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
	)	

DECLARATION OF CHRISTOPHER I. GRIMES

1. My name is Christopher I. Grimes. I am currently a Senior Nuclear Safety 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 hold a B.S. in Nuclear Engineering from Oregon State University.

2. 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 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.

3. 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.

4. 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 analysis, 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.

5. 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 the safety reviews and developing proposed staff positions to resolve the 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.

6. 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.

7. 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, interoffice coordination, licensing processes, and all of the licensing and inspection activities associated with research and test reactors.

8. 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.

9. I have reviewed the NRC's Safety Evaluation Report ("SER"). In my opinion, and as I explain more fully below, the Atomic Safety and Licensing Board should grant a hearing to determine 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).

10. 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, and 43. 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 wasn't 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 has tried to fix this leak several times, their efforts have not been successful. *Id*. at 57. In fact, after twenty years of leakage, the applicant still cannot identify 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.

11. 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.

12. 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.

13. 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.

14. The August 20, 2009, the NRC Mid-Cycle Performance Review and Inspection Plan for PINGP Units 1 and 2 summarized the performance of PINGP for the period from July 1, 2008 through June 30, 2009. *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. PINGP Units 1 and 2 have been placed into the "Regulatory Response" column of the NRC Reactor Oversight Process Action Matrix. *See* Third Quarter 2009 ROP Action Matrix Summary and Current Regulatory Oversight, U.S. Nuclear Regulatory Commission (2009). The Regulatory Response column identifies plants that are designated for heightened inspection oversight by the NRC because of inspection findings and violations of NRC regulations.

15. 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.

16. 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. 3, U.S. Nuclear Regulatory Commission (July 2000). 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 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.

17. 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, and problem identification and resolution. 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.

18. As noted above, both PINGP Unit 1 and Unit 2 have been placed in the "Regulatory Response" column of the NRC Action Matrix. 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).

19. PINGP Unit 1 is 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.

20. In addition, the NRC is also investigating a preliminary White finding for Unit 2 on the inadequate design of the component cooling water system to ensure that it was protected from high energy line breaks, seismic, or tornado events. *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 in regard to EA-09-167 (September 3, 2009). These events could cause a loss of cooling water and a loss of safety function.

21. All of the above White findings are associated with violations of the NRC requirements and consideration of escalated enforcement action by the NRC. The NRC has also 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. 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. In the NRC Inspection report of PINGP for the period of July 20, 2009 to August 13, 2009, the NRC inspectors continued to have concerns with the applicant's Corrective Action Program (CAP). The CAP is the program the applicant relies upon to effectively accomplish the Problem Identification and Resolution expectations of the Reactor Oversight Process. The NRC inspection team concluded that the applicant's CAP was "functional," but

implementation was lacking in rigor, resulting in inconsistent and undesirable results. *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.*

22. SER Section 3.0.3.2.17 describes the safety-related structures, the applicable aging effects and how the Structures Monitoring Program 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."

23. However, 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.

24. 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.

25. The conclusions in the above Information Notice, the White findings discussed above in regard to PINGP, the identification of a substantive crosscutting issue in the area of human performance, the serious concerns identified by NRC inspectors with the applicant's CAP, and the failure of the applicant to correct the potential damage

to the containment integrity resulting from the refueling cavity leaks, including the failure to effectively correct the safety-significant deficiency for a period of 20 years, are all indicative of a weak safety culture at PINGP. It is not clear how, or indeed, whether, the NRC factored the license renewal inspection findings on the refueling cavity leakage issue into the license renewal findings. Consequently, in my opinion the NRC cannot legitimately find that there is reasonable assurance under the requirements of 10 CFR 54.29(a)(1). 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.

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

Executed this 23rd day of November, 2009, at Rockville, Maryland.

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

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