UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

DIRECT TESTIMONY OF CHRISTOPHER I. GRIMES

Q1: Please state your name and current business address.

A1: My name is Christopher I. Grimes, and my current business address is 27 North Shore Drive Swanton, Maryland 21561-2215.

Q2: What is your educational background?

A2: I hold a B.S. in Nuclear Engineering from Oregon State University.

Q3: For whom do you work and in what capacity?

A3: I am currently self-employed as a consultant. The Prairie Island Indian Community in the State of Minnesota has retained me as a consultant with respect to the above-captioned proceeding. I am currently a Senior Nuclear Safety Consultant and I am a Reactor Safety Expert.

Q4: What is your professional background?

A4: A copy of my resume is attached to this testimony as Exhibit 1. I retired from the United States Nuclear Regulatory Commission (NRC) in June 2006 after 32 years of Federal Service. My career at the NRC included a broad range of positions of increasing responsibility covering most aspects of nuclear power plant regulation. At the time of my retirement, I served as the Director of the Policy and Rulemaking Division in the NRC's Office of Nuclear Reactor Regulation. In that position, I was responsible for all of the reactor-related rulemaking activities, financial assurance, regulatory analysis, generic communications, generic project management, interoffice coordination, licensing processes, and all of the licensing and inspection activities associated with research and test reactors. During the course of my career at the NRC I managed the completion of the NRC's Systematic Evaluation Program, the licensing of the Comanche Peak nuclear power plant, issuance of the improved Standard Technical Specifications and implementation of the Technical Specification Improvements Program, development and implementation of the NRC's license renewal process, and improved application of risk-informed decisions in the regulatory process. I also was trained as an Incident Investigation Team Leader, which included evaluation and interviewing techniques to assess organizational contributions to the root cause of incidents. I served as Team Leader for the Oyster Creek Nuclear Generating Station Diagnostic Evaluation Inspection Team.

Q5: Have you published or presented in the fields of nuclear reactor safety?

A5: I have authored numerous NRC documents including Safety Evaluation Reports, Environmental Impact Statements, affidavits in licensing hearings, staff technical positions, regulatory analyses, Commission papers, and routine correspondence with licensees, industry groups, public interest groups, and individuals. I have also contributed to reports on nuclear safety developed by industry standards groups and the International Atomic Energy Agency (1973-2003).

Q6: Have you testified as an expert previously in any jurisdiction or proceeding?

A6: No.

Q7: Do you have a written summary of your education, employment, experience and background, and papers and presentations you have made over your career?

A7: The copy of my resume is attached as Exhibit to this testimony supplies such a summary.

Q8: What materials have you reviewed and actions have you taken in preparation for your testimony?

A8: I am familiar with the application of Northern States Power, a Minnesota Company ("NSPM"), for a license renewal for the two nuclear reactors at the Prairie Island Nuclear Generating Plant ("PINGP"). I have reviewed excerpts of NSPM's License Renewal Application, the Draft Environmental Impact Statement ("FEIS") prepared by the staff of the NRC, and the Safety Evaluation Report ("SER") prepared by the staff of the NRC, including certain sections relating to scoping and screening, aging mechanisms, aging management programs, time-limited aging analyses, severe accident mitigation alternatives, and inspection reports, together with related documents submitted in this matter.

Q9: Have you given affidavits or declarations in support of or in connection with this license renewal proceeding?

A9: Yes, I submitted a declaration in support of the PIIC's initial petition to intervene on August 18, 2008. (ML082391038). I also submitted a declaration in support of the admission of PIIC's safety culture contention on November 23, 2009 (Exhibit 2).

Q10: What background and experience do you have in nuclear reactor safety?

A10: I am knowledgeable of and experienced in nuclear reactor safety. As a reactor engineer and systems analyst, I am familiar with a broad variety of reactor designs. I have been responsible for performing containment response analyses, evaluating reactor system designs and preparing safety evaluation reports for construction permit and operating license applications. I have contributed to the development of a computer code for analyzing containment subcompartment pressurization using compressible fluid flow theory. I have served as the Task Manager for the Mark I containment Long Term Program to resolve pool dynamic loads in boiling water reactor designs. I have served as Emergency Officer in the NRC's Incident

Response Program. I was qualified as an Incident Investigation Team Leader. I served as Team Leader for the Oyster Creek Diagnostic Evaluation Inspection Team.

Q11: What background and experience do you have in nuclear reactor safety management?

A11: I am knowledgeable of and experienced in nuclear reactor safety management. I was appointed to the senior executive service in 1984, upon selection as the Chief of the Systematic Evaluation Program (SEP). That program evaluated ten of the oldest power reactors against current requirements and used risk insights in integrated safety assessments to develop backfitting recommendations. I was responsible for directing safety reviews and developing proposed staff positions to resolve safety issues. I served as Deputy Director in the Division of Engineering from 2003 to 2005. In that position, I was responsible for directing engineering-related safety evaluations of licensees' implementation of NRC requirements, changes to existing license requirements, and applications for new facilities or designs. I was also responsible for directing the application of engineering expertise to support special inspections, projects, programs, and policy activities in the areas of mechanical, civil-structural, materials, metallurgy, chemical, instrumentation and control systems, and electrical engineering, as well as applying that engineering expertise to conduct failure analysis, structural analysis, and represented the NRC on domestic and international codes and standards groups.

Q12: What background and experience do you have in nuclear reactor license renewal and environmental impacts?

A12: I am knowledgeable of and experienced in nuclear reactor license renewal and environmental impacts. I served as Director of the License Renewal and Environmental Impacts Program from 1997 to 2002. In that position, I was responsible for developing and implementing the license renewal review process for power reactors based on the requirements which were codified in 10 CFR Part 54 in 1995. I was responsible for establishing the plans and schedules for the first license renewal reviews, as well as developing the review standards for the associated environmental reviews. Upon completion of the first three renewed licenses, I established a five-year schedule of license renewal reviews and implemented a process to manage changes to the license renewal review guides and related staff positions.

Q13: What background and experience do you have in rulemaking and regulatory analysis?

A13: I am knowledgeable of and experienced in rulemaking and regulatory analysis. I was appointed as Director of the Policy and Rulemaking Division in 2005. In that position, I was responsible for all of the reactor-related rulemaking activities, financial assurance, regulatory analysis, generic communications, generic project management, and interoffice coordination of licensing processes.

Q14: Are you familiar with the operation of, and safety analysis associate with, pressurized water nuclear reactors?

A14: I am very familiar with the operation of, and safety analyses associated with, pressurized water nuclear reactors (PWRs), including the reactor design currently in operation at the Prairie Island Nuclear Generating Plant (PINGP) located near Red Wing, Minnesota.

Q15: What are the topics of your testimony?

A15: My testimony addresses continuing safety culture weaknesses at Prairie Island Units 1 and 2, as demonstrated by a series of events, incidents and NRC enforcement actions.

Q16: Please summarize your opinion regarding whether there is reasonable assurance for the NRC to determine that the applicant will manage the effects of aging during the period of extended operation on the functionality of structure and components as required by 10 C.F.R. 54.29(a)(1).

A16: I have reviewed the NRC's Safety Evaluation Report ("SER"). In my opinion, and as I explain more fully below, there presently is not reasonable assurance for the NRC to determine that the applicant will manage the effects of aging during the period of extended operation on the functionality of structure and components as required by 10 C.F.R. 54.29(a)(1).

Q17: Your previously submitted Declaration identified the Applicant's handling of the leakage of borated water from the Unit 1 and 2 refueling as one of the reasons you believe there presently is not reasonable assurance for the NRC to determine that the applicant will manage the effects of aging during the period of extended operation on the functionality of structure and components as required by 10 C.F.R. 54.29(a)(1). Can you please describe what the leakage of borated water issue is?

A17: In the NRC license renewal inspection and audit of PINGP in the fall of 2008, the staff "noticed" that PINGP had identified the leakage of borated water from the Unit 1 and 2 refueling. *See* Safety Evaluation Report, Related to the License Renewal of Prairie Island Nuclear Generating Plant Units 1 and 2, U.S. Nuclear Regulatory Commission, at 3-142 (October 2009). The NRC staff closed this issue by requiring additional applicant commitments for visual and other types of inspections and sampling programs in subsequent refueling outages. *See id*. at Appendix A, Items 41, 42, 43 and 44.

Q18: Although the NRC staff required additional applicant commitments regarding this issue, do you believe that the Applicant's performance calls into question the Applicant's ability to implement the aging management program during the period of extended operation?

A18: Yes.

Q19: Why?

A19: While these additional commitments, if followed by the Applicant, may provide assurance that no further damage to the containment vessel will result, Applicant's deficient performance and dereliction of its obligations to promptly and effectively correct deficient conditions call into question the Applicant's ability to effectively implement the aging management program during the period of extended operation. As noted by the Applicant's expert at the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee meeting, PINGP has experienced intermittent refueling cavity leakage since the late 1980s. *See* Transcript, Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee, Prairie Island Nuclear Generating Station (ACRS Transcript), July 7, 2009, at 48. However, the applicant's expert also noted that although there was not any evidence of the leak prior to 1987, they assume that leakage has been going on for the entire life of the plant. *Id*. at 72. The cumulative leak rate was estimated to be approximately one to two gallons per hour. Although, the Applicant had tried to fix this leak several times, its efforts had not been successful. *Id*. at 57. In fact, after twenty years of leakage, the Applicant still had not identified the exact source of the leak. *Id*. at 69. The potential hazard of this leakage is that the borated water appears to be settling at the bottom of the containment liner, posing a danger to the integrity of the containment.

The SER describes three specific staff concerns related to the refueling cavity leakage: (1) the leaking borated water may contact the containment vessel and remain in contact with the vessel between outages, (2) the leaking borated water may contact the concrete reinforcement and cause degradation, and (3) the leaking borated water may react with the concrete and cause degradation. *See* Section 3.0.3.2.17, "Structures Monitoring," at 3-148.

The allowable containment leakage for a design basis accident is equivalent to a 0.003 square-inch hole in the containment (about one-sixteenth of an inch in diameter). *See* Inspection

Manual Chapter 0609, Containment Integrity Significance Determination Process, Appendix H, U.S. Nuclear Regulatory Commission (May 6, 2004). If the leakage from the refueling cavity stays in contact with the steel liner and concrete structure for an extended period, corrosion could eat through the containment liner and weaken the concrete structure to such an extent that, should an accident occur, the containment leakage could result in radiological exposures in excess of 10 CFR Part 100.

The statements from ACRS consultants and ACRS members at the Subcommittee meeting captured the issue of concern to the Community in terms of the performance of the applicant. For example, ACRS Consultant John Barton asked, "[t]his thing has gone on for so long. Why now do you decide you're going to fix it?" ACRS Transcript at 64. The applicant's Site Vice-President replied, "It's not acceptable to continue to have this leak. Too many unknowns." *Id*. at 65. Furthermore, as ACRS Member Said Abdel-Khalik stated, "[y]et this has been going on for more than 20 years. Is this sort of a new management attitude?" *Id*. at 75. It is exactly this "attitude" that calls into question the applicant's ability to carry out its aging management program. These leaks could have potentially disastrous consequences for the Community and the populace of the surrounding area. Yet, the Applicant did not acknowledge the importance of these problems to aging management until the NRC audit in the Fall of 2008 – years after the Applicant began efforts to address the problem. The implications of this type of dereliction are further underscored by the enforcement history of the Applicant, discussed more fully below.

Q20: How is the effectiveness of an aging management program measured?

A20: SER Section 3.0.3.2.17 describes how the Structures Monitoring Program, as enhanced, addresses the elements of an effective aging management program. The SER states that: "The staff found that the program identifies 60 items as listed in Table 8.1 'Managed Aging Effects' to be monitored or inspected and linked them to the degradation of the particular SCs intended functions." The Standard Review Plan for License Renewal, NUREG-1800, Revision 1, U.S. Nuclear Regulatory Commission (2005), describes the ten elements of an effective aging management program, which include:

> 7. Corrective actions, including root cause determination and prevention of recurrence, should be timely.
>
> 8. Confirmation process should ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.
>
> 9. Administrative controls should provide a formal review and approval process.
>
> 10. Operating experience should provide objective evidence to support the conclusion that the effects of aging will be managed adequately so that the structure and component intended function(s) will be maintained during the period of extended operation.

See id., Table A.1-1. "Elements of an Aging Management Program for License Renewal."

Q21: Does the enforcement history of the Applicant you just referenced above include the performance issues raised in NRC Information Notice 2009-11?

A20: Yes.

Q22: Do you believe that the Applicant's performance issues raised in NRC Information Notice 2009-11 violate certain elements of an effective aging management program?

A22: Yes.

Q23: Why?

A23: In the NRC Information Notice 2009-11, describing the configuration control problems that resulted in the White Finding associated with EA 08-272, the NRC states that "[t]he recent events show that component mispositionings have occurred or remained undetected due to one or more of the following causal factors:

- Failure to use or establish administrative controls, including: proper component labeling, proper valve locking, use of valve checklists, work and testing procedures use of postmaintenance flow testing confirmation (when necessary)

- Dependence on a single administrative control to prevent valve mispositioning events

- Insufficient training (lack of refresher training) or experience in determination of valve position by individuals, (such as using rising stem position to help confirm valve position)

- Improper independent verification or incorrect valve locking techniques

- Lack of operator awareness of unique valve design or valve operating characteristics

- Unrecognized operator burdens that increase the likelihood of error

- Failure to effectively apply station and industry operating experience."

See Information Notice 2009-11, U.S. Nuclear Regulatory Commission, at 2.

The conclusions in the NRC Information Notice are further evidence that there is a safety culture at Prairie Island that potentially fails to achieve four of the ten elements of an effective management program (items 7 through 10 above). It is not clear how the NRC can now conclude that there is reasonable assurance that NSPM can achieve all ten elements of effective aging management for the period of extended operation.

Q24: Are you familiar with the Reactor Oversight Process, including crosscutting issues of human performance, safety conscious work environment, and problem identification and resolution?

A24: Yes, I am familiar with the Reactor Oversight Process including crosscutting issues of human performance, safety conscious work environment, and problem identification and resolution. I am also familiar with the underlying safety culture components of a safety-conscious work environment.

Q25: Please describe the NRC regulatory framework for inspecting the safety of operating reactors.

A25: The NRC Regulatory Oversight Process (ROP) is the NRC regulatory framework for inspecting the safety of operating reactors. *See* Reactor Oversight Process, NUREG-1649, Rev. 4, U.S. Nuclear Regulatory Commission (December 2006). The Operating Reactor Assessment Program evaluates the overall safety performance of individual operating reactors. It consists of three key strategic performance areas: reactor safety, radiation safety, and safeguards. Within each strategic performance are cornerstones that reflect the essential safety aspects of facility operation. These seven cornerstones for reactor safety include initiating events, mitigating systems, barrier integrity, emergency preparedness, public radiation safety, occupational radiation safety, and physical protection. Satisfactory licensee performance in the cornerstones provides reasonable assurance of safe facility operation. NRC inspection findings are classified by color designations that indicate the severity of the inspection concern. "Green" designates acceptable performance. However, "white", "yellow", and "red" findings indicate more serious safety problems.

Q25: Please describe how the NRC regulatory framework evaluates "crosscutting" issues in the areas of human performance, safety conscious work environment, problem identification and resolution, and other organizational components.

A25: In addition to the three strategic performance areas, the regulatory framework also evaluates "crosscutting" issues in the areas of human performance, safety conscious work environment, problem identification and resolution, and other organizational components.

Q26: **How do these "crosscutting" issues in the areas of human performance, safety conscious work environment, problem identification and resolution, and other organizational components reflect the Applicant's safety culture?**

A26: These crosscutting issues are fundamental performance attributes that extend across all of the cornerstones. The crosscutting issues incorporate components that are important to the concept of "safety culture." Safety culture is defined as the assembly of characteristics and attitudes in organizations and individuals who establish that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance. *See* Inspection Manual Chapter 0305, Operating Reactor Assessment Program, U.S. Nuclear Regulatory Program, at 04.16 (August 11, 2009). A weak licensee safety culture was identified as a root cause of the reactor vessel head degradation at the Davis-Besse nuclear power plant. *See generally*, Information on the Changes Made to the Reactor Oversight Process to More Fully Address Safety Culture, Regulatory Issue Summary 2006-13, U.S. Nuclear Regulatory Commission (July 31, 2006). The components of safety culture are directly related to the crosscutting areas of human performance, a safety conscious work environment, and problem identification and resolution.

Q27: Does the enforcement history of the Applicant you referenced above include the performance issues that have resulted in PINGP Unit 1 and Unit 2 being placed in the "Regulatory Response" column of the NRC Action Matrix?

A27: Yes.

Q28: What does it mean to be placed in the "Regulatory Response" column of the NRC Action Matrix?

A28: Both PINGP Unit 1 and Unit 2 have been placed in the "Regulatory Response" column of the NRC Action Matrix in 2009. For plants in the Regulatory Response column, the NRC conducts additional inspections beyond the normal inspection program, and takes other actions, to focus on potential safety problems (approximately 10 to 20% of all operating reactors are in the Regulatory Response column).

Q28: Why was PINGP Unit 1 placed in the Regulatory Response column?

A28: PINGP Unit 1 was in the Regulatory Response column because of a "White" finding in the Mitigating Systems cornerstone and a White finding in the Public Radiation Safety cornerstone. The White finding in the Mitigating Systems cornerstone from the fourth quarter of 2008 involved the failure of the PINGP staff to adequately control the position of a normally open valve necessary for a turbine driven auxiliary feedwater pump to operate. *See* Letter from James L. Caldwell, Regional Administrator, U.S. Nuclear Regulatory Commission to Michael D. Wadley, Site Vice President, Prairie Island Nuclear Generating Plant in regard to EA-08-272 (January 27, 2009). The other White finding involved a radioactive material shipment from PINGP that did not conform to applicable regulations. This White finding also applied to Unit 2. *See* Letter from James L. Caldwell, Regional Administrator, U.S. Nuclear Regulatory Commission to Michael D. Wadley, Site Vice President, Prairie Island Nuclear Generating Plant in regard to EA-08-349 (February 10, 2009). In this case, the NRC found that when the package containing radioactive material from PINGP arrived at its destination, the radiation levels were five times higher than allowed by NRC and Department of Transportation limits. This resulted from "ineffective" packaging and the fact that the PINGP personnel who prepared the package were not properly trained and qualified.

Q29: Has the NRC also issued White findings for PINGP Unit 2?

A29: Yes.

Q30: Please describe the White findings for PINGP Unit 2.

A30: In EA-09-167, dated September 3, 2009 (05000306/2009013) the NRC issued a White finding for PINGP Unit 2 in the Mitigating Systems Cornerstone. *See* Letter from Mark A. Satorius, Regional Administrator, U.S. Nuclear Regulatory Commission to Mark A. Schimmel, Acting Site Vice President, Prairie Island nuclear Plant (ML092450624). This finding related to the Applicant's failure to design the component cooling water system ("CCWS") such that it would be protected from a high-energy line break ("HELB"), or seismic or tornado events. A high energy line break could result in flooding effects which could lead to the failure of redundant safety systems. These design deficiencies could result in a loss of safety functions because of inadequate protection from flooding and dynamic effects for high energy line breaks associated with certain design basis events. The confirmation of this White finding for the inadequate design of the CCWS kept Unit 2 in the Regulatory Response column of the ROP Action Matrix for 2010. See March 3, 2010 Annual Assessment Letter from Stephen West, Director of Reactor Projects, U.S. Nuclear Regulatory Commission to Mark A. Schimmel, Site Vice President, Prairie Island Nuclear Generating Plant (ML100610286). It also resulted in an NRC enforcement action against the applicant for violating 10 C.F.R. Part 50, Appendix B, Criterion III. Because of this White finding, Unit 2 remains in the Regulatory Response column for 2010.

Q31: Do you believe that the Applicant's performance issues associated with this White finding call into question the Applicant's ability to implement the aging management program during the period of extended operation?

A31: Yes.

Q32: Why?

A32: As with the refueling cavity leakage issue, the Applicant knew of the HELB issue for many years but limited its evaluation of the issue to the auxiliary building and missed the turbine building.

Q33: What is the significance of the performance issues associated with all of the recent White findings?

A33: All of the above White findings are associated with violations of the NRC requirements and consideration of escalated enforcement action by the NRC. During the Mid-cycle review for PINGP in 2009, the NRC identified an open substantive crosscutting issue in the area of human performance, with crosscutting themes in the aspects of systematic process, conservative assumptions, procedural adequacy, and procedural compliance. See Letter of from K. Stephen West, Director of Reactor Projects, U.S. Nuclear Regulatory Commission to Mark A. Schimmel, Site Vice President (Acting), Prairie Island Nuclear Generating Plant, September 1, 2009 (ML092440367).

Q34: How do the performance issues associated with all of the recent White findings relate to safety culture?

A34: These human performance crosscutting issues are one aspect of the NRC's assessment of the "safety culture" at an operating reactor. In addition to the human performance crosscutting issue identified in relation to the PINGP White findings, a fundamental characteristic of an effective safety culture is that "[t]he organization ensures that issues potentially impacting safety or security are promptly identified, fully evaluated, and promptly addressed and corrected…." See U.S. Nuclear Regulatory Commission, Draft Policy Statement on Safety Culture, 74 Fed. Reg. 57525, 57528, (November 6, 2009). This is directly relevant to the refueling cavity issue, specifically, as well as the Applicant's ability to resolve other potentially risk-significant deficiencies associated with long-term, age-related degradation.

Q35: The December 21, 2007 Problem Identification and Resolution ("PI&R") Inspection Report raised another aspect of human performance at the PINGP. Can you please explain what was discussed in that PI&R Inspection Report?

A35: Yes. The December 21, 2007 Problem Identification and Resolution ("PI&R") Inspection Report noted a common theme during the last four PI&R inspection reports was that the licensee tended to focus on detecting problems rather than preventing problems. The inspection report goes on to explain that "the team did acknowledge that the station's performance had improved since the 2005 PI&R inspection; but, the improvement has been at a slow rate." The inspection report describes how operating experience is being better utilized and how plant personnel had properly implemented the employee concerns program.

Q36: How does this relate to Applicant's safety culture?

A36: In August 2008, the United Service Alliance conducted a safety culture assessment which found that a culture of prevention has not been embraced (Executive Summary, item 4) and that there is a perception that the plant is challenged with problem-solving. *See* Xcel Energy, Prairie Island Nuclear Generating Plant, Nuclear Safety Culture Assessment, United Service Alliance ("USA"), August 25-29, 2008 (PROD00006391). Plant employees interviewed as part of the assessment indicated that they do not have time to be proactive and as a result always seem to be in the reactive mode. Being in the reactive mode prevents focusing on backlog, improving cumbersome processes, and monitoring low level indicators to identify precursors before they reveal themselves as events. As noted by the USA "self-assessment" team, "prevention" is an item that provides a foundation for much of nuclear safety culture.

Q37: Do you think the Applicant is aware of the performance deficiencies?

A37: Yes.

Q38: Why?

A38: NSPM held a "Stand-Down" on January 5, 2009. The "Required Briefing by Department Managers" (PROD00003603) describes the reasons for the stand-down which include the independent assessments in 2008 for configuration management, reactivity management, the INPO mid-cycle assessment, organizational effectiveness, the Management Safety Review Committee, and an independent human performance assessment. The messages to be delivered by the department managers include the weaknesses described in the August 2008 USA Self Assessment which include the lack of confidence in station processes and a culture of prevention has not been fully embraced. The messages also include a description of a regulatory perception: "Our behavior does not demonstrate that we understand the significance or the uniqueness of nuclear power, nor do we appear to respect the power of the reactor."

Q39: In your opinion, despite Applicant's awareness of the performance deficiencies, has Applicant adequately addressed this area of deficient performance?

A39: No.

Q40: Why not?

A40: In the September 25, 2009 Biennial Problem Identification and Resolution ("PI&R") Inspection Report (EA 06-178, ML092680208), the NRC noted that the Corrective Action

Program ("CAP") at PINGP was "functional" but implementation was lacking in rigor resulting in inconsistent and undesirable results. The CAP is the program the applicant relies upon to effectively accomplish the Problem Identification and Resolution expectations of the Reactor Oversight Process. *See* Inspection Report 05000282; 05000306/2009-009, U.S. Nuclear Regulatory Commission, at 1 (July 20, 2009 to August 13, 2009). Furthermore, "[s]ignificant issues went unrecognized." *Id*. The inspectors also noted that "the backlog of corrective actions was large and growing."

Q41: Do you think it is reasonable for the NRC to conclude that the Applicant has adequately addressed this area of deficient performance?

A41: No.

Q42: Why not?

A42: The NRC's Annual Assessment Letter for Prairie Island, dated March 3, 2010 (ML100610286) states that Prairie Island fully met all of the cornerstone objectives and explains how the White findings have been or will be closed with supplemental inspections. However, the assessment also states:

> [d]uring this assessment period, there were 26 findings documented with cross-cutting aspects in the HP [human performance] area. Of these 26 findings, four shared a common cross-cutting aspect of systematic process, four findings with conservative assumptions, six findings with procedural adequacy, and seven findings with procedural compliance. The number of findings in the SCCI themes increased during this assessment period in two out of the four themes. You discussed your actions to address the ongoing HP SCCI during a December 1, 2009, public meeting and some improvement has been observed. However, these actions have not yet proven effective in mitigating the cross-cutting themes. Therefore, the NRC has a concern with your progress in addressing this cross-cutting area and has concluded that the SCCI will remain open until all cross-cutting themes have been cleared in the area of HP. To clear each of the themes, we will assess whether your corrective actions have resulted in a positive sustainable improvement in the area and will consider the number of findings in the theme.

Thus, NSPM has yet to demonstrate to the NRC inspectors that the human performance weaknesses have been corrected.

In a recent meeting of the PINGP Management Safety Review Committee, the Committee concluded that "[m]oving Prairie Island solidly forward with the large scope of work on its plate will be determined by the strength and consistency of Station leadership. The leadership team – senior executives through first line supervisors – must continue stepping up the level of engagement with the workforce. Much of what ails Prairie Island is deeply imbedded in its culture. Actions taken at both site and Fleet levels to strengthen the leadership team are vital and will be followed closely by the Committee." *See* Xcel Management Safety Review Committee Meeting Summary Meeting No. 2010-01, MSRC Meeting Date March 17 and 18, 2010, at Summary (PROD00000267).

Q43: In your opinion, is there reasonable assurance for the NRC to determine that the applicant will manage the effects of aging during the period of extended operation on the functionality of structure and components as required by 10 C.F.R. 54.29(a)(1).

A43: No.

Q44: Why not?

A44: There is a pattern of cultural performance issues revealed by the continuing human performance (HP) and problem identification and resolution (P&IR) issues at Prairie Island that go too deep to be addresses by a simple follow-up inspection. As described in the NRC's most recent Annual Assessment Letter, the Applicant must demonstrate that the cultural corrective measures are both effective and sustainable.

The failure of the applicant to correct the potential damage to the containment integrity resulting from the refueling cavity leaks, the safety culture weaknesses associated with the causal factors described in Information Notice 2009-11, the series of White findings associated with one or both of the PINGP units, the identification of substantive crosscutting issues in the area of human performance, the serious concerns identified by NRC inspectors with the applicant's corrective action program, and failure to effectively manage the plant design and effectively resolve potentially the safety-significant flooding issues identified 20 years ago, are all indicative of a weak safety culture at PINGP. It is not clear how, or indeed, whether, the NRC factored the license renewal these inspection findings on the refueling cavity leakage issue into the license renewal findings. Consequently, in my opinion the NRC cannot logically find that there is reasonable assurance the aging management programs can be effectively implemented under the requirements of 10 CFR 54.29(a)(1).

Q45: In your opinion, what is necessary for there to be reasonable assurance for the NRC to determine that the applicant will manage the effects of aging during the period of extended operation on the functionality of structure and components as required by 10 C.F.R. 54.29(a)(1)?

A45: The NRC should direct the applicant to conduct a third party assessment of safety culture as described in Section 10.02 of NRC Inspection Manual Chapter 0305. After the review of this third party assessment, the NRC can address what corrective actions by the applicant are necessary before the renewal should be granted.

Q46: Do you still agree with the statements and opinions you expressed in your declaration dated November 23, 2009 that was filed in support PIIC's safety culture contention (CIG0002)?

A46: Yes.

Q47: Does this conclude your testimony?

A47: Yes.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 30th day of July, 2010, at Rockville, Maryland.

/Executed by Christopher I. Grimes in Accord with 10 C.F.R. 2.304(d)/

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