August 12, 2010

**UNITED STATES OF AMERICA**
**NUCLEAR REGULATORY COMMISSION**

Before the Atomic Safety and Licensing Board

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| 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) | ) | |

# REBUTTAL TESTIMONY OF STEVEN C. SKOYEN ON SAFETY CULTURE CONTENTION

## I. WITNESS BACKGROUND

**Q1.** Please state your full name.

**A1.** My name is Steven C. Skoyen.

**Q2.** By whom are you employed and what is your position?

**A2.** I am employed by Northern States Power Company, a Minnesota corporation ("NSPM") as engineering programs manager at the Prairie Island Nuclear Generating Plant ("PINGP").

**Q3.** Have you previously provided written testimony in this proceeding?

**A3.** Yes. On July 29, 2010 I provided direct written testimony entitled "Testimony of Steven C. Skoyen on Safety Culture Contention" ("Skoyen Testimony").

**Q4.** What is the purpose of your testimony at this time?

**A4.** The purpose of my testimony is to address certain matters contained in the "Direct Testimony of Christopher I. Grimes" ("Mr. Grimes' testimony") and the "Prairie Island Indian

Community Initial Statement of Position on Safety Culture Contention" ("PIIC SOP"), both submitted by the Prairie Island Indian Community ("PIIC") in this proceeding.

## II. DISCUSSION

**Q5.** What aspects of the Mr. Grimes' testimony and the PIIC SOP will you address in your Rebuttal Testimony?

**A5.** I will address the allegations in those materials that refer to identification and correction of leakage from the refueling cavity inside the containments of both PINGP units as indicative of the existence of a weak safety culture at the plant. I will note that most of the allegations in those materials are similar, if not identical, to those contained in the "Prairie Island Indian Community's Submission of a New Contention on the NRC's Safety Evaluation Report" (Nov. 23, 2009) and the "Declaration Of Christopher I. Grimes" of the same date. I have responded to most of these allegations in the Skoyen Testimony and will not repeat here the statements made in the Skoyen Testimony but will cross-reference them as appropriate.

**Q6.** In Mr. Grimes' testimony (A19 at 7), the witness makes repeated reference to testimony presented and discussions that transpired during a meeting of the Subcommittee on License Renewal of the Advisory Committee on Reactor Safeguards ("ACRS") that took place on July 7, 2009. Did you attend that meeting?

**A6.** Yes, I attended and testified at that meeting.

**Q7.** Is the description by Mr. Grimes of what transpired at that meeting accurate?

**A7.** No. Mr. Grimes cites a number of statements out of context and provides misleading quotes from the testimony and comments from witnesses and ACRS members. In so doing, he presents an incorrect portrayal of the status of the refueling cavity leakage issue, both at the time of the meeting over a year ago and at the present time.

**Q8.** Would you please describe in what respects is Mr. Grimes' testimony incorrect or misleading?

**A8.** First, it is incorrect to assert that "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." Mr. Grimes is referring to my statement at the ACRS Subcommittee meeting in which I explained that in performing engineering evaluations of potential degradation of the steel containment vessel, the concrete, and the rebar, we assumed for calculation purposes that the leakage had occurred for the entire plant life. This theoretical assumption was made for the purpose of conservatively estimating the maximum potential degradation that may have affected these structures. As I stated at the meeting and in the Skoyen Testimony, there is no documentation of refueling cavity leakage at PINGP prior to 1987. See Skoyen Exhibit 16 (ACRS July 2009 Meeting Transcript) at 48.

Second, it is misleading to claim that although NSPM "had tried to fix this leak several times, its efforts had not been successful." As I stated at the ACRS meeting, our coating and caulking procedures had been successful in some instances of preventing leakage during refueling, including during the 2008 Unit 1 outage. Id. at 64. However, because these indirect sealing measures had inconsistent results, NSPM performed a comprehensive root cause evaluation from 2008-2009, the results of which enabled us to isolate and repair the ultimate leakage sources. As I stated in the Skoyen Testimony at A25, PINGP has, at present, eliminated 95-97 percent of previous leakage in Unit 1 and 97.5 percent of previous leakage in Unit 2.

Third, the allegation that "after twenty years of leakage, the Applicant still had not identified the exact source of the leak" is inaccurate with respect to both the situation at the time of the ACRS Subcommittee meeting in 2009 and the status of the leakage issue today. At the ACRS meeting, NSPM witnesses stated that NSPM had "high confidence" that we had identified the exact sources of the leak at that time. Skoyen Exhibit 16 at 69. A 95 percent or more reduction in leakage following subsequent repairs to the floor embedment plates for the reactor internals stands and the rod control cluster ("RCC") change fixture confirmed that NSPM had indeed correctly identified these leak sources. As I explained in my direct testimony (Skoyen Testimony at A24), the results of testing and repairs have led us to conclude that the sources of lower cavity leakage, in addition to those mentioned above, are the wall embedment plates for the RCC assembly guide tube supports. We believe the source of upper cavity leakage to be the sandplug covers or those of the Nuclear Instrumentation System.

Fourth, it is misleading to imply that the refueling cavity leakage at PINGP has been "posing a danger to the integrity of the containment." As I discussed previously (Skoyen Testimony at A20 and A26), we have performed repeated inspections from 1998 to 2010, all of which have found no evidence of degradation in the containment vessel, the rebar, or the concrete. In addition, multiple independent engineering evaluations (Skoyen Exhibits 5, 6, and 8) have concluded that potential exposure of the containment vessel and structures to refueling cavity water has not had an adverse impact on their ability to meet design requirements, and that any potential, theoretical degradation would be so limited as to have no safety significance.

**Q9.** In the same testimony (A19 at 7-8) Mr. Grimes provides a description of the potential safety risks posed by the refueling cavity leakage. He states that "[t]he 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)" and indicates that "[i]f 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." Is Mr. Grimes correct in his assessment of the safety risks posed by the leakage?

**A9.** No, for several reasons. First, Mr. Grimes' reference to a 0.003 square-inch hole as the allowable containment leakage is entirely incorrect, since it is inapplicable to PINGP. This figure, allegedly derived from Appendix H of the Inspection Manual Chapter 0609, is – if at all – a generic value that is not applicable to PINGP, because the units have a dual containment configuration, as described in the Skoyen Testimony at A7. (Having reviewed Appendix H in its entirety, I find no basis for PIIC's derivation of this figure.) A nuclear station's technical specifications generally identify containment leakage acceptance criteria in terms of a percentage of air weight volume, not a "hole size." PINGP's Technical Specifications (Technical Specification 5.5.14.c) establish acceptable criteria as follows: "The maximum allowable primary containment leakage rate, at the containment internal design pressure, shall be 0.25% of primary containment air weight per day." The NRC has, however, estimated the threshold containment leak rate in accordance with Part 100 dose limits, "beyond which the release may become significant to [large early release frequency]," to be approximately equivalent to a hole of 2.5 to 3 inches in diameter for plant containment configurations of the same type as PINGP. See Skoyen Exhibit 17 (NUREG-1765) at 17 n. 2.

Second, the independent engineering evaluation performed by Dominion Engineering, Inc. ("DEI") in 2009 (Skoyen Exhibit 8) concluded that the calculated theoretical conservative upper limit of 0.25 inches of corrosion to the steel containment vessel wall "clearly does not raise a risk of causing leakage through the 1.5 in. thick steel containment vessel in the event of an accident." Skoyen Exhibit 8 at 4-3. (As I explained in the Skoyen Testimony at A26, PINGP does not have a "containment liner," but rather a containment vessel wall of 1.5 inch thickness.) The DEI evaluation established that the actual total corrosion after 36 years is 0.010 inches or less, since the pH of solution in contact with the steel containment vessel will be buffered by alkalinity from the cement in the concrete, which is expected to raise the pH to > 12.5 and reduce actual corrosion rates to close to zero. Id. at 4-4. DEI's conservative assessment, assuming conditions that do not exist (aerated conditions, concentrated boric acid solutions, and no buffering), and taking no credit for the corrective actions already taken to stop the leakage, identified that the upper limit on corrosion would be 0.25 inches after 36 years. Based upon this very conservative analysis, it would take 216 years before corrosion would corrode the full 1.5 inch wall thickness of the containment pressure vessel. Id.

Third, DEI determined that any degradation effects on the concrete due to contact with refueling water to the maximum postulated depth of 0.31 inches would be "negligible." Id. at 5-4.

Fourth, as I discussed previously in the Skoyen Testimony at A19, recent engineering evaluations performed by NSPM (Skoyen Exhibits 11 and 12) have determined that the DEI evaluation's conclusions as to the lack of safety significance of the leakage remain valid, and that the DEI evaluation's postulated maximum degradation levels remain significantly below the design margins required to maintain integrity and functionality of the containment vessel and concrete structures.

Finally, as I explained in the Skoyen Testimony at A20, NSPM is committed to ensuring that the leakage has posed and continues to pose no safety risks by conducting further inspections and tests to confirm that no vessel degradation has occurred.

**Q10.** In the same testimony (A19 at 8), Mr. Grimes asserts that "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." In the PIIC SOP, PIIC's counsel further claims: "the Applicant did not reveal this leakage to the NRC until the fall of 2008, approximately twenty-five years after the Applicant knew about the problem" (PIIC SOP at 7). Are these claims accurate?

**A10.** Absolutely not. With respect to the first claim, as I stated in the Skoyen Testimony at A28, PINGP took significant steps to address the refueling cavity leakage and ascertain its safety significance in 1998, when we ordered an independent safety evaluation from AES. Despite the evaluation's conclusion that any leakage effects would have no safety significance, which was again confirmed by AES in 2006, we conducted numerous tests and implemented a series of repairs to stop the leakage.

The second claim regarding the timing of NSPM's disclosure of the leakage issue to the NRC is also entirely incorrect. Since 1998, PINGP has periodically reported the issue to the NRC by including documentation of refueling cavity leakage, when observed, in its Inservice Inspection ("ISI") summary reports in accordance with the requirements of 10 C.F.R. § 50.55a. See listing of such reports included in Skoyen Exhibit 4 at 32-33. In 2008, after NSPM recognized that it had originally failed to include a discussion of observed refueling cavity leakage in its

2006 Unit 2 ISI summary report, it entered a corrective action and duly amended that report to include a discussion of leakage. Id.; Skoyen Exhibit 18 (CE 1140617-03) at 1-2. Additionally, we performed a condition evaluation to determine any other instances where leakage was inadequately reported, identifying only one other such instance in 2003. Skoyen Exhibit 18 at 1-2. (The 2003 ISI report was not supplemented with an update on cavity leakage because the applicable inspection interval had already been closed by that time. Skoyen Exhibit 4 at 33.)

**Q11.** In the PIIC SOP at 7, PIIC Counsel quotes from a document that is identified as "October 19, 2009 "TRD" on the refueling cavity leakage." What document is this, and what was its purpose?

**A11.** The document PIIC Counsel quotes from is an informal summary of actions and commitments associated with refueling cavity leakage that was prepared for internal use by Tom Downing, one of PINGP's engineers, in 2009 based on his review of site documents and correspondence. The purpose of this document was to provide the status of these actions and commitments for management and projects personnel. It contains inaccuracies, for example, the statement that "[r]efueling cavity leakage has been an issue since the early 1980s," contrary to the fact that PINGP has no records of leakage having occurred prior to 1987. Its content was not verified at the time of its issuance, and it should thus not be relied upon as an accurate or comprehensive description of the refueling cavity leakage issue. As stated above, the root cause evaluation of the refueling cavity leakage issue (Skoyen Exhibit 4) contains the documented chronological history of the issue.

**Q12.** Mr. Grimes' testimony (A34 at 15-16) refers to one of the principles set forth by the NRC in its November 2009 draft Policy Statement on Safety Culture: "[t]he organization ensures that issues potentially impacting safety or security are promptly identified, fully evaluated, and promptly addressed and corrected…." Mr. Grimes appears to conclude that the refueling cavity leakage issue is an instance where this safety culture principle has not been satisfied. Is he correct?

**A12.** No. As I stated previously in the Skoyen Testimony at A20, the PINGP Structures Monitoring Program and the ASME Code Section XI, Subsection IWE Program continually involve monitoring the refueling cavity for leakage and evaluating the condition and integrity of containment vessel structures. Utilizing these programs, NSPM originally discovered refueling cavity leakage, conducted prompt and appropriate evaluations and repairs commensurate with the issue's independently evaluated safety significance, and has further planned inspections and repairs in place to ensure no recurrence and no adverse safety consequences of such leakage. PINGP's Corrective Action Program ("CAP") provides an effective framework for problem identification and resolution, and NSPM will not close the open corrective actions regarding the leakage issue until resolution has been achieved and verified.

**Q13.** Mr. Grimes also testifies (A44 at 19-20): "The failure of the applicant to correct the potential damage to the containment integrity resulting from the refueling cavity leaks . . . [is] indicative of a weak safety culture at PINGP." Do you agree with his conclusion?

**A13.** No. In fact, NSPM has succeeded in identifying and repairing the leak sources and in substantially eliminating leakage, and we have confirmed that the effects of the leakage pose no risk to the integrity of the containment vessel structures. Moreover, as I noted in the Skoyen Testimony at A22 and A23, PINGP currently exhibits a strong safety culture that is the result of a strict process-driven approach to handling identified problems under the CAP procedures. Past insufficiencies in accountability at the organizational level have been remedied, such that identified issues are promptly documented, evaluated, assigned to a manager whose level corresponds to the issue's significance, and approved by the Performance Assessment Review Board prior to corrective action closure. Further, PINGP's root cause evaluation regarding the refueling cavity issue noted positive safety culture assessments relating to PINGP's safety conscious work environment. Skoyen Exhibit 4 at 31.

**Q14.** Mr. Grimes expresses the opinion (A19 at 7) that "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." Is his opinion valid?

**A14.** No. As I stated in the Skoyen Testimony at A27, 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 has repaired the components identified as the source of the leakage. We have committed to conduct further visual inspections in subsequent refueling outages to ensure that the leakage issue has been fully resolved, and to perform further testing of the integrity of the containment vessel in both units. We will not be satisfied with our resolution efforts until the leakage is fully eliminated. The NRC Staff and the Advisory Committee on Reactor Safeguards have both concluded that NSPM's remedial measures and commitments demonstrate its ability to effectively implement the aging management program during the period of extended operation. Skoyen Exhibits 13 at 3 and 14 at 3-23.

**Q15.** Does that conclude your testimony?

**A15.** Yes.