rebar did not exhibit any degradation or corrosion. Rebar thickness remains almost identical to what it was when it was installed during plant construction. Skoyen Dir. at A18 and Skoyen Exhibit 4 (NSP000004).

e. Future Actions

57. During the 2011 Unit 1 outage, NSPM will repair the RCC change fixture guide tube supports and will repair a liner to floor embedment plate fillet weld porosity indication. It has also entered a work order to install gaskets to repair leaks in the sandplug covers in both units. Skoyen Dir. at A20 and Skoyen Exhibits 7 (NSP000007) and 9 (NSP000009).

58. During the two consecutive refueling outages following cavity leak repairs in each unit, NSPM will perform visual inspections of the areas where reactor cavity leakage had been observed previously to confirm resolution of the issue (Commitment 42). Open corrective actions will remain so through the end of the next refueling outage in each unit to confirm resolution of the leakage issue. Skoyen Dir. at A20 and Skoyen Exhibits 10 (NSP000010), 13 (NSP000013), 14 (NSP000014) and 15 (NSP000015).

59. The Structures Monitoring Program and the ASME Code Section XI, Subsection IWE Program will continue to monitor for any remaining leakage and evaluate the condition and integrity of containment vessel structures. Additionally, NSPM will be conducting further inspections and tests to ensure no vessel degradation has occurred. Skoyen Dir. at A20.

60. The containment vessels in both units are subject to ongoing inservice inspections and evaluations to ensure that they have not degraded and maintain their required

thickness, in accordance with ASME Code Section XI, Subsection IWE and 10 CFR § 50.55a. In the unlikely event that these inspections present indications not acceptable by evaluation under ASME Code Section XI, the PINGP Subsection IWE program requires that such indications be repaired or replaced in order to be acceptable for continued service. NSPM has committed (Commitment 41) that, during the next refueling outage for each unit, following removal of concrete from sump C below the reactor vessel, a contractor will be performing a visual and ultrasonic evaluation of the containment vessel to determine the thickness and validate the prior results showing no degradation. NSPM will also perform a petrographic examination of any removed concrete, including an evaluation of any water found at the location of removed concrete. Skoyen Dir. at A20 and Skoyen Exhibit 14 (NSP000014).

61. Further, NSPM has committed (Commitment 44) to removing, during the next outage for each unit, a concrete sample that has been wetted by borated water and testing it for compression strength as well as a petrographic examination. Id.

62. NSPM will complete all leakage-related commitments (Commitments 41, 42, and 44) prior to the license renewal period. It will continue to manage aging in the containment structure and the vessel using the Structures Monitoring Program, as well as the ASME Code Section XI, Subsection IWE Program. Any items of concern will be entered into the Corrective Action Program for evaluation and correction. Id.

f. Safety Culture Implications

63. Some safety culture issues were identified in the RCE investigation of the reactor cavity leakage. Those issues stem from past insufficiencies in accountability at the organizational level that have since been remedied by NSPM's shift to a strict process-driven approach to handling identified problems. Under the current Corrective Action Program ("CAP"), the manager or supervisor responsible for a corrective action is accountable for it regardless of who handles it personally. Corrective action documents are now classified by issue significance, such that the more significant the issue is, the greater the levels of management accountability

and responsibility that apply. The formal requirements of PINGP's current CAP ensure proper organizational decision-making and oversight, preventing a single individual from undermining its effectiveness. Skoyen Dir. at A22 and Skoyen Exhibit 4 (NSP000004).

64. The RCE noted that numerous CAP documents over the years indicate a continuing recognition of the refueling cavity issue at all levels of the organization and appear to have objectively assessed the issue, including an evaluation of the potential for and consequences of any theoretical degradation. Also, NSPM management has encouraged personnel involved with refueling cavity leakage to report the issue and take corrective action. These are positive safety culture indications. Skoyen Dir. at A23 and Skoyen Exhibit 4 (NSP000004).

g. Safety Culture Contention Claims

65. Contrary to the claims in the Safety Culture Contention, NSPM currently has a very good understanding of the sources of the refueling cavity leaks. It has performed repairs that have eliminated 95-97 percent of previous leakage in Unit 1 and 97.5 percent of previous leakage in Unit 2, has identified the RCC assembly guide tube wall embedment plates as a likely source of the minor remaining leakage, and is scheduled to repair these plates for Unit 1 during the next Unit 1 outage in 2011 and will identify any additional inspections and repairs prior to that time. Further, based on the identification of the sandplug covers as the likely source of the remaining upper cavity leakage in Unit 2, NSPM will install gaskets for those covers in both units during the next refueling outages. Skoyen Dir. at A24 and A25.

66. Likewise, contrary to the contention, there is no safety significance to the leakage. After repeated inspections and evaluations over many years, no evidence has been found of degradation in the containment vessels, the rebar, or the concrete resulting from the leakage. Past inspections and evaluations and those to be performed in the future, together with the repair programs NSPM has conducted and will continue to perform, provide assurance that the theoretical danger to the integrity of the containment will not materialize. Skoyen Dir. at A26.

67. The claim in the Safety Culture Contention that NSPM did not acknowledge the importance of the refueling cavity leakage to aging management until the NRC audit in the fall of 2008 is also incorrect. In addition to multiple work orders and other measures to address the issue, NSPM ordered an independent safety evaluation from AES in 1998 to assess the potential degradation effects of borated water leakage on the containment structure. Even though this evaluation, the conclusions of which were confirmed in 2006, found that any leakage effects would have no safety significance, NSPM conducted numerous tests and implemented a series of repairs to stop the leakage. These measures were taken starting in 1999 under both the Structures Monitoring Program and the ASME Section XI, Subsection IWE Program and have continued since that date. Skoyen Dir. at A28.

68. There is also no basis for the assertion in the Safety Culture Contention that the refueling cavity leakage issue negatively reflects on NSPM's ability to conduct an aging management program during the license renewal period. NSPM's performance has not been deficient nor has there been dereliction of its obligations as licensee. NSPM has been proactive in pursuing multiple avenues to resolve the leakage issue and conducting a detailed root cause evaluation when previous measures proved less than completely effective. NSPM has repaired, or will soon repair, the components identified as the source of the leakage, and will continue to do so if new leakage sources are identified. In the SER, the NRC staff concluded that NSPM's remedial measures and commitments demonstrate its ability to effectively implement the aging management program. Following its own investigation and analysis, the Advisory Committee on Reactor Safeguards agreed with the NRC staff's assessment. Those independent assessments reinforce the conclusion that NSPM will be able to conduct an effective aging management program during the license renewal period. Skoyen Dir. at A27 and A29 and Skoyen Exhibits 13 (NSP000013) and 14 (NSP000014).

2. Other Safety Culture Claims

(1) White Findings

69. When the NRC performs an inspection of an operating reactor, any discrepancies that are found are evaluated and given a color designation based on their safety significance. Green inspection findings indicate a deficiency in licensee performance that has very low risk significance and therefore has little or no impact on safety. White, Yellow, or Red inspection findings each, respectively, represent a greater degree of safety significance, resulting in a corresponding increase in regulatory attention. The significance of findings in colors other than Green is associated with the potential risk that the condition will result in an increased core damage frequency. White findings are findings of low to moderate safety significance. Northard Dir. at A29 and Northard Exhibit 13 (NSP000031).

70. The NRC determines its regulatory response to White findings in accordance with an Action Matrix that provides for a range of actions commensurate with the significance of the inspection results. Findings "greater than Green" trigger increased regulatory attention. For plants that do not have all Green inspection findings, the NRC will perform additional inspections beyond the baseline program and initiate other actions commensurate with the safety significance of the issues. White findings have the least safety significance of the greater than Green findings, being classified as having "low to moderate" significance. Northard Dir. at A30.

71. There have been three White findings assessed against PINGP since January 2008. First, on July 31, 2008, both Unit 1 auxiliary feedwater pumps ("AFW Pumps") auto-started following a Unit 1 reactor trip. One of the pumps, the 11 turbine-driven auxiliary feedwater pump, tripped 42 seconds later. A subsequent investigation found that the instrument manifold isolation valve for the discharge pressure switch was out of position (closed instead of open), which caused the pump to trip on a low discharge pressure. This occurrence resulted in a White finding against Unit 1. Second, on October 29, 2008, NSPM shipped nuclear fuel inspection tooling containing radioactive material that was not adequately surveyed or packaged to assure that, under conditions normally encountered with over-the road

transportation, the radiation level on the external surface of the package would not exceed allowable limits set in the U.S. Department of Transportation regulations for radioactive material shipments. This resulted in a White finding against both Units 1 and 2. Third, on July 31, 2008, NSPM identified that a failure of a Unit 1 or a Unit 2 turbine building high energy line could impact the continued operability of the Unit 2 component cooling water ("CCW") system. This condition rendered the Unit 2 CCW system inoperable because a high energy line break could cause a complete loss of CCW inventory, if the CCW piping was severed. This condition resulted in a White finding being assessed against Unit 2. Id. at A31.

72. The existence of the conditions resulting in the three White findings has not compromised the overall safe operation of PINGP. In its Annual Assessment letter for the calendar year 2009 the NRC, while making reference to these White findings, concluded: "Overall, Prairie Island Nuclear Generating Plant, Units 1 and 2, operated in a manner that preserved public health and safety and fully met all cornerstone objectives." Id. at A32 and Northard Exhibit 14 (NSP000032).

73. The PIIC's characterization of the White findings as indicative of a substantive cross-cutting issue in the area of human performance is incorrect. While both the White findings and the open substantive cross-cutting issue in the area of human performance were mentioned in the same letter from the NRC to NPSM on the agency's mid-cycle performance review for PINGP for the period mid-2008 to mid-2009, the determination made in that letter of a substantive cross-cutting issue in the area of human performance related to the existence of "25 findings documented with cross-cutting aspects in the HP area," and not to the White findings. The classification of a deficiency as a "White" finding relates only to its perceived safety implications and is not in itself indicative of a deficiency in the safety culture at a facility. Northard Dir. at A95 and Northard Exhibit 21 (NSP000039).

(a) Mispositioned Manifold Isolation Valve Switch

74. NSPM conducted a comprehensive evaluation of the AFW trip event and prepared and issued a Root Cause Evaluation (RCE) Report, RCE 01146005, 11 Turbine-

Driven Auxiliary Feedwater Pump Discharge Pressure Switch Manifold Isolation Mispositioning. Northard Dir. at A33 and Northard Exhibit 15 (NSP000033).

75. The RCE evaluation determined that an I&C technician or plant operator erroneously operated the manifold block isolation valve for Pressure Switch PS-17700 (11 TDAFWP Lo Discharge Pressure Trip Pressure Switch). The following conditions were determined to have existed: (1) During that time period, there were seven surveillance procedures completed that operated valves in the vicinity of the PS-17700 manifold isolation valve. These valves are identical in design to the PS-17700 manifold valve and are in close proximity to the valve. There were no steps in these procedures to check the position of PS-17700 manifold isolation valve because none of the procedures calls for operating this valve. (2) The PS-17700 manifold isolation valve had insufficient configuration controls, for it did not have a locking device. (3) The valve was not included in the equipment database, the complete valve lineups, or relevant drawings, and was not provided with an identification label. The root cause of the event was identified as inadequate configuration controls for components that have the potential to adversely impact the design function of safety-related structures, systems and components. Northard Dir. at A34 and Northard Exhibit 15 (NSP000033).

76. No human performance deficiencies indicative of a weak safety culture were involved in this event because, upon review of all its elements, it was determined that the event was the result of poor human factors of the instrument sensing line that propitiated the failure. In addition, it was an isolated, self-revealing event that was promptly corrected and effectively addressed. Northard Dir. at A36.

77. NSPM took several actions to prevent the event's recurrence. They included: (1) putting in place revised configuration control methodologies; (2) forming a project team to evaluate all safety-related systems to determine if there are other components that, if mispositioned, might prevent a safety-related system from performing its design function; (3) including the type of component for which the mispositioning occurred (level "B" components) in the equipment database and drawings; (4) installing locking devices have been installed in those components;

and (5) labeling all level B components in the field -- over two thousand manifold valves are now newly labeled with permanent valve tags. While these actions were implemented, interim measures were taken to mitigate the configuration control issue by installing locking devices on all of the Auxiliary Feed Water system discharge and suction pressure switch manifold isolation valves. Id. at A37.

78. NSPM also formed a cross-functional team in March 2009 to develop a comprehensive Performance Recovery Plan to address performance issues at the plant and improve station human performance. Id.

79. Once the root cause evaluation report RCE 01146005 had been prepared and submitted to the NRC, the NRC conducted an inspection to examine the analysis performed by NSPM and the corrective actions to prevent recurrence that had been taken. In Supplemental Inspection Report 05000282/2009011 issued on October 15, 2009, the NRC noted that a plant component labeling, blocking and locking program had been initiated to address the configuration control issue, and that NSPM had a Performance Recovery Project underway to broadly address performance issues at the plant. The NRC judged these actions to be an acceptable way of addressing the valve mispositioning issue and removed the White finding on that basis. Northard Dir. at A38 and A39 and Northard Exhibit 16 (NSP000034).

80. The PIIC refers to Information Notice 2009-11 issued by the NRC to the operating license holders alerting to a potential problem with configuration control errors at operating reactors, and citing the PINGP Unit 1 mispositioned manifold isolation valve switch as a recent example. However, while NRC Information Notice 2009-11 cited by PIIC does mention several factors as potentially being the causes of configuration control errors, the Information Notice cross-references eighteen other plants where such errors occurred and does not associate any of the factors with the errors at a given plant. In particular, nowhere does the Information Notice indicate that any of the factors it lists were present at PINGP. Northard Dir. at A96 and Northard Exhibit 41 (NSP000059).

(b) Radioactive Material Transportation

81. On October 29, 2008, PINGP shipped contaminated fuel sipping equipment to a vendor in Pennsylvania following decontamination of the equipment after its removal from the spent fuel pool. The equipment was packaged by both NSPM and contractor staff and shipped in an open transport vehicle. Upon receipt by the vendor, package surface dose rates were found to exceed applicable U.S. Department of Transportation limits, primarily due to a radioactive particle being embedded in the fuel sipping equipment, which was located near the outside wall of the shipping container. The fuel sipping equipment was found not to be properly braced or secured and shifted within the package during transport. Based on the results of a radiological risk assessment, a final significance determination for a White finding was issued by the NRC on May 6, 2009. Northard Dir. at A40.

82. NSPM conducted a root cause evaluation of the incident and issued a report, Root Cause Evaluation Report No. 01157726; Radioactive Material Shipment Exceeded DOT Limits. The evaluation made the factual determination that, during Unit 2's refueling outage in the fall of 2008, potentially degraded fuel assemblies were tested for cladding integrity with vendor fuel sipping equipment. That equipment was decontaminated, demobilized, and packaged for shipment back to the vendor. Upon receipt at the vendor's facility, elevated radiation levels were detected. Opening of the shipment package revealed that a small radioactive particle was embedded into the umbilical cable to the lid of the fuel sipping canister. The fuel sipping equipment (lid and umbilical cable) was found to be not properly braced, nor secured as required; apparently, the lid and the umbilical cable shifted from the time of the PINGP shipping package departure to its arrival at the vendor. Additionally, two other radioactive particles were detected inside the shipping box. These facts indicated that the on-site radiological surveys prior to shipment were not sufficient for detecting highly radioactive small particles, and that the fuel sipping equipment was not properly braced nor secured in its package for shipment under conditions normally incident to transport. Also, Station procedures required that formal job planning be conducted before removing items from the spent fuel pool, but this requirement was not fulfilled in this instance; thus, there was not adequate planning

and evaluation to assess the hazard and the potential radiological impact to the workers. Northard Dir. at A41 and A42 and Northard Exhibit 17 (NSP000035).

83. The evaluation determined that the incident had two root causes: (1) inadequate procedures and methods to successfully evaluate, package and ship radioactive materials in accordance with NRC and DOT regulations; and (2) an inadequate risk management process leading to inadequate management oversight of the radioactive material shipment program. In addition, there was ineffective incorporation of industry operating experience into the radioactive material shipment program, and deficient training and certification programs for radiation protection personnel that perform shipment-related activities. Northard Dir. at A43 and Northard Exhibit 17 (NSP000035).

84. NSPM took several corrective actions to address this incident. The actions included the development of new shipping procedures and enhancement of existing ones, improvements to the training and qualification program for staff involved in shipment activities, and the implementation of an integrated risk management program to assure management engagement and adequate oversight of potentially risk-significant shipments. Northard Dir. at A44.

85. The root cause evaluation determined that several safety culture discrepancies contributed to the occurrence of this incident: there were human performance deficiencies in the areas of decision making, resource allocation, work control, and work practices. There was also insufficient consideration of operating experience with radioactive material shipment issues. Id. at A45 and Northard Exhibit 17 (NSP000035).

86. NSPM addressed the identified safety culture issues through the corrective actions taken in response to this incident. More active engagement by management in shipment activities was mandated, including required reviews for higher risk shipments, has been added to the program requirements. Also, appropriate training for workers involved in shipping activities is required and verified, and the use of specified radiation monitoring equipment is prescribed in the procedures. Finally, the Radiation Protection Manager's review of the plant staff's evaluation of

shipping-related operating experience throughout the industry is mandated. Northard Dir. at A46.

87. Once the RCE had been prepared and submitted to the NRC, the NRC conducted on December 4, 2009 an inspection to examine the analysis performed by NSPM and the corrective actions to prevent recurrence that had been taken. In Supplemental Inspection Report 05000282/2009015 and 05000306/2009015 issued on January 12, 2010, the NRC reviewed the root and contributory cause analyses and corrective actions taken, judged these actions to be an acceptable way of addressing the issue, and removed the White finding on that basis. Northard Dir. at A47 and A48 and Northard Exhibit 20 (NSP000038).

(c) CCW System Vulnerability

88. The design of the Unit 2 component cooling water ("CCW") system includes piping routed through the turbine building. While the CCW system is safety-related, the CCW piping in the turbine building served only non-safety related loads. An evaluation by NSPM issued on July 31, 2008, identified that a failure of a turbine building high energy piping line could sever the adjacent CCW piping, thereby impacting the continued operability of the Unit 2 CCW system. An additional operability review determined that this scenario could cause a complete loss of CCW inventory, because operators might not have sufficient time to isolate the CCW piping in the turbine building from the portion of the CCW system that performs safety-related functions prior to the loss of suction to the operating component cooling pumps. NSPM also determined that the operators' ability to bring Unit 2 to a cold shutdown condition following a high energy line break ("HELB") and a failure of the CCW system was adversely impacted. Northard Dir. at A49.

89. When the CCW System vulnerability was discovered, Operations personnel immediately declared both trains of the Unit 2 CCW system inoperable and entered a Technical Specification requiring a Unit 2 shutdown. In the process of preparing to shut down the unit, operations personnel closed multiple CCW system manual isolation valves, isolating the non-safety related CCW piping in the turbine building

from the rest of the system. The CCW piping in the turbine building has now been capped and is no longer used. Id. at A50.

90. NSPM performed a root cause evaluation of the condition and issued a report, RCE 01145695, "Component Cooling Piping Adjacent to HELB Location in Turbine Building" ("CCW RCE"). The CCW RCE evaluation determined that, while there were a number of analyses of HELB interaction events in the Auxiliary Building (where the safety-related piping for the CCW is contained), no comparable analysis had been performed for the Turbine Building, even though the potential need for such an analysis had been identified several years earlier. From 2000 through 2008, several opportunities existed for the CCW/HELB interaction in the Turbine Building to be identified and referred to the Corrective Action Program. For a variety of reasons, the opportunities were missed. Id. at A52 and A53 and Exhibit 19 (NSP000037).

91. Weaknesses in the following Safety Culture components were identified as either root causes or contributing causes of the CCW vulnerability condition: human performance, work practices, management and supervisory oversight; problem identification and resolution; Corrective Action Program; complete, accurate and timely identification of issues; and systematic evaluation of relevant internal and external operating experience. The principal weaknesses involved in this incident were (1) the failure to update the original HELB analysis of the Turbine Building, and (2) the failure to properly assess and investigate comments in a vendor report that indirectly suggested that a safety concern may be posed by the design configuration in the building. With respect to the second weakness, there were statements from a draft study prepared by a vendor examining options to resolve cold chemistry laboratory piping issues which, if followed up through a detailed review, could have identified the CCW vulnerability. However, because this study was concerned with non-safety related modifications to the Turbine Building, it was not reviewed in a timely manner with a focus on potential plant operability issues. Northard Dir. at A54 and A55.

92. NSPM took several actions to address the CCW vulnerability. The piping line whose failure could affect CCW operability has been permanently disconnected from the Turbine Building and capped. A HELB design basis document and program document are being prepared and implemented. These documents will establish the HELB requirements at PIGNP and complete actions necessary to ensure the site is in compliance with the requirements. Also, the short term and long term personnel resource requirements for sustainability of the HELB program have been established. Id. at A56.

93. NSPM also took several actions to address the Safety Culture weaknesses. A timely review of project studies completed by vendors is now required, to ensure potential issues are reviewed for plant impacts; a new procedural requirement calls for the assignment of a Project Manager for all significant plant projects; the requirements and expectations for CAP initiation by Engineering have been strengthened; and Human Performance training has been provided to Engineering using this issue as a specific example. Id. at A57.

(d) Current Status of White Findings

94. All White findings against Unit 1 have been resolved and the unit is back in the "Licensee Response Column," meaning that NSPM will be subject to only the NRC baseline inspection program, and identified deficiencies will be addressed through NSPM's corrective action program. Unit 2 remains in the Regulatory Response Column due to the remaining White finding with respect to the CCW system. The NRC performed an inspection in June 2010 which identified that some additional extent of condition reviews are needed to ensure no additional HELB interactions exist for the CCW system. Once these reviews are completed, a follow-up inspection will be scheduled later in 2010 to close the CCW/HELB White Finding, which may return Unit 2 to the normal Licensee Response Column level. Id. at A58.

95. Under the Action Matrix of the NRC's Reactor Oversight Process, if a plant has no more than one White input in any cornerstone and no more than two White inputs in any strategic performance area during a review cycle, it is placed in the Regulatory

Response column, which signifies that the NRC will increase the regulatory attention given to that plant. This increased regulatory attention typically involves a public meeting to discuss the findings and a supplemental NRC inspection. Being in the Regulatory Response column does not signify that a plant is unsafe. At any given time, there are a number (currently over a dozen) of operating units in the Regulatory Response column. Id. at A59.

(2) Cross-cutting Issues

96. There are certain fundamental attributes of an operating plant licensee's performance that cut across all of the NRC reactor oversight process cornerstones of safety. These cross-cutting attributes are human performance, problem identification and resolution, and safety conscious work environment. Northard Dir. at A60.

97. Certain components of safety culture are directly related to one or more of the cross-cutting attributes. These are: Corrective Action Program; decision-making; environment for raising concerns; operating experience; preventing, detecting, and mitigating perceptions of retaliation; resources; self and independent assessments; work control; and work practices. In turn, issues relating to these cross-cutting attributes can be characterized as "substantive cross-cutting issues" if they become recurring aspects of a licensee's performance. If the agency identifies four or more inspection findings for a facility with the same cause in one year and the cause relates to a cross-cutting attribute, the agency determines that a substantive cross-cutting issue exists. Id. at A61.

98. The NRC evaluates whether a substantive cross-cutting issue exists at each operating reactor twice a year. If the NRC determines that a substantive cross-cutting issue exists at a given plant, assessment letters are issued by the NRC summarizing the specific substantive cross-cutting issues and the actions that should be taken to resolve them. The next mid-cycle or annual assessment letter will either state that the issue has been satisfactorily resolved or summarize the agency's assessment and licensee's progress in addressing the issue. While the NRC may conduct meetings with the licensee to ensure that the issue is being properly

addressed, no specific NRC enforcement action results from the identification of a substantive cross-cutting issue. Id. at A62 and A63.

99. Identification of a substantive cross-cutting issue does not mean that the plant is unsafe, but rather that there is a potential performance trend that deserves attention. Id. at A64.

100. In the mid-2009 performance review report for PINGP, the NRC noted that it had identified a substantive cross-cutting issue ("SCCI") in the area of human performance ("HU") with cross-cutting themes in the aspects of systematic process, conservative assumptions, procedural adequacy, and procedural compliance. The NRC determined that there were 25 findings in the previous 4 calendar quarters documented with cross-cutting aspects in the HU area, and indicated that the SCCI would remain open until all HU cross-cutting themes have been satisfactorily addressed. Id. at A66 and Northard Exhibit 21 (NSP000039).

101. To address the identification of this SCCI, NSPM took actions to improve human performance through a Target Zero Human Performance Improvement Plan and increased measures for management's assessment of the plant staff's performance. The Target Zero Human Performance Improvement Plan successfully reversed the negative trend that existed on Human Performance-related events. It included actions in the areas of Human Performance Fundamentals, Risk Management, Effective Solutions, Management Engagement and Oversight, and Behaviors. In addition, NSPM developed a Performance Recovery Plan and established a Recovery Team to implement the plan. The Performance Recovery Plan implemented Human Performance improvement initiatives in the areas of Systematic Processes, Conservative Assumptions, Procedural Adequacy and Procedural Compliance. This effort is yielding further improvements in the Human Performance area, as evidenced by the reduction in the number of Human Performance-related NRC findings in 2010 as compared to the number of findings in 2009. Northard Dir. at A67 and Northard Exhibits 22 (NSP000040) and 23 (NSP000041).

102. In the area of systematic processes, training on Operability and Functionality decision-making was provided to all Operations Senior Reactor Operator license-holders, Engineers, and Managers. Integrated Plant Knowledge training was established for engineering to aid in better operability recommendations to Operations. Risk Management Principles and Behaviors were also introduced to improve and develop more conservative behaviors. Finally, the nuclear industry's Principles for a Strong Nuclear Safety Culture, as published by INPO, were adopted as part of NSPM's corporate policy and are used daily at meetings to coach workers on safe and conservative plant operations. Northard Dir. at A68 and Northard Exhibit 6 (NSP000024).

103. An "Operational Decision-Making" tool has been developed to include two types of decision-making situations: Type 1, to address emergent challenges faced by operators; and Type 2, to address the larger, more significant decisions involving multiple departments. These tools employ a systematic approach to decision-making and require that all of the relevant facts be obtained and considered in the final decision. Also, the nuclear industry's Principles were publicized at PINGP, and posters of these principles were placed in the main conference rooms. Managers have been provided pocket-sized books of the principles and their attributes. The safety culture principles are emphasized every day, from the time workers walk into the "explosives monitors" in the Security Building and hear the recorded messages, in daily D-15 meetings, through required safety moments at meetings, in weekly Leadership Alignment meetings, as well as in Pre-Job Briefs and other daily interactions. Northard Dir. at A69 and Northard Exhibit 24 (NSP000042).

104. A major focus has been placed on use of the "STOP When Unsure" HU tool. Workers who apply the STOP tool and involve their supervisor in decision-making are formally recognized, to encourage such behavior. A monthly Employee Recognition Luncheon is held where the senior leadership team, including the Site Vice President and the Chief Nuclear Officer, recognize employees for their behaviors among other positive achievements. Additionally, NSPM recognizes employees weekly for situations where risk was recognized, avoided and

documented in the corrective action process through the site's "Good Catch" program and the Risk Prevention/Mitigation program. Northard Dir. at A69 and Northard Exhibits 25 (NSP000043), 26 (NSP000044) and 27 (NSP000045).

105. Regarding the resources and documentation aspect of human performance, new standards governing the quality of plant procedures and work packages have been adopted. These include prescribed templates for the development of work packages, a thorough and rigorous review process before procedures and work orders are finalized, and specific mitigating measures identified for medium- and high-risk work activities. Procedures associated with the specific findings in this area were also revised to reduce steps and instructions likely to result in errors. Northard Dir. at A70.

106. To improve performance with respect to work practices and procedural compliance, expectations have been defined and communicated to all site workers on procedure use and adherence. Critical steps (i.e., those that are irreversible and consequential) are discussed in pre-job briefings, and the specific HU tool(s) for the critical steps are identified and agreed to by the workers to prevent errors. Supervisors have been provided stamps to mark the critical steps in procedures. Procedure levels of usage have been reviewed and several procedures have been adjusted, where appropriate. Id. at A71.

107. PINGP uses performance indicator data to measure the status of and improvements in human performance. Metrics to measure human performance effectiveness include site and department human performance clock reset rate, percent of work order tasks screened for risk, number of significant and noteworthy events per month, number of critical observations per month, and number of risk situations prevented per month. These indicators show that, overall, PINGP's human performance is improving. Id. at A72.

108. In particular, the plant runs a "clock" that registers the occurrence of a significant human performance error, as defined through specified criteria such as worker injuries, reactivity changes, and loss of foreign material control. When such a deficiency occurs, the clock is reset to zero days. The longer the plant operates

between clock resets, the better the human performance can be said to be. The average time between clock resets significantly improved at the end of 2009 (>90 days) as compared to that at the beginning of 2009 (<30 days). The most recent time between clock resets has been registered as 65 days. Id.

109. Also improved (that is, decreased) are the numbers of lost/restricted injuries and OSHA-recordable injuries. For example, in 2010, there have been no Lost or Restricted injuries. In 2009, there was 1 Lost Time injury and 1 Restricted case injury. In 2008, there were 2 Lost Time and 4 Restricted case injuries. Similarly, OSHA Recordable injuries have decreased from 15 in 2008 to 9 in 2009 and 5 so far in 2010. Additionally, the Components out of Position index value improved from a value of 80 at the end of 2008 to a value of 90.5 at the end of 2009. (This index is measured as 100 minus the number of instances of out of position indicators, so the higher the index, the fewer the number of instances and the better the plant's performance.) Id.

110. Another measure of improved human performance is the number of NRC findings issued with HU cross-cutting aspects. In 2008, there were 12 findings with HU crosscutting aspects, 26 in 2009 (14 in the first half of the year), and 5 in the first half of 2010. Id.

111. In its end-of-year performance assessment for 2009, the NRC referred to the actions that NSPM had described in a December 1, 2009 public meeting to address the ongoing HU SCCI and noted that some improvement has been observed, but concluded that these actions have not yet proven effective in mitigating the cross-cutting themes. Id. at A73.

112. NSPM responded to the NRC's end-of-year performance assessment by holding a number of meetings to discuss human performance issues. A Human Performance Exposition ("EXPO") was held for all site employees, including contractors, prior to Unit 1 Cycle 26 Refueling Outage (1R26). The EXPO included many booths, staffed by plant employees, where the use of the HU tools was explained and reinforced, teaching employees on the use of the tools and how to apply them in the field. Dynamic Learning Activities were also developed to reinforce correct

behaviors and a case study of a large refinery accident where multiple HU barriers were broken was reviewed. The EXPO concluded with a senior manager discussion of the significant takeaways from the day's activities. Id. at A74.

113. Along the same lines, Site All-Hands meetings were conducted in February and March 2010 where the station's performance was compared to industry performance. A major focus of the meeting was on human performance improvement, with a strong employee emphasis on accountability, coaching and behaviors; use of Human Performance tools; risk management principles and behaviors; and procedure use and adherence. Monthly department meetings were started in March 2010 to improve communication of site performance and department improvement focus areas. A common message has been and will be promulgated at each department meeting that emphasizes that the site's number one performance objective is to improve human performance. This message is also conveyed through postings of supervisors and individual contributors using human performance tools. This is another step taken to shift the responsibility for use of error reduction tools down to the worker level. As evidenced by an increased number of disciplinary cases, personnel who choose not to use and/or enforce the use of error reduction practices face adverse consequences. Id.

114. At the managerial level, continuing leadership training is being provided, focusing on reviews of the coaching tools. Managers then are required to conduct a number of "Coach the Coach" observations. Id.

115. NSPM is committed to continue providing the attention and resources needed to further reduce the number of occurrences and significance of human performance-related events. The successful implementation of this commitment is evidenced by the improving trend in the number of NRC violations and the increase in the number of days between site clock resets, as well as the other metrics. Id. at A74 and A75.

116. PIIC identifies the existence of a substantive cross-cutting issue in the area of human performance as indicative of a weak safety culture at PINGP. However, the actions taken by NSPM to address the human performance findings have been effective at reducing both the severity and frequency of the human performance-

related events. Because the number of human performance-related NRC findings has dropped below three in any one aspect area, it is expected that the NRC will at a future date close the current Substantive Crosscutting Issue in Human Performance. Id. at A97.

(3) Corrective Action Program

117. The PINGP CAP is designed to meet the requirements of 10 CFR 50 Appendix B Criteria XV and XVI and applicable NRC and industry guidance (NRC Standard Review Plan (NUREG 0800) Section 17.3; Regulatory Guide 1.33, Revision 2; and ASME NQA-1, 1994), as set forth in the NSPM Quality Assurance Topical Report. Through the execution of specific procedures, NSPM has established a process for documenting and tracking the resolution of issues. The framework instituted through this process provides reasonable assurance that potential deviations from performance expectations, including conditions adverse to quality, employee concerns, operability issues, functionality issues and potentially reportable conditions are promptly identified, evaluated and corrected as appropriate. Id. at A76 and Northard Exhibits 26 (NSP000044) and 29 (NSP000047).

118. A summary level sequence of the handling of an issue by the CAP is as follows: (1) Issue is identified (an Action Request [AR] is generated); (2) AR is screened for, among other attributes, Severity Level (significance of issue); Evaluation Level – to establish corrective action(s); Due Date; and Assignee; (3) Assignee completes evaluation determining "Why" the condition exists; (4) Corrective actions are defined to correct the condition; (5) Corrective actions are completed; (6) there is a Supervisory review of the AR documentation to ensure all actions are complete; and (7) closure of the issue is approved. Northard Dir. at A76 and Northard Exhibit 26 (NSP000044).

119. The AR screening is a formal meeting in which a specified quorum must be met, including the Plant Manager. One meeting attendee should hold a SRO license at the facility or be designated by the Plant Manager and must have specific knowledge of Plant Technical Specifications. This diversity provides critical inputs for the evaluation of those ARs that identify operational and risk-significant issues.

The AR program utilizes a graded approach to the evaluation of conditions, so that those issues that have greater significance receive a more rigorous evaluation. Id.

120. The overall CAP is audited by the Nuclear Oversight Department on a quarterly basis. The Nuclear Oversight Department's function is to conduct independent reviews of the execution of PINGP programs. This group has a separate reporting structure that inspects, observes, and compares program and process execution against the PINGP implementing procedures, ensuring compliance. These comparisons and assessments are documented in reports and formally delivered to senior station management. Northard Dir. at A77 and Northard Exhibit 30 (NSP000048).

121. An additional audit of the overall CAP is performed through the Focused Self-Assessment ("FSA") process. This is a formal process that includes among its participants at least one representative from outside the company; has a formal plan and checklist of investigation focus areas; requires that the plan be reviewed and approved by the senior management team prior to execution; and results in a formal report that includes areas for improvement, enhancements and strengths, that is reviewed and graded by the senior management team. Northard Dir. at A77 and Northard Exhibit 31 (NSP000049).

122. The NRC performs an inspection of the Problem Identification and Resolution ("PI&R") programs under Inspection Procedure 71152. The PI&R inspection objectives are to: 1) provide an early warning of potential performance issues, 2) help the NRC gauge supplemental response should future action matrix thresholds be crossed, 3) provide insights into whether licensees have established a safety conscious work environment, 4) allow for follow-up on previously identified issues, 5) provide additional information related to cross-cutting issues, and 6) determine whether a licensee is complying with the NRC regulations regarding corrective action programs. The PI&R inspections at PINGP focus on the CAP for the station. Northard Dir. at A77.

123. The PINGP CAP program is constantly under evaluation. The evaluations by a variety of internal and external groups are a reflection of the value NSPM places on

the CAP program. The results of these evaluations are documented in the CAP program, are analyzed, and any needed corrective actions are established. This is a never-ending process that reflects NSPM's desire to constantly improve the CAP's performance. Id.

124. As part of the Focused Self-Assessment process, NSPM conducted an internal review of the effectiveness of the Corrective Action Program at PINGP in late January 2009. From that evaluation, several Areas for Improvement ("AFIs") were identified: (1) There was a lack of effective issue evaluation, such that issues were repeated; (2) indicators were not providing management with an accurate picture of CAP health; (3) corrective actions were not generated for all causal factors and some actions were not logically tied to any causal factor; (4) implementing procedures were not always followed; and (5) the root cause template did not address all of the requirements found in the NRC Inspection Procedure 95002. Id. at A78 and Northard Exhibit 32 (NSP000050).

125. Upon the identification of the above AFIs for the CAP, an Action Request was generated. This AR led to the initiation of a Root Cause Evaluation in order to establish the causal factors and develop corrective actions to address the identified causal factors. RCE01166830-01, "SCAQ-Inadequate CAP Resolution of Significant Issues" (January 26, 2009) ("CAP RCE") was then issued. Northard Dir. at A78 and Northard Exhibits 33 (NSP000051) and 34 (NSP000052).

126. The RCE process includes the development of a formal charter approved by the screening team. It uses a team approach, requiring at least one team member with formal RCE training. In the case of the CAP RCE, an independent industry expert was also added to the team to ensure independent analysis was performed during this investigation. The RCE process is designed to identify one (or more) root cause(s) that, when corrected, will eliminate the condition found. The RCE process also identifies contributing causes, that is, other causes that may directly or indirectly impact, but not cause, the condition found. The RCE final report is submitted to the Performance Assessment Review Board for grading and approval. Northard Dir. at A78 and Northard Exhibit 35 (NSP000053).

127. The CAP RCE determined the root cause of the AFIs was: "Management has failed to consistently enforce quality standards and set work priorities based upon procedural requirement and risk/benefit to the plant." Other contributing causes were also identified. Northard Dir. at A79 and Northard Exhibit 34 (NSP000052).

128. The CAP RCE identified two main corrective actions to prevent recurrence of the conditions reported in the AFIs. They were: (1) Develop and implement a CAP priority matrix designed to interface with work management processes and engineering work management system, and (2) Develop and implement a department CAP health indicator. In addition, the RCE recommended several other actions: (3) Develop and implement a Site CAP resolution quality and timeliness Key Performance Indicator; (4) Establish management expectations and accountability for CAP process implementation and timeliness of resolution; (5) Revise the CAP procedure, FP-PA-ARP-01, to address identified issues and enhancements; (6) Provide Root Cause Evaluation refresher training to all qualified RCE personnel; and (7) Complete successive Focused Self-Assessments on CAP process effectiveness in early 2010 and then at the end of the year. Northard Dir. at A80 and Northard Exhibit 34 (NSP000052).

129. All corrective actions recommended in the CAP RCE have been completed. To ensure that these corrective actions were properly taken, NSPM will perform an Effectiveness Review, a formal assessment of the results of a particular set of corrective actions completed, progress made, and whether sustainable improvements have been made with respect to those actions. The Effectiveness Review will be completed at the end of this year, and it is expected that it will show that the CAP RCE corrective actions have achieved the desired improvements. Northard Dir. at A81.

130. Implementation of the RCE recommendations has resulted in a number of performance improvements. These improvements include improved performance in the quality of causal evaluations; creation of corrective actions that are focused on correcting the identified problem; and increased management oversight of

evaluations and significant corrective actions to ensure a quality product. Id. at A82.

131. Once every two years, the NRC performs a team inspection of the Problem Identification and Resolution (PI&R) program at each operating reactor. These inspections are conducted under NRC Inspection Procedure IP 71152 and cover four areas of licensee PI&R performance: (1) the effectiveness of the licensee's corrective action program in identifying, evaluating, and correcting problems, (2) the licensee's use of operating experience information, (3) the adequacy of completed licensee audits and self-assessments, and (4) the existence of a safety conscious work environment to determine whether there are any indications of reluctance to report safety issues by licensee personnel. Id. at A83.

132. The NRC conducted such a team inspection at PINGP in August 2009. The results of the inspection were presented in a September 25, 2009 Inspection Report, IR 05000282/2009009; 05000306/2009009. The NRC is scheduled for another PI&R inspection in September 2010 as part of its inspection processes. Id. and Northard Exhibit 36 (NSP000054).

133. In its report on the inspection, the NRC concluded that "in general, problems were properly identified, evaluated, and corrected." The report also concluded that: (1) the licensee had a low threshold for identifying problems; (2) most items were screened and prioritized in a timely manner; (3) most issues were properly evaluated commensurate with their safety significance; (4) corrective actions were generally implemented in a timely manner; (5) audits and self assessments were determined to be performed at an appropriate level to identify deficiencies, but the station was not taking full advantage of the processes and results; and (6) workers at the site were willing to enter safety concerns into the CAP. Northard Dir. at A84 and Northard Exhibit 36 (NSP000054).

134. The inspection report also identified some concerns along with the favorable conclusions: (1) implementation was lacking in rigor resulting in inconsistent and undesirable results; (2) some significant issues went unrecognized and therefore CAPs were not issued for these; (3) there was inconsistency and lack of rigor in the

screening process; and (4) the inspectors identified significant examples of issues with evaluation and corrective action shortcomings. Id.

135. These concerns had been previously recognized by NSPM and two ARs had been generated in May 2009 to address them. These were AR01183116 (Corrective Action Implementation Resolution) and AR01183117 (Thorough Evaluation of Problem Resolution). Northard Dir. at A85 and Northard Exhibits 37 (NSP000055) and 38 (NSP000056).

136. The NRC inspectors evaluated the CAP RCE and generally agreed with the issues identified in NSPM's self-assessment, which were "consistent with the conclusions of the inspectors." In fact, the NRC observations in its PI&R inspection were essentially the same as those already identified by NSPM. Northard Dir. at A86 and Northard Exhibit 36 (NSP000054).

137. The NRC inspectors also acknowledged that PINGP has implemented improvement programs and efforts toward improving the CAP since the last PI&R inspection, although recognizable improvement in most areas had not been observed. This is attributable to the fact that, at the time the inspection was performed (August 2009), implementation of the improvement programs was only in its initial stages. Id.

138. The NRC's inspection findings did not reveal any new information, since NSPM had previously identified those issues and had initiated actions to address them. Nonetheless, following the NRC inspection, PINGP conducted an internal review of all of the individual issues and associated actions in the CAP and those relating to Human Performance. This was done to provide an aggregate view of PINGP's overall performance and actions to address identified performance gaps. NSPM hired an outside expert in the review. The review focused on three elements of the CAP program: (1) thoroughly evaluating identified problems such that the resolutions address causes and extent of conditions, as necessary; (2) properly classifying, prioritizing, and evaluating for operability and reportability conditions adverse to quality; and (3) taking appropriate corrective actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity. Northard Dir. at A87.

139. The review identified a significant number of actions that had previously been initiated to address these three CAP performance components. A gap analysis was performed to determine if there were any gaps between NSPM's performance in these areas and what could be considered as "excellent performance." From this gap analysis, some pending corrective actions were consolidated and additional corrective actions were defined. These actions are compiled under AR01183116 and AR01183117. Id. and Northard Exhibits 37 (NSP000055) and 38 (NSP000056).

140. Actions taken in response to these two ARs included: (1) Improvement of problem statement during CAP initiation; (2) Formal vs. informal Apparent Cause training; (3) Formalizing what constitutes an effective corrective action; (4) Improving CAP screening through using risk/consequence/uncertainty considerations; and (5) Formalizing the requirement to perform AR closure review ensuring the issue(s) were resolved. Also, all CAP-related procedures for oversight and execution were reviewed to validate and changed, if necessary, to reflect upgrades and improvements identified in this review. Id.

141. All corrective actions in response to the two ARs have been completed. The completion of these efforts has resulted in a solid corrective action program consistent with industry standards. NSPM has also created a new senior level position, Recovery Manager, to manage the Recovery Plan and subsequent resolution of these issues. Northard Dir. at A88.

142. The PIIC alleges that the NRC has expressed "serious concerns" about the CAP at PINGP, and cites the NRC findings in its September 25, 2009 inspection report in support of its allegation that deficiencies in the CAP are indicative of a weak safety culture at PINGP. However, the conditions that the NRC identified in its September 25, 2009 report represented a backwards look into the CAP program. These conditions do not represent the current conditions at PINGP. The station has taken actions that demonstrate recognition of the importance of the corrective action program. Station management has invested considerable time and focus on ensuring appropriate rigor for analysis, development, and execution of corrective actions.

Individual contributors demonstrate their support by actively identifying potential issues through the CAP program. Id. at A98.

D. Independent Evaluation of Safety Culture at PINGP

143. A nuclear safety culture assessment ("NSCA") was conducted at PINGP on June 21-25, 2010 under the auspices of, and in accordance with, the process established by the Utilities Service Alliance ("USA") (a consortium of nuclear power generating stations). The NSCA was performed by a team of independent industry experts and PINGP personnel. Northard Dir. at A89.

144. The NSCA team conducted a pre-assessment written survey that was provided to all PINGP employees, based on a standard set of questions common to all assessments. The PINGP pre-assessment survey had a response rate of 88 percent, which is substantially higher than the NSCA average of approximately 65 percent. This high response rate reflects strong engagement of the work force with safety culture. Id. at A91.

145. The team selected and scheduled 62 employees for interviews, chosen at random from site organization charts. The team aimed to select interviewees from the following groups of PINGP personnel: 60 to 65 percent at the individual contributor level, 20 to 25 percent at the mid-level manager level, and the remaining 10 to 20 percent at the senior management level. The team conducted interviews on-site at PINGP in accordance with the NSCA process, posing a standard series of questions at each employee level, corresponding to the INPO Principles. The team also attended routine plant meetings and activities and recorded observations relevant to safety culture principles. Id.

146. The main conclusion of the USA 2010 NSCA was that "the PINGP nuclear safety culture supports all of the INPO *Principles for a Strong Nuclear Safety Culture* and has a healthy respect for nuclear safety. Additionally, . . . Prairie Island personnel feel that they can raise any nuclear safety concern, without fear of retaliation." Id. at A92 and Northard Exhibit 39 (NSP000057).

147. The concerns voiced by PINGP personnel during the NSCA about the effectiveness of station processes and programs appear to be driven by their desire for the station to achieve higher levels of performance. One of the major themes voiced by PINGP personnel during the assessment is a desire for increased employee communications. Employees at the individual contributor level are highly engaged with safety culture and performance issues, and they want additional information about what the leadership team is doing to further improve performance. Northard Dir. at A93.

148. The vast majority of PINGP respondents believe that safety culture has improved over the last two years, and the assessment results provide evidence that the members of the PINGP staff know and understand the nuclear safety culture principles and practices required to maintain that improvement. Id.

149. The various assessments, audits, surveys, causal evaluations and the plant's performance history show that there is a strong safety culture at PINGP. The NSPM staff has responded and addressed each specific operational challenge and occurrence where human performance was a contributing factor and completed actions to correct the condition and/or prevent recurrence. Significant improvement in human performance is indicated in the various metrics used to track organizational and individual performance, including both nuclear and industrial safety. Employees have continually shown a willingness to identify and correct performance deficiencies, and to change their behaviors as needed to improve work task execution. And, finally, a reduction in the number and significance of employee errors is continuing. Id. at A99.

~~150.~~ Each of the matters identified in PIIC's contention are individual issues that do not necessarily reflect a weak safety culture. Safety culture, at its core, embodies a collective set of characteristics and attitudes that permeate an organization. The USA safety culture assessment performed at PINGP indicates that the PINGP work force has a strong knowledge and understanding of nuclear safety, as well as a healthy respect for nuclear safety at the individual level. In addition, the vast majority of employee respondents (88 percent) believe that nuclear safety has improved over the last two years. PINGP personnel's openness to sharing perceived weaknesses and areas for station improvement reflects a low tolerance for process program and equipment deficiencies and a healthy refusal to accept the status quo. This feedback reflects the engagement of the work force and their desire to see and take part in improved plant performance. These organizational attributes exemplify

150. the type of individual engagement with and commitment to nuclear safety issues that is at the heart of a strong safety culture. Id.

Respectfully Submitted,

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Dated: July 30, 2010