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August 13, 2010

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF'S REBUTTAL STATEMENT OF POSITION
ON THE SAFETY CULTURE CONTENTION

INTRODUCTION

Pursuant to 10 CFR §§ 2.1207(a)(2) and 2.337(g)(2) and the Atomic Safety and Licensing Board's ("Board") April 20, 2010 Order,[1] the Staff of the U.S. Nuclear Regulatory Commission ("Staff") submits its written statement in rebuttal to Prairie Island Indian Community's ("PIIC") and Northern States Power Company's ("NSP" or "Applicant") initial statements of position, written testimony, and exhibits.[2]

SUMMARY OF ARGUMENT

Based on Christopher Grimes' testimony, PIIC has failed to satisfy its burden of going forward. PIIC's testimony and late-filed exhibits fail to make a showing "sufficient to require reasonable minds to inquire further." *Vermont Yankee Nuclear Power Corp. v. N.R.D.C.*, 435 U.S. 519, 554 (1978). In his testimony, Mr. Grimes, PIIC's only proffered expert, did not

[1] Order (Summarizing Prehearing Conference Call and Amending Hearing Schedule), (Apr. 20 2010) (unpublished) ("Scheduling Order").

[2] On July 30, 2010, NSP, PIIC, and the Staff submitted their initial statements of position and written testimony to the Electronic Information Exchange ("EIE"). Also on July 30, 2010, NSP and the Staff submitted their exhibits and exhibit lists in support of their written testimony. On July 31, 2010, PIIC submitted its first exhibit list. Three days later, counsel for the Staff left a voicemail and email for PIIC's counsel, Attachment 1, regarding PIIC's missing exhibits. Two days later, and 5 days after they were due, PIIC submitted its exhibits and a revised exhibit list without explanation regarding the delay. *Compare* Prairie Island Indian Community's Exhibit List, transmitted on July 31, 2010, 12:17 a.m.; *with* Prairie Island Indian Community's Revised Exhibit List, transmitted on August 4, 2010, 4:32 p.m.

establish his expertise with respect to measuring, analyzing, and predicting the safety culture of organizations, especially those engaged in electricity production at nuclear power plants. *See* NRC Staff Rebuttal Testimony of Dr. Valerie E. Barnes, June Cai, Molly Jean Keefe, and Audrey L. Klett Concerning Safety Culture and NRC Safety Culture Policy Development and Implementation ("Safety Culture Rebuttal Testimony"), at A3.

In addition, the Staff has reviewed NPS's safety culture assessment (Applicant Exhibit NSP000057). While the report provides a good deal of information about the PINGP workforce's perceptions of the current state of safety culture at PING, the assessment does not support an overall conclusion as to the current or future "strength" of safety culture at PINGP.

As the Staff explained in its Initial Statement of Position, the Staff has reasonable assurance that the effects of aging will be adequately managed during the period of extended operation based on its review of the Prairie Island Nuclear Generating Plant's ("PINGP") license renewal application and its aging management programs.[3] The Reactor Oversight Process ("ROP") provides reasonable assurance that NSP will comply with its licensing basis, including modifications to the licensing basis resulting from license renewal, in the period of extended operation.

ARGUMENT

I. PIIC Bears the Burden of Going Forward

In this proceeding the burden is on NSP to demonstrate by a preponderance of the evidence that its aging management program satisfies the reasonable assurance standard. *See* *AmerGen Energy Co., LLC* (Oyster Creek Nuclear Generating Station), CLI-09-7, 69 NRC 235,

[3] The following documents are appended to this filing: "Attachment 1;" "NRC Staff Rebuttal Testimony of Richard A. Plasse Concerning Aging Management Programs;" "NRC Staff Rebuttal Testimony of Abdul H. Sheikh and Dr. Dan J. Naus Concerning the Safety Culture Contention and Reactor Refueling Cavity Leakage;" "NRC Staff Rebuttal Testimony of John Giessner Concerning the Safety Culture Contention and the Reactor Oversight Process;" "NRC Staff Rebuttal Testimony of Dr. Valerie E. Barnes, June Cai, Molly Jean Keefe, and Audrey L. Klett Concerning Safety Culture and NRC Safety Culture Policy Development and Implementation;" NRC Staff Exhibit List (dated August 13, 2010); and NRC Staff Exhibit Nos. NRC000055, NRC000057, NRC000058, NRC000001, and NRC000006.

263 (2009); *Commonwealth Edison Co.* (Zion Station, Units 1 & 2), ALAB-616, 12 NRC 419, 421 (1980) (stating that applicants are not held to an absolute standard or required to prove a matter conclusively but rather, consistent with the Administrative Procedure Act, applicants are held to a preponderance standard). PIIC, however, has a burden too: it must put forth a prima facie case in support of its contention by providing probative evidence or expert testimony. *See Id.* (citing *Louisiana Power & Light Co*. (Waterford Steam Electric Station, Unit 3), ALAB-732, 17 NRC 1076, 1093 (1983) (*quoting Consumers Power Co.* (Midland Plant, Units 1 & 2), ALAB-123, 6 AEC 331, 345 (1973)). The Commission recently observed that showing that an event occurred in the past, without more, does not impose a burden on the applicant to show that the event will not occur again during the license renewal period. *See Oyster Creek,* CLI-09-7, 69 NRC at 270. The Supreme Court stated that use of a threshold test requiring an intervenor to make a "showing sufficient to require reasonable minds to inquire further" was reasonable. *Vermont Yankee Nuclear Power Corp.*, 435 U.S. at 554. Only after the intervenor puts forth its prima facie case will the burden shift back to the Applicant, as part of its overall burden of proof, to provide sufficient rebuttal. *Id*. Furthermore, to require NSP to rebut every allegation and every data point of evidence propounded by PIIC would be to hold NSP to essentially a beyond-a-reasonable doubt standard. A showing of reasonable assurance beyond a reasonable doubt, however, is not required by law, regulation, or Commission precedent. *See North Anna Env. Coalition v. NRC*, 533 F.2d 655, 667 (D.C. Cir. 1975). This principle is pertinent here because PIIC has advanced a wide-ranging safety culture contention that cites numerous recent inspection finding in support of its claim that NSP is incapable of managing the effects of aging through its aging management programs ("AMPs").

A. Mr. Grimes Testimony Demonstrates That No Further Inquiry Is Required

Besides Mr. Grimes' lack of expertise, discussed *infra* in Section I.B, his testimony lacks sufficient facts and analysis to warrant additional inquiry. Mr. Grimes states that in his opinion, "there … 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 required by 10 CFR [§] 54.29(a)(1)." Grimes at 6, A16. Mr. Grimes bases his conclusion on the borated water leak from the refueling cavity, among other things. His analysis is based on unsupported assumptions, relies on incorrect data, and does not address all of the relevant facts. Further, his testimony is largely conclusory: Mr. Grimes quotes the Staff's or NSP's documents and follows that with a contrary conclusion that is unaccompanied by substantive analysis. The bulk of his testimony is simply a repetition of the declaration he submitted in support of the contention's admissibility. While that testimony might suffice for contention admissibility, substantially more fact and analysis is required to proceed forward at this point to place the burden on the Applicant to disprove the contention's broad conclusions. Accordingly, the testimony fails to demonstrate that PINGP's safety culture is not adequate to provide reasonable assurance that aging effects will be managed during the period of extended operation.

Mr. Grimes' testimony regarding the refueling cavity leak also contains fundamental flaws identified by the Staff's experts. First, he claims that leakage poses an immediate risk to the containment integrity. Grimes at 7, A19. Then, he posits that the allowed leakage for containment is a 0.003 square inch hole and that continued contact with the borated water would likely result in violation of 10 CFR Part 100. Finally, Mr. Grimes concludes that NSP's new attitude regarding repair of the refueling cavity leakage demonstrates that it is not capable of implementing an AMP.

Mr. Grimes' testimony, that the risk from the leak poses a serious and immediate challenge for containment, is not supported by the facts or the underlying physical corrosion processes. In the Staff's initial and rebuttal testimony, Mr. Sheikh and Dr. Naus explain that samples and testing to date show no appreciable corrosion. NRC Staff Rebuttal Testimony of Abdul H. Sheikh and Dr. Dan J Naus Concerning the Safety Culture Contention and the Refueling Cavity Leakage ("Sheikh and Naus Rebuttal Testimony") at A5; NRC Staff Testimony of Dr. Dan J Naus and Abdul H. Sheikh Concerning the Safety Culture Contention and the Refueling Cavity Leakage ("Sheikh and Naus Initial Testimony") at A9, A19. The steel containment has been examined using multiple methods and retains a measured thickness in excess of its installed nominal thickness. *Id.*; NRC Staff Rebuttal Testimony of John Giessner Concerning the Safety Culture Contention and the Reactor Oversight Process at A2. The Staff's expert explained that the necessary conditions for rapid corrosion of the containment are simply not present. Sheikh and Naus Rebuttal Testimony at A5. The presence of concrete on either side of the steel containment structure minimizes the rate of corrosion. *Id.* The presence of the concrete buffers the boric acid in the leaked water and thus raises the pH level of the water to neutral or slightly basic, thereby reducing the rate of corrosion. *Id*. And the lack of space between the steel containment structure and the concrete on either side of it, minimizes the presence of oxygen, a necessary component for the type of corrosion Mr. Grimes asserts. *Id.* Dr. Naus and Mr. Sheikh confirmed that under the worst corrosion rate, based on the conditions found to date, the reduction in thickness for the steel containment would be only 0.003 inches after 60 years of leakage. *Id.* at A6. The steel containment design has a corrosion allowance of 0.25 inches, or more than 80 times the expected worst case corrosion after 60 years. *Id.* Thus, Mr. Grimes' conclusion that the risk the leak poses is serious and immediate is not supported by fact.

Mr. Grimes, also, asserts that design basis leakage is a 0.003 square inch hole. In support of that assertion, he cites Appendix H of Inspection Manual Chapter 0609, PIIC's Exhibit 5. However, after a careful review of Appendix H, the Staff could not find any information regarding a design basis leakage rate equivalent to a 0.003 square inch hole. Sheikh and Naus Rebuttal Testimony at A4. Appendix H merely provides a probabilistic risk framework for evaluating the risk significance of findings.

Mr. Grimes' conclusion that a 0.003 square inch hole in the containment could result in a radiological exposure in excess of 10 CFR Part 100 is also flawed. Mr. Grimes' analysis does not appear to account for the fact that any corrosion that is occurring is located below grade and abuts concrete. *See* Sheikh and Naus Rebuttal Testimony at A8. It does not address the fact that plants, like Prairie Island, that have steel containments abutting concrete have continued to pass integrated containment leakage tests even when the containment steel has exhibited perforations. *Id.* Simply put, Mr. Grimes' conclusory testimony ignores the support that the concrete provides to the containment.

Finally, Mr. Grimes asserts that NSP's "attitude" to resolving the refueling cavity leakage, which has been determined to have no safety significance, shows that NSP does not have the appropriate safety culture to provide reasonable assurance that effects of aging will be managed. Grimes at 8, A19. But, the NRC Staff points out that NSP's efforts to address the reactor refueling cavity leakage are reasonable in light of the inspection results showing no corrosion, the intermittent nature of the leakage, and the limited window of time to address the leakage during each refueling. Sheikh and Naus Rebuttal Testimony at A3. Mr. Grimes' simple assertions account for none of these factors.

B. <u>Mr. Grimes Experience and Education Fail to Establish His Expertise on Safety Culture</u>

PIIC's contention presumes that PINGP's past and present safety culture can be used to predict its future safety culture. Assuming, for the sake of argument, that safety culture can be predicted, PIIC would then have to establish the effect that the predicted safety culture would have on the AMPs that the Staff has evaluated and determined to be acceptable. At a minimum, PIIC must put forward an expert in safety culture to support its prediction and link the prediction to the implementation of AMPs. The Staff's expert, Dr. Barnes, has testified that in order to qualify as a safety culture expert a witness would need extensive education and practical experience. Safety Culture Rebuttal Testimony at A3 – A4. According to Dr. Barnes, a safety culture expert should have a master's degree in psychology, sociology, or organizational behavior and at least 1 year of experience performing and evaluating safety culture assessments under the supervision of a qualified person. *Id.* Mr. Grimes does not have either the education or the practical experience necessary to qualify him as an expert on safety culture.

He appears to conflate his experience with reactor safety and safety management with safety culture. The Staff's experts, however, explain that safety culture expertise is not simply obtained through experience with reactor safety or safety management. *Id.* at A6 – A8. While Mr. Grimes has experience with license renewal, environmental impacts, rulemaking, regulatory analysis, and the operation and safety of pressurized water reactors, these do not qualify him as an expert on safety culture. Moreover, Mr. Grimes has not designed, performed, or evaluated any safety culture assessments. He possesses no particular educational experience that allows him to evaluate a safety culture assessment or design a valid safety culture assessment. His educational background is in nuclear engineering. While that education might allow him to testify as an expert regarding aspects of reactor design, it does convey expertise or insight into complex organizational behavior necessary to opine on safety culture. Mr. Grimes cannot provide expert testimony regarding the safety culture at PINGP and its effect on PINGP's aging management program.

II. <u>NSP'S Safety Culture Assessment Is Informative But Not Conclusive</u>

NSP's safety culture assessment is informative of the PINGP workforce's perceptions of safety culture at PINGP, but it does not support an overall conclusion about the "strength" or "weakness" of safety culture at PINGP either now or in the future. Safety Culture Rebuttal Testimony at A22 & A24. As explained in the attached testimony, there is no agreed-upon or empirically validated measure or threshold for determining the "strength" or "weakness" of an organization's safety culture. *Id.* at A12. NPS' safety culture assessment appears to have relied upon a methodology the nuclear industry is currently developing. *Id.* at A21. The NRC Staff reviewed a version of that methodology and continues to have concerns about the reliability and validity of safety culture assessments performed using that methodology. *Id.*

Therefore, NPS's safety culture assessment does not support an overall conclusion about the "strength" (or "weakness") of the current safety culture at PINGP. *Id.* at A22. Moreover, the assessment is not probative of future state of safety culture at PINGP. *Id.* at A24. Instead, the continuous oversight provide by the Rector Oversight Process is the most appropriate approach to ensuring that PINGP continues to address safety culture issues as they arise and that it implements its aging management programs effectively. *Id.* at A26.

CONCLUSION

As previously asserted in the Staff's Initial Statement of Position, the Board should not sustain PIIC's claim that PINGP's safety culture is inadequate to provide the reasonable assurance required by 10 CFR § 54.29(a)(1). Mr. Grimes lacks the requisite expertise to opine on safety culture, and his testimony is based on unsupported assumptions and incorrect data and fails to address all of the facts. Thus, PIIC has not presented evidence or expert opinion sufficient to require further inquiry or to shift the burden of proof to the Applicant. The NRC's review of the aging management programs and continuing regulatory oversight provide reasonable assurance that PINGP is operating and will continue to operate safely and consistent with its licensing basis and NRC regulations. There is no factual or regulatory basis for PIIC's safety culture contention and it should not be sustained.

Respectfully Submitted,
/Executed in Accordance with 10 CFR § 2.304(d)
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CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing NRC Staff's Rebuttal Statement of Position on the Safety Culture Contention; Attachment 1; NRC Staff Rebuttal Testimony of Richard A. Plasse Concerning Aging Management Programs; NRC Staff Rebuttal Testimony of Abdul H. Sheikh and Dr. Dan J. Naus Concerning the Safety Culture Contention and Reactor Refueling Cavity Leakage; NRC Staff Rebuttal Testimony of John Giessner Concerning the Safety Culture Contention and the Reactor Oversight Process; NRC Staff Rebuttal Testimony of Dr. Valerie E. Barnes, June Cai, Molly Jean Keefe, and Audrey L. Klett Concerning Safety Culture and NRC Safety Culture Policy Development and Implementation; NRC Staff Exhibit List (dated August 13, 2010); and NRC Staff Exhibit Nos. 60 through 63 & 4A, dated August 13, 2010, have been served upon the following by the Electronic Information Exchange, this 13th day of August, 2010:

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