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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 INITIAL STATEMENT OF POSITION
ON THE SAFETY CULTURE CONTENTION

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July 30, 2010

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INTRODUCTION

Pursuant to 10 CFR §§ 2.1207(a)(1) and 2.337(g)(2) and the Atomic Safety and Licensing Board Panel's ("Board") April 20, 2010 Order,[1] the Staff of the U.S. Nuclear Regulatory Commission ("Staff") submits its initial written statement of position and written testimony with supporting affidavits on the Prairie Island Indian Community's ("PIIC") admitted contention.[2]

SUMMARY OF ARGUMENT

Based on its review of the Prairie Island Nuclear Generating Plant's ("PINGP") license renewal application and its aging management programs, the Staff has reasonable assurance that the effects of aging will be adequately managed during the period of extended operation.

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1 Order (Summarizing Prehearing Conference Call and Amending Hearing Schedule), (Apr. 20 2010) (unpublished) ("Scheduling Order").

2 The following documents are appended to this filing: "NRC Staff Testimony of Richard A. Plasse Concerning Safety Culture Contention;" "NRC Staff Testimony of Abdul H. Sheikh and Dr. Dan J. Naus Concerning the Safety Culture Contention and the Reactor Refueling Cavity Leakage;" "NRC Staff Testimony of John Giessner Concerning the Safety Culture Contention and the Reactor Oversight Process;" "NRC Staff 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;" and NRC Staff Exhibit Nos. 1 through 59.

Through its operating reactor oversight program, the Staff has already fully evaluated and addressed the operating issues that PIIC relies on as bases for its contention. The Staff found that those issues do not impair the Staff's determination that there is reasonable assurance that PINGP can operate safely during its current period of operation. The NRC will rely on that same oversight process to ensure that PINGP operates safely and in conformance with its plans to manage aging during the period of extended operation, should a renewed license be issued. The NRC will also rely on that process to systematically monitor the safety culture at PINGP. Thus, for the reasons set forth below and in the testimony filed herewith, the Staff submits that PIIC's safety culture contention challenging Northern States Power Company's ("NSP" or "Applicant") application to renew the operating licenses for the PINGP Units 1 and 2 should not be sustained.

BACKGROUND

This proceeding arises from the application of NSP to renew the operating licenses for the Prairie Island Nuclear Generating Plant Units 1 and 2 for an additional 20-year period following the current expiration dates of August 9, 2013 (Unit 1) and October 29, 2014 (Unit 2).[3] In the course of this proceeding, PIIC filed eleven contentions challenging NSP's application to renew the operating licenses for PINGP Units 1 and 2.[4] The Board admitted seven contentions,[5] six of which were resolved by the parties.[6] The Board dismissed the seventh as moot on January 5, 2010.[7]

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[3] *See* Letter from Michael D. Wadley, Nuclear Management Company, LLC, to U.S. NRC, Re: Application for Renewed Operating Licenses (April 11, 2008) (Agencywide Documents and Access Management System (ADAMS) Accession No. ML081140720). Subsequent to submission of the license renewal application, the NRC approved the transfer of operating authority over Prairie Island Nuclear Generating Station, Units 1 and 2 from Nuclear Management Company, LLC ("NMC") to NSP. *See* Order Approving Transfer of License and Conforming Amendment, 73 Fed. Reg. 55550 (Sept. 25, 2008).

[4] Prairie Island Indian Community's Notice of Intent to Participate and Petition to Intervene (August 18, 2008) (ADAMS Accession No. ML082391038).

[5] *Northern States Power Co.* (formerly Nuclear Management Co., LLC) (Prairie Island Nuclear Generating Plant, Units 1 and 2), LBP-08-26, 68 NRC 905 (2008).

[6] Order (Approving Settlement and Dismissal of Contentions 1, 6 and 11) (Apr. 14, 2009) (unpublished) (ADAMS Accession No. ML091040305); Order (Granting Motions to Dismiss PIIC Contentions 7 and 8) (June 12,

On November 23, 2009, PIIC submitted the sole remaining contention in this proceeding.

> Contrary to the conclusion in the Safety Evaluation Report (SER), the Community does not believe that "the requirements of 10 CFR 54.29(a) have been met." Due to recent significant non-compliances with NRC regulations, as well as the applicant's failure to address a known potentially serious safety problem identified in the SER, the Community does not believe that there is any justification for a reasonable assurance determination by the NRC that the applicant will "…manag[e] the effects of aging during the period of extended operation on the functionality of structure and components" as required by 10 CFR 54.29(a)(1). [8]

On January 28, 2010, the Board rephrased and admitted the new safety contention as follows:

> PINGP's safety culture is not adequate to provide the reasonable assurance required by 10 C.F.R. § 54.29(a)(1) that PINGP can manage the effects of aging during the requested period of extended operation.[9]

On February 12, 2010, the Staff and NSP filed petitions for interlocutory review of the Board's January 2010 Order.[10] These petitions for interlocutory review are currently pending before the Commission.

2009) (unpublished) (ADAMS Accession No. ML091630288); Order (Approving Settlement and Dismissal of Contention 2) (July 16, 2009) (unpublished) (ADAMS Accession No. ML091970234).

[7] Order (Granting Motion to Dismiss PIIC Contention 5) (January 5, 2010) (unpublished) (ADAMS Accession No. ML100050289). PIIC also filed three contentions challenging the Staff's draft Supplemental Environmental Impact Statement. *See* Prairie Island Indian Community's Motion for Leave to File New Contentions on NRC's Draft Supplemental Environmental Impact Statement (Dec. 14, 2009) (ADAMS Accession No. ML093480606). The Board denied admission of these contentions. Order (Granting Motion for Leave to File New Contentions and Denying Their Admission) (February 25, 2010) (unpublished) (ADAMS Accession No. ML100560382).

[8] Prairie Island Indian Community's Submission of a New Contention on the NRC Safety Evaluation Report (Nov. 23, 2009) (ADAMS Accession No. ML093270615).

[9] Order (Narrowing and Admitting PIIC's Safety Culture Contention) (January 28, 2010) (unpublished) (Agency Document Access & Management System (ADAMS) Accession No. ML100280537) ("January 2010 Order") at 14.

[10] NRC Staff's Petition for Interlocutory Review of Atomic Safety and Licensing Board Decision Admitting Late-Filed and Out of Scope Safety Culture Contention (Feb. 12, 2010) (ADAMS) Accession No. ML100431768); Northern States Power Company's Petition for Interlocutory Review of an Order Admitting a Safety Culture Contention (Feb. 12, 2010) (ADAMS Accession No. ML100431198). The parties have filed answers to both petitions and replies to the answers. *See* NRC Staff's Answer to Northern States Power Company's Petition for Interlocutory Review of an Order Admitting a Safety Culture Contention (Feb. 22, 2010) (ADAMS Accession No. ML100540288); Northern States Power Company's Answer Supporting NRC Staff Petition for Interlocutory Review (Feb. 22, 2010) (ADAMS Accession No. ML100540291); Prairie Island Indian Community's Answer to the NRC Staff's and Northern States Power Company's Petitions for Interlocutory Review of the Atomic Safety and Licensing Board Decision

LEGAL & REGULATORY REQUIREMENTS

I. Scope of License Renewal Proceedings

The Commission has limited the scope of license renewal proceedings to the "review of plant structures and components that will require an aging management review for the period of extended operation and the plant's systems, structures and components that are subject to an evaluation of time-limited aging analyses." *Duke Energy Corp.* (McGuire Nuclear Station, Units 1 & 2; Catawba Nuclear Station, Units 1 & 2), CLI-01-20, 54 NRC 211, 212 (2001). The scope of license renewal reviews is limited because "[l]icense renewal reviews are not intended to 'duplicate the Commission's ongoing review of operating reactors.'" *Florida Power & Light Co.* (Turkey Point Nuclear Generating Plant, Units 3 & 4), CLI-01-17, 54 NRC 3, 7 (2001) (*citing* Final Rule, "Nuclear Power Plant License Renewal," 56 Fed. Reg. 64,943, 64,946 (Dec. 13, 1991)). The Commission's on-going regulatory process, which includes generic and plant-specific reviews, plant inspections, and enforcement actions, ensures the adequacy of and compliance with the current licensing basis ("CLB"). *Entergy Nuclear Generation Co. and Entergy Nuclear Operations, Inc.* (Pilgrim Nuclear Power Station), CLI-10-14, 71 NRC __ (Jun. 17, 2010)(slip op. at 4). Thus "the Commission has concluded that the 'only issue' where the regulatory process may not adequately maintain a plant's current licensing basis involves the potential 'detrimental effects of aging on the functionality of certain structures, systems, and components in the period of extended operations." *Id*. at __ (slip op. at 4) (*citing* "Nuclear

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Admitting the Community's Contention on Safety Culture (Feb. 22, 2010) (ADAMS Accession No. ML100540292); Northern States Power Company's Reply to Answers to Its Petition for Interlocutory Review (Mar. 1, 2010) (ADAMS Accession No. ML100601334); NRC Staff's Reply to Prairie Island Indian Community's Answer to the Staff's Petition for Interlocutory Review (Mar. 1, 2010) (ADAMS Accession No. ML100601332).

On March 3, 2010 the Staff filed a motion for leave to supplement its February 12 petition for interlocutory review. NRC Staff's Motion for Leave to Supplement Its Petition for Interlocutory Review of Atomic Safety and Licensing Board Decision Admitting Late-Filed and Out of Scope Safety Culture Contention (ADAMS Accession No. ML100670555). NSP filed an answer supporting the Staff's motion. Northern States Power Company's Answer in Support of NRC Staff's Motion for Leave to Supplement Petition for Review (Mar. 18, 2010) (ADAMS Accession No. ML100770546). PICC filed an answer opposing the Staff's motion. Prairie Island Indian Community's Answer in Opposition to the NRC Staff's Motion for Leave to Supplement Its Petition for Interlocutory Review (Mar. 18, 2010) (ADAMS Accession No. ML100770585).

Power Plant License Renewal; Revisions," 60 Fed. Reg. 22,461, 22,464 (May 8, 1995)).

The principle that current performance issues are outside the scope of license renewal is explicitly incorporated into the Commission's license renewal rule. Pursuant to 10 C.F.R. § 54.30(a), if there is not reasonable assurance that licensed activities will be conducted in accordance with the CLB during the current license term, the licensee is required by its current license to take appropriate actions to ensure that the intended functions of systems, structures and components will be maintained in accordance with the CLB throughout the term of its current license. Section 54.30(b), in turn, provides that licensee compliance with section 54.30(a) is not within the scope of license renewal. Thus, current performance issues, and the adequacy of licensee actions to address those issues, are outside the scope of this proceeding.[11]

II. Regulatory Requirements & Guidance for License Renewal

10 C.F.R. § 54.29 establishes the criteria for the issuance of renewed operating licenses. It provides, in relevant part:

> A renewed license may be issued . . . if the Commission finds that:
>
> (a) Actions have been identified and have been or will be taken with respect to matters identified in paragraphs (a)(1) and (a)(2) of this section, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB . . . . These matters are:
>
> (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under § 54.21(a)(1); and
>
> (2) time-limited aging analyses that have been identified to require review under § 54.21(c).

Thus, the NRC's reasonable assurance determination for license renewal is based on whether the applicant's aging management programs will effectively manage aging. All other matters

[11] Concerns about ongoing operational issues may be raised via the 10 CFR § 2.206 process. *See AmerGen Energy Co.* (Oyster Creek Nuclear Generating Station), CLI-08-23, 68 NRC 461, 486 (2008) (stating that the appropriate avenue for raising concerns about a license renewal applicant's maintenance of certain equipment and commitment tracking system is the 10 CFR § 2.206 process).

are addressed by the NRC's on-going regulatory oversight. *See Turkey Point*, CLI-01-17, 54 NRC at 7.

The NRC Staff's safety review of license renewal applications is guided by two documents: NUREG-1800 "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," Rev. 1 (September 2005) ("SRP-LR") and NUREG-1801 "Generic Aging Lessons Learned Report," Rev. 1 (September 2005) ("GALL Report"). The SRP-LR guides the Staff's review by assigning responsibilities among staff technical organizations, describing methods identifying structures, systems, and components subject to aging management review. The SRP-LR also identifies and defines the ten elements of an effective aging management programs. The ten elements are: (1) scope of program; (2) preventive actions; (3) parameters monitored/inspected; (4) detection of aging effects; (5) monitoring and trending; (6) acceptance criteria; (7) corrective actions; (8) confirmation process; (9) administrative controls; (10) operating experience. Each element is defined in SRP-LR Section A.1.2.3 "Aging Management Program Elements."

The GALL Report was prepared at the direction of the Commission to enhance the predictability, consistency, and efficiency of NRC reviews of license renewal applications. *Entergy Nuclear Vermont Yankee LLC. and Entergy Nuclear Operations, Inc.,* (Vermont Yankee Nuclear Power Station), CLI-10-17, 72 NRC __ (Jul. 18, 2010)(slip op. at 45); SECY-01-0074, *Approval to Publish Generic License Renewal Guidance Documents* (July 2, 2001) (ADAMS Accession No. ML011860200). The GALL Report contains generic aging management programs that are acceptable to the Staff, "based upon experiences and analyses of existing programs at operating plants during the initial license period." *Oyster Creek*, CLI-08-23, 68 NRC at 467. License renewal applicants may reference the GALL Report to demonstrate compliance with the requirements of the license renewal rule and a "license renewal applicant's use of an aging management program identified in the GALL Report constitutes reasonable assurance that it will manage the targeted aging effect during the renewal period." *Oyster*

Creek, CLI-08-23, 68 NRC at 468. A license renewal applicant's commitment to implement an aging management program that the Staff finds is consistent with the GALL Report "constitutes reasonable assurance that it *will* manage the targeted aging effect during the renewal period." *Vermont Yankee*, CLI-10-17 72 NRC at__ (slip op. at 44) (emphasis in original).

The contention at issue in this proceeding asserts that PINGP's safety culture is not adequate to provide the reasonable assurance required by 10 CFR § 54.29(a)(1) that PINGP will manage the effects of aging during the period of extended operation. January 2010 Order at 14. The contention does not challenge the adequacy of PINGP's aging management programs to manage the detrimental effects of aging, nor does PIIC contend that any particular one of PINGP's aging management programs fails to manage the targeted aging effect during the proposed period of extended operation. Instead, PIIC asserts, that because of performance deficiencies identified by the NRC through its on-going regulatory oversight, there is no reasonable assurance that PINGP will implement its aging management programs. There is, however, no basis in law or fact for PIIC's assertion. Consequently, PIIC's contention cannot be sustained.

STAFF WITNESSES

The Staff is proffering the testimony of eight witnesses in support of this Initial Statement of Position: Richard A. Plasse, Abdul H. Sheikh, Dr. Dan J. Naus, John (Jack) B. Giessner, Dr. Valerie Barnes, June Cai, Molly Jean Keefe, and Audrey L. Klett. Information about each of the witnesses is footnoted when the witness's testimony is first cited.

DISCUSSION

I. The NRC Has Reasonable Assurance that the Applicant Will Adequately Manage the Effects of Aging During the Period of Extended Operations in Light of the Review the NRC Staff Completed on the License Renewal Application

A. The NRC Staff Conducted a Complete Review of the License Renewal Application

The Staff's review of the PINGP license renewal application fully satisfies the regulatory requirements for a finding of reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis and that aging effects will be managed during the period of extended operation. The Staff reviewed the license renewal application, issued requests for additional information, reviewed the responses thereto, and conducted on-site inspections and audits. NRC Staff Testimony of Richard A. Plasse Concerning Safety Culture Contention ("Plasse"), at A8.[12] The Staff scrutinized each one of NSP's aging management programs from a technical and engineering standpoint to determine whether it would adequately manage the effects of aging. *Id.*

Some of the aging management programs ("AMPs") are reflected in existing programs. In other words, the applicant currently has in place programs to manage the effects of aging for some systems, structures, and components and intends to continue to use those existing programs during the period of extended operation. Plasse at A9. With respect to other AMPs, the applicant has committed to implement those AMPs when it enters the period of extended operation. *Id*.

If an existing program is credited for managing the effects of aging in license renewal, it is part of the applicant's current licensing basis; if a proposed AMP is a commitment it will become part of the applicant's current licensing basis if and when the applicant moves into the period of extended operation. *Id*. Regardless, whether the AMP currently exists or whether it is a program that will go into effect at the start of the license renewal period, if it was credited for license renewal purposes, it was subject to Staff review. Plasse at A8. The Staff ultimately found that all of the AMPs in the license renewal application were acceptable and that, because

[12] Mr. Richard A. Plasse is a project manager in the NRC's Division of License Renewal. Mr. Plasse holds a B.S. degree in chemical engineering and has twenty eight years of engineering experience primarily in the nuclear industry. Mr. Plasse began his engineering career at the Norfolk Naval Shipyard where he worked first as a Shift Test Engineer and then as an Assistant Chief Test Engineer. From 1987-1995, Mr. Plasse worked for the NRC as a Resident Inspector at Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant. From 1995 to 2008, Mr. Plasse worked for the New York Power Authority and, subsequently, Entergy, at James A. Fitzpatrick Nuclear Power Plant, holding the positions of Project Manager, Senior Licensing Engineer, and Licensing Manager. Since returning to the NRC in 2008, Mr. Plasse has worked in the Division of License Renewal as a project manager, where he is responsible for coordinating the safety review of license renewal applications.

of them, there is reasonable assurance that aging effects will be managed during the period of extended operation. The Staff's findings of reasonable assurance are memorialized in the Safety Evaluation Report, and the Staff's conclusion was subject to review by the Advisory Committee for Reactor Safeguards ("ACRS"). Plasse at A7 and A10. The ACRS also concluded that there was reasonable assurance that PINGP would be operated in accordance with its current licensing basis during the period of extended operation. Plasse at A11.

> The programs established and committed to by the applicant to manage age-related degradation provide reasonable assurance that PINGP, Units 1 and 2 can be operated in accordance with their current licensing bases for the period of extended operation without undue risk to the health and safety of the public.

Advisory Committee on Reactor Safeguards Report on Safety Aspects of the License Renewal Application for the Prairie Island Nuclear Generating Plant, Units 1 &2 (December 10, 2009), p.3 (~~NRC Staff~~Applicant Exhibit No. ~~3~~NSP000013).

Neither the Staff nor the ACRS evaluated whether the applicant would actually implement the AMPs. Plasse A16 and A17. But this does not mean that implementation of the AMPs is exempt from NRC oversight. All of the AMPs will become part of PINGP's licensing basis before the plant enters the period of extended operation, either because they are existing programs and thus already part of the licensing basis or because they will become part of the licensing basis through the update to the Final Safety Analysis Report (FSAR) pursuant to 10 C.F.R. §§ 54.3 and 54.37(b). This means that some of them already are, and ultimately all of them will be, subject to inspection and assessment by NRC resident inspectors and Regional Staff. Plasse at A9. These inspections and assessments, an integral part of the Reactor Oversight Process ("ROP"), will continue through the period of extended operation. Plasse at A17. The ROP's inspection and assessment programs are designed to verify that plants are operated safely and in compliance with the current licensing basis, including any applicable AMPs. *Id.* In addition, after the license is issued, and prior to the period of extended operation, Regional Staff will perform a focused inspection in accordance with the guidance in NRC

Inspection Manual Chapter 71003, "Post-Approval Site Inspection for License Renewal" (NRC Staff Exhibit No. ~~20~~NRC000005). The Post-Approval inspection will examine a sample of the license renewal commitments to determine whether the licensee has implemented them. Plasse at A18. But for determining day-to-day on-going oversight of compliance with the current licensing basis during the period of extended operation, the Staff looks to the ROP. Enforcement of the current licensing basis is under the auspices of the ROP, not the license renewal program.

The Staff's position is that the licensee's implementation of AMPs will be appropriately addressed by the ROP on an on-going basis when the plant is actually in the period of extended operations. That position is bolstered by the regulatory structure of the license renewal process. The license renewal review process ensures that the AMPs submitted in the license renewal application are robust enough to manage the effects of aging for the period of extended operation. Oversight of the provisions of the AMPs in the renewed license is managed by the ROP in the period of extended operation. The license renewal review process and the associated regulations ensure that existing programs that are credited for managing aging and proposed programs that are submitted in the license renewal application are incorporated into the licensing basis through incorporation into the FSAR pursuant to 10 C.F.R. §§ 54.3 and 54.37(b), after the renewed license is approved. The regulatory process simply does not contemplate the kind of review required by the contention. It does not require the Staff to determine whether an applicant will actually implement an AMP at some point in the future. What it does require is that the Staff determine whether the AMPs, on engineering and technical grounds, will adequately address the effects of aging in the future. And this is the determination that the Staff has made and that is memorialized in the Staff's SER.

The fact that compliance with the current licensing basis is outside of the scope of license renewal does not mean that the Staff ignores compliance issues. On the contrary, compliance with the current licensing basis is one of the fundamental tenets of the ROP.

Compliance need not be examined in license renewal because it is already subject to comprehensive and continuous examination under the ROP. Final Rule, 56 Fed. Reg. 64,946 (Dec. 13, 1991).

B. Refueling Cavity Leakage

During its on-site audit of aging management programs at PINGP, the Staff identified an ongoing issue involving water seeping from the refueling cavity into the containment sumps. NRC Staff Testimony of Dr. Dan J. Naus and Abdul Sheikh Concerning the Safety Culture Contention and the Reactor Refueling Cavity Leakage (hereinafter cited by last name(s) of testifying expert or experts), at A11.[13] PIIC cites this leakage as a basis for its contention that the applicant's safety culture is inadequate to support a finding of reasonable assurance. On the contrary, the way in which the Staff addressed the leakage demonstrates that its conclusion of reasonable assurance is justified.

PINGP has experienced intermittent leakage of borated water from the reactor refueling

[13] Dr. Dan J. Naus is employed as a Distinguished Research Staff Member at Oak Ridge National Laboratory ("ORNL") by UT-Battelle, LLC. Dr. Naus has over thirty years of experience as a materials research engineer primarily addressing concrete and concrete-related materials. For the last twenty years, Dr. Naus has been responsible for conducting concrete materials-related research addressing aging management of nuclear power plant safety-related structures. The most recent of these activities addressed an assessment of the effects of elevated temperature on concrete properties and performance. Dr. Naus is currently involved in license renewal activities for the NRC's Division of License Renewal related to assessments of safety-related structures. This work involves technical support in the civil/structural area, including audits and reviews of aging management programs, aging management reviews, and time-limited aging analyses of license renewal applications at six nuclear power plants. Dr. Naus serves on the technical committees of a number of organizations that develop standards related to nuclear power plant concrete structures, namely, the American Society of Mechanical Engineers ("ASME"), the American Concrete Institute (ACI), and the International Federation for Structural Concrete.

Mr. Abdul H. Sheikh is a Senior Structural Engineer in the Division of License Renewal in the Office of Nuclear Reactor Regulation ("NRR"). Mr. Sheikh has over thirty years of experience as a structural engineer. Prior to joining the NRC, Mr. Sheikh worked for Bechtel Power Corporation for 25 years. At Bechtel, Mr. Sheikh performed or supervised the structural analysis and design of complex structures and components for nuclear power plants. During his time at Bechtel, Mr. Sheikh served as the corporation's technical expert for the design of nuclear power plant containment and internal structures. Since joining the NRC in 2004, Mr. Sheikh has worked in the Office of Nuclear Regulatory Research, where he was a member of the state of the art consequence analysis ("SOARCA") project team and was responsible for performing detailed assessment of pressurized water reactors ("PWR"), boiling water reactors ("BWR"), and ice condenser containments subjected to beyond design basis loads, and Office of New Reactors, where he was involved in the review of design of ASME class 1, 2, 3 components for new reactor designs. Mr. Sheikh's current responsibilities include performing safety reviews and onsite audits of nuclear power plant structures focusing in particularly on borated water leakage from reactor refueling cavity and spent fuel pools, and containment corrosion and degradation. Mr. Sheikh is involved in revisions to the GALL Report with respect to PWR and BWR structures, and serves on the American Institute of Steel Construction ("AISC") committee N690 for Specification for Safety-Related Steel Structures for Nuclear Facilities.

cavity since the late 1980s. Naus & Sheikh at A5. While the leakage has been of a limited volume of water and only occurs during refueling, it has the potential to initiate corrosion of the steel containment vessel and result in the erosion of concrete structures associated with containment. Naus & Sheikh at A5 and A6.

In order to address the issue, the Staff requested additional information from the applicant, held a public meeting on the issue, conducted a supplemental audit at the plant regarding the effects of the leakage, and reviewed the means the applicant proposed to assess the condition of the containment vessel and stop the leakage. Naus & Sheikh at A11. When the Staff issued its SER, it did not have sufficient information to conclude that the applicant had identified appropriate actions to manage the effects of aging related to the refueling cavity leakage, and so the Staff included the issue as an Open Item in the SER pending additional information and action by NSP. Plasse at A12. Safety Evaluation Report With Open Items Related to the License Renewal of Prairie Island Nuclear Generating Plant Units 1 & 2, at 3-139 to 3-143 (NRC Staff Exhibit No. ~~2~~NRC000002). The Staff ultimately closed the Open Item based on commitments offered by the applicant: to permanently repair the leakage, to remove concrete and perform ultrasonic testing of the containment vessel, to inspect exposed steel reinforcements for degradation, and to remove and test concrete from the area wetted by the leakage. Naus & Sheikh at A15; Plasse at A14; Safety Evaluation Report, Appendix A, Commitments 41, 42, and 44, at A-10 to A-11 (NRC Staff Exhibit No. ~~1~~NRC000003). The Staff concluded that actions had been identified to effectively manage the effects of aging related to refueling cavity leakage. And the ACRS and its subcommittee, both of which heard presentations from the Staff and the applicant on the issue of reactor cavity leakage, agreed with the Staff's conclusions on the issue. Naus &Sheikh at A16; Plasse at A15.

The Staff's extensive and in-depth examination of refueling cavity leakage at PINGP, its inspections and audits, its questioning of the applicant, and its review of the applicant's responses and commitments fully support the Staff's conclusion that there is reasonable assurance that the effects of aging, including the effects of refueling cavity leakage, will be addressed during the period of extended operation.

II. The Reactor Oversight Process

The NRC Staff conducted its normal review of the PINGP License Renewal Application (*LRA") and concluded that it met all the applicable requirements for license renewal. The Safety Culture Contention, as admitted by the Board, asserts that the safety culture at PINGP "is not adequate to provide the reasonable assurance required by 10 C.F.R. § 54.29(a)(1) that PINGP can manage the effects of aging during the period of extended operation." January 2010 Order at 8. The Board noted that PIIC relied on several recent inspection findings to support its assessment of the safety culture's adequacy. *Id.* at 11. The NRC has addressed the issues that form the basis for the contention, the NRC inspection findings and safety culture at PINGP, under its ROP.

Managing the effects of aging would be a part of PINGP's licensing basis if the NRC renewed the operating license.[14] 10 CFR § 54.33(b). Thus, the contention, as admitted, challenges the ability of the applicant to comply with its CLB during the period of extended operation. But, under the ROP, the NRC continuously conducts sufficient inspections and assessments to support a finding of reasonable assurance that the Applicant is in compliance with its licensing basis. The events on which PIIC relies to support the contention do not undermine that finding at the present time, let alone at an indeterminate future date. If anything, the events PIIC cites demonstrate that the ROP is functional and adequately identifies and addresses safety issues at PINGP. Consequently, the events PIIC identified support the NRC

[14] Pursuant to 10 C.F.R. § 54.33(b), each renewed license will contain the conditions and limitations the NRC finds "appropriate and necessary" to ensure that systems, structures, and components within the scope of license renewal will perform their intended functions. The steps a licensee will take to manage the effects of aging during the period of extended operation will be included in the licensee's next FSAR update. 10 C.F.R. § 54.37(b). Such regulatory requirements would clearly be included in the plant's licensing basis during the period of extended operation. *See* 10 C.F.R. § 54.3(a) (defining a plant's licensing basis to include the plant's most recent FSAR). Therefore, the Safety Culture Contention essentially alleges that the Applicant will be unable to comply with its licensing basis during the period of extended operation.

Staff's finding that there is reasonable assurance that the Applicant will adequately manage the effects of aging at PINGP because the Applicant has committed to undertake those activities under the ROP. The following section of the Staff's position will explain how the ROP functions, how the NRC has evaluated the inspection findings at PINGP under the ROP, and how these findings support the NRC's finding of reasonable assurance.

A. Overview of the Reactor Oversight Process

The ROP is a risk-informed process for inspecting and assessing licensee performance. NRC Staff Testimony of John Giessner Concerning the Safety Culture Contention and the Reactor Oversight Process ("Giessner") at A5.[15] A risk-informed process assesses the safety risk posed by a non-compliance and uses that risk to drive the level of regulatory response. Giessner at A6. The primary mechanism the NRC uses to direct its oversight under the ROP is the Action Matrix. Giessner at A20. The NRC assigns each plant to one of five columns in the Action Matrix, based on its performance. Giessner at A22. If a plant demonstrates degraded performance, the NRC will move it to a higher-numbered Column in the Action Matrix. Giessner at A22. That reassignment will result in additional inspections. Giessner at A21. In Columns I-IV[16] of the Action Matrix, the NRC has reasonable assurance that the plant will operate safely. Giessner at A24. This reasonable assurance arises from a combination of the plant's operating

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[15] Mr. John (Jack) B. Giessner is Chief of Reactor Projects Branch 4 in the Division of Reactor Projects in the NRC's Region III office in Lisle, Illinois. Mr. Giessner holds a B.S. degree in Physics from the United States Naval Academy and a Master's degree from the Naval War College. Mr. Giessner has twenty-six years of experience in the nuclear field. Prior to joining the NRC in 2004, Mr. Giessner served thirteen years in the nuclear Navy and worked in the commercial nuclear industry for nine years. At Salem Nuclear Generating Station, Mr. Giessner held a Senior Reactor Operating License and obtained a Shift Technical Advisor Qualification. Subsequently, while at Donald C. Cook, Mr. Giessner served as Engineering Director and Assistant Operations Manager. Mr. Giessner qualified as the senior site official (Emergency Director). Upon joining the NRC, Mr. Giessner served as Resident Inspector at Palisades Nuclear Power Plant, Senior Resident Inspector at Duane Arnold Energy Center, Acting Senior Resident Inspector at Indian Point Nuclear Generating Station, and Acting Senior Resident Inspector at Calvert Cliffs Nuclear Power Plant. In his current position, Mr. Giessner is the Branch Chief and supervisor for the day-to-day inspections and assessment at Prairie Island Nuclear Generating Plant, Fermi Nuclear Plant, and Palisades Nuclear Plant. In that capacity, Mr. Giessner reviews all inspections performed by the resident inspectors on site, including findings and violations. Mr. Giessner approves and/or concurs in all inspections and inspection reports for PINGP.

[16] Referring to the columns of the Action Matrix by Roman numerals is a short hand. The formal names for the columns are: I-Licensee Response Column; II-Regulatory Response Column; III-Degraded Cornerstone Column; IV-Multiple/Repetitive Degraded Cornerstone Column; V-Unacceptable Performance Column.

record and the inspections required by the Action Matrix. Giessner at A22. Thus, as a plant's performance degrades, the NRC retains its level of confidence that the plant will operate safely through supplemental inspections. Giessner at A21. If the NRC determines that a plant's safety performance is unacceptable, and that the NRC therefore lacks reasonable assurance that the plant can comply with its licensing basis, the NRC will move the plant to Column V and direct it to shut down. Giessner at A23.

The NRC bases its placement of a plant in the Action Matrix on two inputs: (1) inspection Findings and (2) licensee-identified Performance Indicators. Giessner at A17. When the NRC determines that a licensee has failed to follow a regulation or standard and the licensee should have foreseen the issue, the NRC considers the event a performance deficiency. Giessner at A22. When a performance deficiency is more than minor in safety significance, the NRC counts it as a Finding. In contrast, Performance Indicators are quantitative measures, reported by the licensees. Giessner at A7. Performance Indicators are the same for all plants and assess a plant's performance. They can range from information on the number of times a plant had to significantly reduce power (a reduction by more than 20%) to assessing the impact of unavailable equipment. Giessner at A8.

The NRC assigns a color to each Finding or Performance Indicator to reflect its significance. Giessner at A19. A Green determination indicates a very low safety significance, a White determination corresponds to a moderate safety significance, a Yellow determination reflects a substantial safety significance, and a Red determination marks a high safety significance. Giessner at A18. This determination can be based on qualitative or quantitative measures. Giessner at A19. When either a Finding or Performance Indicator relates to reactor operations and causes the loss of a safety function, the NRC may undertake detailed assessments that include the use of Probabilistic Risk Analysis ("PRA"). Giessner at A19. When the NRC relies on a PRA, it bases the significance determination on the probability of a core damage event (minus the baseline event) or change in core damage frequency ("CDF") caused by the Finding or a change in the large early release frequency ("LERF") caused by the Finding. *Id*. Thus, a Finding or Performance Indicator that increases the CDF by more than 10^{-6} per year is White, by more than 10^{-5} per year is Yellow, and by more than 10^{-4} per year is Red. *Id*. Likewise, a Finding or Performance Indicator that increases the LERF by more than 10^{-7} per year is White, by more than 10^{-6} per year is Yellow, and by more than 10^{-5} per year is Red. *Id*.

The NRC's quantitative measures for assessing significance reflect the safety goals for nuclear reactors promulgated by the Commission. Giessner at A20 (*citing* Safety Goals for the Operations of Nuclear Power Plants; Policy Statement, 51 Fed. Reg. 30029 (Aug. 4, 1986)). The quantitative measures used, CDF and LERF, act as a surrogate for the goals to protect human health established by the Commission. *Id*. (*citing* Inspection Manual Chapter ("IMC") 0308 (NRC Staff Exhibit No. ~~6~~NRC000008)). The NRC also uses such metrics to assess the acceptability of a change to the licensing basis of an operating plant. *Id*. The NRC will only accept such changes if the impact on the plant's CDF and LERF are small. *Id*. The philosophy behind using this approach to assess the significance of Performance Indicators and Findings is that these conditions, if uncorrected, would be equivalent to accepting "a de facto increase in the CDR and LERF metrics." IMC-0308. Thus, the numbers relied on by the NRC Staff in assessing the significance of an inspection finding directly correspond to the implementation of the Commission's safety goals.

NRC inspections assess the licensee in several predefined areas called strategic performance areas and cornerstones. Giessner at A12. The strategic performance areas and cornerstones are the baseline areas that the NRC must inspect to ensure that it makes an accurate assessment of plant safety within the ROP. *Id*. The strategic performance areas are reactor safety, radiation safety, and safeguards (*i.e.* security). *Id*. The cornerstones associated with these strategic performance areas are initiating events, mitigating systems, barrier integrity, emergency preparedness, public radiation safety, occupational radiation safety, and safeguards. *Id*. The NRC may conduct additional inspections, as circumstances require. The NRC conducts about two thousand hours worth of inspections at each power reactor per year. *Id*.

The number and severity of Findings and Performance Indicators, as well as their distribution among the cornerstones, determines a plant's location in the Action Matrix. Giessner at A22. If a plant has no findings or only Green Findings or Performance Indicators, then the NRC will place the plant in Column I. *Id*. If the plant has only one White Finding or Performance Indicator or two White Findings or Performance Indicators that are not in the same cornerstone, then the NRC will assign it to Column II. *Id*. The plant will move to Column III when the NRC finds it has one Yellow Finding or Performance Indicator or two White Findings or Performance Indicators in the same cornerstone. *Id*. Moving to Column III indicates a degraded cornerstone that poses a moderate impact to safety performance. *Id*. When the NRC discovers a Red Finding or Performance Indicator or the licensee has multiple degraded cornerstones (cornerstones with one Yellow or two White Findings or Performance Indicators), the plant will move to Column IV. *Id*. Plants in Column IV have a significant degradation in safety performance. *Id.* When the NRC determines that the plant's safety performance is unacceptable, the plant will move to Column V. Giessner at A23. Examples of unacceptable performance include multiple significant violations, a loss of confidence in the licensee's ability to maintain and operate the facility within the design basis, or a pattern of failure from the licensee management to effectively address significant concerns. *Id*. In that case, the NRC will order the plant to shut down, if it has not done so already. *Id*. The Action Matrix counts a Finding or Performance Indicator for one year or until an NRC Supplemental Inspection Team clears the finding, whichever is longer. Giessner at A27. An NRC Supplemental Inspection Team clears a finding by verifying that the licensee properly evaluated the cause of the finding, took appropriate corrective action to prevent recurrence, and evaluated whether the problem could exist elsewhere. *Id*.

In addition to Findings and Performance Indicators, the ROP also considers cross-cutting aspects. Giessner at A28. Cross-cutting aspects may be in three areas: human performance, problem identification and resolution, and safety conscious work environment.[17] *Id*. Cross-cutting aspects are the most significant contributors to performance deficiencies that result in Findings. *Id*. But, not all Findings have cross-cutting aspects. *Id*. For example, if a Finding rests on an issue that is not indicative of current performance, the NRC will not find a cross-cutting aspect. *Id*. During an assessment period, if the NRC finds more than four cross-cutting aspects in a single area, the NRC will determine that a theme has developed. Giessner at A30. If the license has not effectively addressed the NRC's concerns in that area, the NRC will determine that the licensee has a Substantive Cross-Cutting Issue ("SCCI") in the area. *Id*.

For example, in Inspection Report 2009-010 (ADAMS Accession No. ML092170122 (August 5, 2009) (~~NRC Staff~~Applicant Exhibit No. ~~59~~NSP000069), the NRC inspectors identified a violation of 10 CFR Part 50 Appendix B, Criterion III Design Control with respect to the Unit 2 component cooling water system. Reviewing the most likely cause of the performance deficiency that resulted in this Finding, the inspectors assigned the finding to the decision-making aspect, which is a part of the cross-cutting area of human performance.

Licensees should follow-up on SCCIs by placing the issues in their corrective action program, performing an analysis of the issues, and developing appropriate corrective actions. NRC Staff Testimony of Dr. Valerie E. Barnes, June Cai, Molly Jean Keefe, and Audrey L. Concerning Safety Culture and NRC Safety Culture Policy Development and Implementation (hereinafter cited by last name(s) of testifying expert or experts), at A32. An SCCI will remain open until closed by the criteria established by the NRC in the assessment letter identifying the SCCI are fulfilled. Examples of criteria for closure of an SCCI include: fewer findings with the causal factor and increased confidence in the licensee's corrective action program and the licensee's ability to correct the issue. Klett at A31.[18] Mid-cycle and end-of-cycle assessment

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[17] The NRC relies on cross-cutting aspects to evaluate safety culture under the ROP, as discussed below at III.B.2.

[18] Ms. Audrey L. Klett is employed as a Reactor Operations Engineer in the Performance Assessment Branch of the Division of Inspection and Regional Support in NRR. Ms. Klett holds a B.S. degree in Electrical

letters report on licensee progress in resolving SCCIs. *Id.*

When an SCCI is indentified by the NRC, the NRC will monitor the SCCI. *Id*. If the SCCI is not closed within 18 months, and the licensee has not made reasonable progress towards addressing the issue, the NRC will typically request that the licensee perform a safety culture assessment. Keefe at A26.[19] If the licensee refuses, the NRC can perform the assessment itself using Inspection Procedure 95003, Supplemental Inspection Procedure for Repetitive Downgraded Cornerstones, (NRC Staff Exhibit No. ~~24~~NRC000026). Keefe & Klett at A34. In addition, at that time, the regional office will work with the Executive Director's Office to determine what actions to take. Giessner at A31.

But, an SCCI does not move the licensee through the columns of the Action Matrix and it does not enhance the Finding's significance. Giessner at A30. The ROP is built on the philosophy that inspection Findings and Performance Indicators move a plant through the columns of the Action Matrix, as discussed above, because they are indicative of degraded performance. *Id*. In contrast, SCCIs *may* indicate that a plant is more susceptible to degraded

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Engineering. Ms. Klett joined the NRC in 2003 as a Reactor Engineer Inspector in the NRC's Region III Office, performing inspections in the areas of problem identification and resolution, plant design, modifications, and fire protection, and served a rotation in the Resident Inspector's office at the Point Beach Nuclear Power Plant. In October 2006, Ms. Klett became an Electrical Engineer in the NRR's Division of Engineering, where she prepared safety evaluations for license amendment requests and backfit determinations, developed inspection procedures for testing emergency diesel generators, and served as the technical lead for emergency diesel generator and cable testing issues. In May 2008, Ms. Klett assumed her current position where she has worked on a variety of issues related to the ROP program including developing ROP implementation training for inspectors, working on the development of the NRC policy statement on safety culture and substantive cross-cutting issues, and problem identification and resolution and supplemental inspection procedures. Currently Ms. Klett is the staff lead for the ROP's Operating Reactor Assessment Program, which includes the NRC's process for oversight of operating reactor license's safety culture and identification of substantive cross-cutting issues.

[19] Ms. Molly Jean Keefe is a Human Factors Specialist in the Health Physics and Human Performance Branch of the Division of Inspection and Regional Support in NRR. Ms. Keefe holds B.A. and master's degrees in sociology. Ms. Keefe joined the office of Nuclear Regulatory Research as a Human Factors Analyst in 2003. In that capacity, Ms. Keefe assisted in developing changes to the ROP to enhance oversight of safety culture, designed survey instruments for assessing safety culture, served on a working group to enhance and modify inspection process and procedures, developed safety culture training for NRC inspectors, participated in inspections of nuclear power plant human performance, organizational management, and safety culture at Davis-Besse, Salem and Hope Creek, Duane Arnold, and Palo Verde Nuclear Generating Plant. From 2007 to the beginning of this year, Ms. Keefe worked in the NRC's Office of New Reactors, where she worked on developing a new construction inspection program with an emphasis on safety culture. Currently, as a Human Factors Specialist in NRR, Ms. Keefe is the NRR contact for safety culture. Her responsibilities include developing a new definition and traits of safety culture for the NRC's draft safety culture policy statement and addressing issues associated with implementation of the policy statement that is now under development, as well as serving as the safety culture contract for regional inspection and assessment issues.

performance. As a result, SCCIs, while an important consideration, can never actually compromise the NRC's reasonable assurance finding because, in and of themselves, they do not reflect degraded performance. *Id.*

B. PINGP Inspection Findings

As mentioned above, PIIC relies on several recent events at PINGP to support its claim that, essentially, PINGP's safety culture is so deficient that the NRC cannot find reasonable assurance that the applicant will comply with its licensing basis during the period of extended operation. But, the NRC has considered all of these events under the ROP and concluded that these events do not undermine the NRC's finding of reasonable assurance that the applicant will comply with its licensing basis during the current term of operation. Thus, these events can hardly lead to a conclusion that the Applicant will fail to comply with its licensing basis, through an inadequate safety culture or otherwise, in distant years.

The first event PIIC relies on related to an out-of-position valve on 11 Turbine Driven Auxiliary Feedwater Pump ("TDAFWP"). Giessner at A42. As a result of the valve's position, a safety-related component, the Auxiliary Feedwater train, was inoperable for nearly 138 days. *Id*. at A44. Under PINGP Technical Specification ("TS") 3.7.5.B, if one Auxiliary Feedwater train is inoperable, it must be restored to operability within 72 hours. *Id*. Otherwise, the TS requires the Applicant to lower the plant's power output within certain time frames. *Id*. Because, the applicant did not restore the pump to operable status within the allowable time, the NRC found a violation of TS 3.7.5.B. *Id*. In evaluating the risk posed by the performance deficiency, the NRC found that the dominant core damage sequence was a scenario in which a control room fire resulted in abandonment of the control room, after which the 11 TDAFWP failed, and the operator failed to recover the pump. *Id*. at A46. The NRC concluded that the out-of-position valve resulted in an increase in the CDF of about $2x10^{-6}$ per year. *Id*. Thus, the NRC determined that the event was a White Finding. *Id*. at A45.

PIIC's second issue occurred during an October 29, 2009, shipment of radioactive material. Giessner at A48. Upon arrival, the recipient detected elevated radiation levels from the package that exceeded the applicable limit. *Id*. at A50. The NRC conducted a qualitative assessment of the risk posed by the shipment and determined that no members of the public were exposed to the package. *Id*. at A52. But, the Staff noted that the consequences could have been greater under less favorable circumstances. *Id*. Because the risk was more than minimal, the NRC concluded that the Finding was appropriately characterized as White. *Id*. at A51.

Third, PIIC points to a White Finding based on the design of Unit 2's Component Cooling Water ("CCW") system. Giessner, at A54. Specifically, the applicant did not design the CCW system in manner to protect it from the impact of a high-energy line break or seismic or tornado events. *Id*. at A56. This violated 10 C.F.R. Part 50, Appendix B, Criteria III, Design Control, which requires the Applicant to establish measures that assure that the design basis for safety-related functions of structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. *Id*. As of July 29, 2008, the Applicant had failed to implement design control measures to ensure that the safety-related function of the CCW system would be maintained following a high-energy line break. *Id*. The NRC performed a PRA for this Finding and determined that it resulted in a change to the CDF greater that 10^{-6} per year but less than 10^{-5} per year. *Id*. at A58. Thus, the NRC staff determined that it was a White finding. *Id*. at A57.

In addition, although not the subject of an NRC inspection finding, PIIC argues that the reactor refueling cavity leakage, identified by the NRC Staff during its license renewal review, also supports its claim. The reactor refueling cavity leakage is fully discussed above in the discussion of the Staff's license renewal review. In addition to its consideration of the issue during license renewal, the NRC Staff also determined whether it could form the basis for an NRC inspection Finding. *Id*. at A41. The NRC conducted an assessment of the leakage to determine whether it created an immediate safety issue, such as an impact to the containment structure, liner, or other required supports. *Id*. That assessment determined that the leakage has no immediate safety significance. *Id*. Moreover, previous evaluations have not revealed any degradation of the containment pressure vessel, concrete, or rebar due to the refueling cavity leakage. *Id*.

PIIC also points to NRC identified problems with the Corrective Action Process ("CAP") to support its claim. The NRC has made Findings with respect to the CAP, but determined those Findings had a significance level of Green. Giessner, at A39. The NRC reached this conclusion because the Finding was of very low safety significance in that the non-compliances resulted in no loss of safety function. *Id*. at A40. Specifically, the NRC concluded that the CAP was functioning, but that the Applicant could take some actions to improve the process. *Id*. at A35.

Finally, PIIC points to an SCCI in the area of Human Performance to support its contention. But, as discussed above, cross-cutting issues are neither inspection Findings nor Performance Indicators because they do not indicate degraded performance. Giessner, at A30. Rather, SCCIs may indicate that a plant is more susceptible to degraded performance. The NRC monitors SCCIs but does not use them to move a plant through the columns of the Action Matrix. *Id*. Hence, this is not the type of issue that can undermine the NRC's ability to find reasonable assurance that a plant will operate safely.

C. <u>The Findings PIIC Identifies Show that the Reactor Oversight Process is Robust</u>

PIIC essentially contends that in light of the issues it has identified, the safety culture at PINGP is inadequate to provide reasonable assurance that the Applicant will comply with its licensing basis during the period of extended operation. But, the Findings upon which PIIC bases the Contention do not demonstrate that the NRC cannot have reasonable assurance that the Plant will comply with its licensing basis. Rather, the Findings suggest that the ROP is working and that the NRC is capable of identifying problems as they occur at the plant and ensuring that the Applicant will take appropriate action to correct them.

As discussed above, the NRC bases its finding of reasonable assurance on rigorous oversight that includes approximately two thousand hours of inspection per year at each reactor. Giessner, at A12. The results of these inspections inform the level of oversight the NRC exercises over a given plant. *Id*. at A22. Consequently, even when a plant shows degraded performance, the NRC maintains its reasonable assurance that the plant will operate safely through increased oversight. Therefore, the NRC's finding of reasonable assurance does not arise from confidence that the applicant will always be in compliance with its licensing basis. Rather, the NRC's reasonable assurance comes from, in part, confidence that even when the applicant does not comply with the licensing basis, the NRC will identify the non-compliance and take appropriate measures.

When it promulgated the license renewal regulations, the Commission understood that not all licensees complied with their licensing bases at all times. 56 Fed. Reg. 64,945. Nonetheless, the Commission concluded that its ongoing oversight would suffice to ensure that plants would continue to comply with their licensing basis during the period of extended operation. *Id*. at 64,945-46. Should the applicant fail to adequately manage the effects of aging during the period of extended operation ("PEO"), for whatever reason, including an inadequate safety culture, the NRC will identify the deficient performance and implement the appropriate measures necessary to ensure the NRC continues to have reasonable assurance that the plant will operate safely. These appropriate measures may include an order to shut down, if the NRC no longer has reasonable assurance that the plant can operate safely. Giessner at A23. The inspection findings on which PIIC relies demonstrate that the NRC is capable of identifying problems as they arise and pursuing appropriate regulatory avenues to correct them. *Id*. at A42-60. This confirms the wisdom of the Commission's decision to rely on the NRC's ongoing oversight process to ensure continued compliance with the licensing basis during the period of extended operation.

D. Conclusion

In essence, the Safety Culture Contention rests on the assertion that in light of some prior performance issues at PINGP, the safety culture at that site is so inadequate that the NRC cannot find a reasonable assurance that the applicant will comply with the PINGP licensing basis during the PEO. But, the NRC Staff has assessed all of these events under the ROP, which is the process the NRC uses to determine whether it has reasonable assurance that a plant will comply with its licensing basis. That assessment has determined that none of the issues raised by PIIC, individually or combined, alter the NRC's finding of reasonable assurance that the applicant will comply with the current licensing basis. If these issues do not indicate that the Applicant cannot comply with its licensing basis now, then they hardly suggest that the Applicant will be unable to do so years in the future. Moreover, the NRC's identification and resolution of these issues indicate the oversight process is sufficiently robust and adequate to ensure that Applicant will comply with its licensing basis during the PEO.

III. Safety Culture and the Reactor Oversight Process

The foregoing section explained the ROP, how the NRC has evaluated the inspection findings at PINGP under the ROP, how these findings support the NRC's finding of reasonable assurance, and thus why PIIC's contention lacks merit. This section focuses on how safety culture fits into the ROP and how the ROP provides continuous oversight of licensee safety culture. This section then explains why the recent inspection findings cited by PIIC do not indicate either that PINGP's safety culture is currently inadequate or that PINGP's safety culture will be inadequate in the future and, therefore, why PIIC's contention lacks merit from a safety culture perspective. However, before turning to a discussion of how safety culture fits into the ROP, it is important to have some background on safety culture in general and the current status of the Commission's safety culture policy.

A. Safety Culture Background

"Safety culture" refers to those dimensions of an organization, including its underlying assumptions, values, and norms, which influence the behavior of the organization's members with respect to safety. Barnes at A4.[20] The NRC currently defines safety culture as "that assembly of characteristics and attitudes in organizations and individuals that establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance." *Id*.

There is no regulatory requirement that licensees establish and maintain a positive safety culture, that is, a culture that ensures that nuclear power plant safety issues receive the action warranted by their significance. Cai at A11.[21] There are, however, regulations explicitly prohibiting discrimination against employees who raise safety concerns and regulations requiring licensees to establish quality assurance programs. *Id.; see* 10 CFR § 50.7 "Employee

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[20] Dr. Valerie Barnes is employed by the NRC's Office of Nuclear Regulatory Research as a Senior Technical Advisor in Human Factors. Dr. Barnes holds B.S., master's, and PhD degrees in psychology. She has over 25 years of experience consulting and performing research related to human performance in complex technologies, particularly the nuclear industry. Prior to joining the NRC in 2005, Dr. Barnes managed or played a key technical role in numerous projects for the U.S. Department of Energy ("DOE"), NRC, and other private sector and government sponsors to enhance the reliability of the human contribution to system performance. Dr. Barnes has been involved in many inspections and event investigations addressing a variety of human performance issues, including management and organization and safety management assessments. Dr. Barnes has also worked at nuclear power plants, including the Millstone site in 1997-1998 while it was improving its safety conscious work environment and Davis Besse Nuclear Generating Station following the discovery of the degradation in the reactor pressure vessel head. She provided organizational analysis and consulting, technical assistance, and training and mentoring of plant personnel. She oversaw the first independent, third-party safety culture assessment the NRC required a nuclear plant to perform while at Davis Besse and, after joining the NRC staff, led the NRC's first safety culture assessment at Palo Verde in 2007. She was also the primary author of Inspection Procedure 95003, which specifies the NRC's methods for conducting and criteria for evaluating safety culture assessments at nuclear power plants. Currently, Dr. Barnes is involved in a number of activities related to the NRC's safety culture initiatives, including advising the cross-agency Safety Culture Policy Statement Working Group, providing technical oversight of a research project to evaluate the construct validity of safety culture concepts and potential quantitative measures, and advising NRR on the scientific and technical validity of an industry-proposed approach to assessing and monitoring safety culture at commercial reactor sites.

[21] June Cai is the Senior Safety Culture Program Manager in the NRC's Office of Enforcement ("OE"). Ms. Cai has a B.S. in applied psychology and a master's degree in psychology focused on human factors and applied cognition. Ms. Cai joined the NRC in 2002 as a Risk and Reliability Analyst in the NRC's Office of Nuclear Regulatory Research. From 2003-2007, Ms. Cai worked in NRR as a Human Factors Analyst. Her work in NRR included multiple onsite inspections of safety culture, safety conscious work environment, and human performance at nuclear power plants. It also included leading agency efforts to develop multi-stage training for inspectors on safety culture. Also while in NRR, Ms. Cai worked to improve agency treatment of safety culture and safety conscious work environment for operating reactors by enhancing the inspection and oversight program. In 2007, Ms. Cai joined OE where she has served as the assistant team lead of the Internal Safety Culture Task Force, and worked on the development of policies and activities to enhance the agency's treatment of licensee safety culture, including a Commission policy statement, common set of terminology for the nuclear industry, and enhancements to the revised ROP. Ms. Cai was involved in the safety culture assessment activities and review of corrective action plans for the Palo Verde 95003 inspection in Fall 2007. Currently Ms. Cai is responsible for advising, developing, and implementing activities related to supporting and improving the NRC's internal safety culture. She is also responsible for supporting the NRC's external safety culture activities in the oversight of licensees.

Protection," 10 CFR § 50.34 "Contents of Application; Technical Information," and 10 CFR Part 50 Appendix B "Quality Assurance Program Criteria Nuclear Power Plants and Fuel Reprocessing Plants." In lieu of specific requirements for safety culture, the Commission has issued two policy statements related to safety culture to inform licensees of the Commission's expectation that licensees foster development of and maintain a positive safety culture and a safety conscious work environment. Cai at A12. The Commission is currently working on an expanded policy statement on safety culture. *Id.* The Commission's expectations for nuclear power plant licensee safety culture are implemented through the ROP. *Id*. at A18. In light of lessons learned from the discovery of severe boric acid corrosion of the reactor vessel head at the Davis-Besse Nuclear Generating Station, a General Accounting Office report recommending that the NRC more fully inspect the area of safety culture, and concerns raised by members of Congress with respect to NRC oversight, consideration of safety culture was expressly incorporated into the ROP in 2006. Keefe & Klett at A21.

B. <u>Reactor Oversight Process Oversight of Safety Culture</u>

The ROP provides oversight of licensee safety culture though the ROP inspections program, which was described above, and the ROP assessment program. As explained above, the ROP framework for safety culture describes three cross-cutting areas: (1) Problem Identification and Resolution; (2) Human Performance; and (3) Safety Conscious Work Environment. Keefe & Klett at A23. Each of these cross-cutting areas contains two to four safety culture components called cross-cutting components. For example, three cross-cutting components are associated with the cross-cutting area of the problem identification and resolution: corrective action program, self- and independent assessments, and operating experience. Each component is subdivided into aspects related to the component. The chart at *Id*. shows the relationship between cross-cutting areas and the nine cross-cutting components that are considered during ROP baseline inspections.

1. Reactor Oversight Process Inspection Program Oversight of Safety Culture

The ROP's inspection program provides for continuous oversight of cross-cutting aspects of safety culture. Klett at A25. As explained above in Section II.A, under the ROP inspection program, when inspectors make a Finding, they evaluate whether any aspect of the nine-cross-cutting components of safety culture are applicable to the finding. Giessner at A28-A29; Keefe & Klett at A23. The NRC's determination to assign a cross-cutting aspect to an inspection finding, however, is separate and distinct from the NRC's determination of the safety significance of an inspection finding, i.e., a White finding is a White finding regardless of whether a cross-cutting aspect is assigned. Giessner at A28-A29. The ROP inspection program also includes "Problem Identification and Resolution" inspections under IP 71152, which include evaluations of the licensee's corrective action program, employee concerns program, safety conscious work environment, and periodic self-initiated and NRC-requested safety culture assessments. Klett at A25. This inspection is not intended to provide a conclusion about the licensee's safety culture overall. *Id.*

In addition to assigning cross-cutting aspects to inspection findings, the ROP inspection program monitors safety culture through inspections based on licensee performance. Keefe & Klett at A23. These inspections include NRC review of NRC-requested safety culture assessments. Keefe & Klett at A23. The inspection program procedures IP 95001, IP 95002, and IP 95003 provide for NRC review and evaluation of licensee efforts to identify and address safety culture issues. Klett at A19(b). IP 95001, which is usually performed when a licensee enters Column II, provides for NRC Staff verification of licensee root cause evaluations to determine if the licensee appropriately considered safety culture components and aspects. *Id.* IP 95002, which is usually performed when a licensee enters Column III, provides for independent NRC review of whether the licensee appropriately considered whether any safety culture component or aspect caused or significantly contributed to any risk-significant performance issue. *Id*. If the NRC determines that a weakness in safety culture caused or significantly contributed to a risk-significant performance issue, and the licensee failed to recognize it, the NRC will request that the licensee perform an independent safety culture assessment. *Id*. at A19. IP 95003 is performed when a licensee enters Column IV of the Action Matrix. This procedure provided for NRC staff review, evaluation, and verification of the results of the third-party safety culture assessment that the NRC expects licensees in Column IV to obtain. *Id*. at A19b.[22]

Thus the ROP inspection program continuously monitors cross-cutting components and aspects of safety culture, but identification of a cross-cutting aspects is separate and distinct from the safety significance of the inspection finding.

2. Reactor Oversight Process Assessment Program Oversight of Safety Culture

The Assessment Program is the ROP's process for monitoring and trending the cross-cutting aspects assigned to inspection findings to determine whether licensees are effectively correcting any performance issues related to cross-cutting aspects. Klett at A24. If however, as discussed in Section II.A above, a licensee accumulates four or more inspection findings with the same cross-cutting aspect within a one-year assessment period and the NRC is concerned about the scope of the scope of efforts or progress in addressing the issue, the NRC will identify a substantive cross-cutting issue. *Id*. at A27.

As discussed in Section II.A above, a substantive cross-cutting issue is a cross-cutting theme about which the NRC has a concern with the licensee's scope of efforts or progress in addressing a cross-cutting theme. *Id*. A cross-cutting theme is identified in the areas of Human Performance or Problem Identification and Resolution when four or more inspection findings are assigned to the same cross-cutting aspect within a one-year assessment period. *Id*.

[22] If the licensee refuses to obtain an independent or third-party safety culture assessment, the NRC will have a safety culture assessment performed. *See* IP 95003.

When the NRC identifies an SCCI, the NRC documents the SCCI in a biannual (mid- or end-of-cycle) assessment letter. Klett, at A31. The first assessment letter documenting the SCCI will summarize the SCCI, state how the NRC intends to monitor the SCCI, and the criteria that must be met to close the SCCI. *Id*. Prior to the next assessment letter, the NRC will determine whether the licensee has met the closure criteria. *Id*. In the second assessment letter identifying the same SCCI the NRC may request an additional licensee response either orally at the next public annual assessment meeting or at separate meeting or in writing. *Id*. If the SCCI is not closed by the third consecutive assessment letter, the NRC may request that the licensee perform an independent safety culture assessment. *Id*. The NRC reviews the results of the assessment and the licensee's response to the results and documents its conclusions in the next assessment letter. *Id*.

C. PIIC Has Not Demonstrated that PINGP's Safety Culture is Inadequate or Will Be Inadequate in the Future

PIIC has not demonstrated that PINGP's safety culture is or will be inadequate. PIIC's contention relies upon a few recent inspection findings to assert that PINGP's safety culture is inadequate and will be inadequate in the future. These performance deficiencies, however, are insufficient to demonstrate that PINGP's safety culture is inadequate now and cannot support a prediction that PINGP's safety culture will be inadequate in the future.

Individual performance deficiencies that result in inspection findings do not indicate inadequate safety culture. Simply put, Action Matrix Column position is based upon inspection findings and performance indicators, not safety culture considerations. Keefe & Klett at A38. Therefore, conclusions about safety culture cannot be based solely on Action Matrix Position. Jack Giessner testifies that none of the White Findings relied upon by PIIC support its assertion that PINGP's safety culture is inadequate. Ms. Klett concurs noting that in both the supplemental inspections performed in response to the White Finding on the turbine-driven auxiliary feedwater pump (05000282/2009011) (~~NRC Staff~~Applicant Exhibit No. ~~46~~NSP000034) and the supplemental inspection performed in response to the White Finding on the radioactive shipment (05000282(306)/2009015) (~~NRC Staff~~Applicant Exhibit No. ~~47~~NSP000036), the NRC found that the licensee properly considered whether a weakness in safety culture caused or contributed to the event and initiated a human performance improvement plan. Klett at A38(b).

Contrary to PIIC's assertion, the identification of an SCCI in the area of human performance does not indicate inadequate safety culture. Klett at A40(b). This is because SCCI's focus on only one cross-cutting area and (usually) only one aspect in that area. Thus, the NRC does not rely on an SCCI alone to indicate an overall inadequate safety culture.

Similarly the "concerns" in the September 25, 2009 Problem Identification and Resolution Inspection Report and PINGP's alleged failure to promptly and effectively correct reactor refueling cavity leakage do not establish that there is an inadequate safety culture at PINGP. Biennial problem identification and resolution inspections do not and are not intended to provide an overall assessment of a licensee's safety culture. Klett at A41(b). Mr. Giessner testifies that the NRC's overall conclusion is that PINGP's corrective action program is functional, but implementation lacked some rigor. Giessner at A35. There are concerns and there have been concerns in the past that PINGP's corrective action program has failed to identify and correct certain issues. These failures, however, have not been systemic and thus do not indicate that the program is not functioning. *Id*. Mr. Giessner further testifies that the reactor refueling cavity leakage is not an immediate safety issue in light of the results from the license renewal inspection on the impact of the leakage on nearby safety structures. *Id*. at A41. He testifies that the leakage must be corrected and that the PINGP has taken steps to correct the problem. Mr. Giessner's testimony with respect to the safety significance of reactor refueling cavity leakage and the steps PINGP has taken to address the leakage is supported by the testimony of Dr. Naus and Mr. Sheikh.

As further evidence that individual inspection findings and identification of an SCCI does not demonstrate that PINGP's safety culture is inadequate, Dr. Barnes explains what a safety culture assessment should include to increase the likelihood that conclusions drawn from it will be valid and useful. That is, a safety culture assessment should be conducted systematically and go beyond employee surveys by including, for example, review of documents and interviews, case studies of work groups that expressed more negative views on surveys or events that had a significant impact on the plant personnel, observations of meetings and the performance of work on site, and interviews of plant personnel from all levels of the organization. Barnes, at A5.

Finally, not only do PIIC's assertions, based on PINGP's current performance, fail to demonstrate PINGP's safety culture is currently inadequate; they fail to demonstrate that PINGP's safety culture will be inadequate in the future. Dr. Barnes testifies that although it is possible to maintain a positive safety culture and it is possible to improve safety culture, it is not possible to predict future safety culture. Barnes at A6-A7. The factors that can influence safety culture and their impact on future safety culture are simply not foreseeable. *Id*. at A9. Therefore, it is unlikely that an organization's current safety culture is predictive of its future safety culture, particularly over the long term (e.g., beyond 5 years into the future). *Id*.

Consequently, while the ROP provides continuous oversight of licensee safety culture, the performance deficiencies relied upon by PIIC to support its contention do not demonstrate that PINGP's safety culture is inadequate nor do they support a prediction that PINGP's safety culture will be inadequate in the future.

CONCLUSION

PIIC's claim that PINGP's safety culture is inadequate to provide the reasonable assurance required by 10 C.F.R. § 54.29(a)(1) is unsustainable. There is no factual or regulatory basis for PIIC's assertion that PINGP must demonstrate the adequacy of its safety culture for purposes of license renewal. The NRC's review of PINGP's aging management programs and the NRC's continuing regulatory oversight of PINGP provides reasonable assurance that PINGP is operating and will continue to operate safely and consistent with its licensing basis and NRC regulations. As discussed above, the oversight process is functional and has fully evaluated the events that support PIIC's contention. Moreover, that process will continue to monitor the safety culture at PINGP now and during the period of extended operation. Consequently, there is reasonable assurance that actions have been or will be taken to ensure that Prairie Island Nuclear Generating Plant will be operated in compliance with its current licensing basis during the period of extended operation. PIIC's contention is thus unsustainable.

Respectfully Submitted,

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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)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing NRC Staff's Initial Statement of Position on the Safety Culture Contention; NRC Staff Testimony of Richard A. Plasse Concerning Safety Culture Contention; NRC Staff Testimony of Abdul H. Sheikh and Dr. Dan J. Naus Concerning the Safety Culture Contention and the Reactor Refueling Cavity Leakage; NRC Staff Testimony of John Giessner Concerning the Safety Culture Contention and the Reactor Oversight Process; NRC Staff 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; and NRC Staff Exhibit Nos. 1 through 59, dated July 30, 2010, have been served upon the following by the Electronic Information Exchange, this 30th day of July, 2010:

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