UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
NORTHERN STATES POWER COMPANY	)	Docket Nos. 50-282-LR/ 50-306-LR
	)	
(Prairie Island Nuclear Generating Plant,	)	
Units 1 and 2)	)	

NRC STAFF TESTIMONY OF RICHARD A. PLASSE
CONCERNING SAFETY CULTURE CONTENTION

Q1. Please state your name, occupation, and by whom you are employed.

A1. My name is Richard A. Plasse ("Plasse"). I am employed as a project manager in the Nuclear Regulatory Commission's Office of Nuclear Reactor Regulation, Division of License Renewal. A statement of my professional qualifications is attached hereto.

Q2. Please describe your current responsibilities.

A2. I am a project manager ("PM") for safety reviews of license renewal applications. I am presently the safety PM for the Prairie Island Nuclear Generating Plant (PINGP) license renewal application and the Seabrook Station license renewal application.

Q3. Please describe your duties with respect to the NRC Staff's review of the PINGP license renewal application.

A3. As the PM, I am the principal point of contact in NRR for the safety review of the PINGP License Renewal Application (LRA). I coordinated the Staff's evaluation of the PINGP LRA and preparation of the Staff's Safety Evaluation Report (SER) with Open Items, which was issued to the public in June 2009. (NRC Staff Exhibit No. 2). In addition, I coordinated the Staff's final SER (Staff Exhibit No. 1), which was issued to the public in October 2009.

Q4. What is the purpose of your testimony?

A4. The purpose of my testimony is to explain the basis for the Staff's conclusion that there is reasonable assurance that the effects of aging will be managed at PINGP during the period of extended operations.

Q5. Have you reviewed the assertions made by the Prairie Island Indian Community in the contention it filed on November 23, 2009, which was admitted by the Atomic Safety and Licensing Board in this proceeding in amended form on January 28, 2010?

A5. Yes.

Q6. Do you agree with the contention's assertion that PINGP's safety culture is not adequate to provide reasonable assurance that PINGP can manage the effects of aging during the period of extended operation?

A6. No, I do not agree.

Q7. Please provide a summary of the bases for your position.

A7. Based on the Staff's safety review, onsite audits, and inspections, the Staff concluded that there is reasonable assurance that PINGP can manage the effects of aging during the period of extended operation. In Section 6 of the SER (NRC Staff Exhibit No. 1), the Staff wrote:

> The staff of the U.S. Nuclear Regulatory Commission (the staff) reviewed the license renewal application (LRA) for the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, in accordance with the NRC regulations and NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," dated September 2005. Title 10, Section 54.29, of the *Code of Federal Regulations* (10 CFR 54.29) provides the standards for issuance of a renewed license. Pursuant to 10 CFR 54.29(a), the Commission may issue a renewed license if it finds that actions have been identified and have been or will be taken, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis (CLB). On the basis of its review of the LRA, the staff determines that the requirements of 10 CFR 54.29(a) have been met.

Q8. What did the Staff do in order to come to its conclusion regarding reasonable assurance?

A8. The 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, dated September 2005, (SRP-LR) (NRC Staff Exhibit No. 4) and NUREG-1801, "Generic Aging Lessons Learned Report," Rev. 1, dated September 2005 (GALL Report). The SRP-LR guides the Staff's review by assigning responsibilities among staff technical organizations, and describing methods for identifying structures, systems and components subject to aging management review. The SRP-LR also identifies the ten elements of an effective aging management program. 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; and (10) operating experience. Each element is defined in SRP-LR Section A.1.2.3 "Aging Management Program Elements." The SRP-LR allows an applicant to reference in its LRA the aging management programs (AMPs) described in the GALL Report. 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 licensing period. License renewal applicants may reference the GALL Report to demonstrate compliance with the requirements of the license renewal rule. A license renewal applicant's use of an AMP identified in the GALL Report constitutes reasonable assurance that the applicant will manage the targeted aging effect during the period of extended operation. Similarly, a license renewal applicant's commitment to implement an AMP that the Staff finds consistent with the GALL Report constitutes reasonable assurance that it will manage the targeted aging effect during the renewal period.

The purpose of the Staff's review was to determine whether PINGP's AMPS are sufficient to manage aging for systems, structures, and components in specific environments and/or subject to specific stressors. During the Staff's in-office technical review, the Staff reviewed the LRA (as supplemented with additional information provided by the applicant) and supporting documentation based on NUREG-1800. Onsite audits and inspections were performed by NRC teams, composed of technical, program, and operational experts from the NRC and its consultants, to review onsite documentation supporting the application and to address any issues that came up during the Staff's review of the application.

The documents submitted in connection with the application were reviewed to determine if the applicant met the technical and regulatory requirements of the regulations. The applicant must identify those systems, structures, and components that are within the scope of license renewal and subject to an aging management review and must also identify applicable aging effects and describe the AMPs that it plans to use to manage aging.

- For AMPs for which the applicant claimed consistency with the GALL AMPs, the staff conducted either an audit or a technical review to verify the claim.
- For each AMP with one or more deviations, the staff evaluated each deviation to determine whether the deviation was acceptable and whether the modified AMP would adequately manage the aging effect(s) for which it was credited.
- For AMPs not evaluated in the GALL Report, the staff performed a full review to determine their adequacy. In its full review, the staff evaluated the AMPs against the 10 program elements defined in SRP-LR.

Based on the reviews, analysis, inspections, and audits described above the Staff determined that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis and that the requirements of 10 CFR 54.29(a) have been met.

Q9. Describe the PINGP AMPs and explain how they become part of PINGP's current licensing basis (CLB).

A9. In its LRA the Applicant described 43 AMPs that it relies on to manage or monitor aging for the PINGP Units 1 and 2. Twenty-nine are existing programs and fourteen are new programs. All existing programs are already part of the plant's CLB. All new programs are documented as commitments in Appendix A of the SER. Following the issuance of the renewed operating license, the summary descriptions of the AMPs and the final list of license renewal commitments provided in Appendix A of the SER will be incorporated into the PINGP updated FSAR as part of the periodic FSAR update in accordance with 10 CFR 50.71(e). When the FSAR is updated, the AMP commitments will become part of the CLB. Ultimately, all AMPs will be in the CLB: AMPs in the form of commitments will become part of the CLB as a result of the FSAR update and AMPs in the form of existing programs that are already part of the CLB will continue to reside there.

Q10. Please describe the role of the ACRS and its Subcommittee in the NRC's license renewal process.

A10. The Advisory Committee on Reactor Safeguards (ACRS) is statutorily mandated by the Atomic Energy Act of 1954, as amended. With respect to license renewal, the ACRS fulfills the requirement of 10 CFR 54.25 to review and report on all license renewal applications and to make recommendations to the Commission. The ACRS and its subcommittees are comprised of academic and scientific experts in various fields. They are structured to provide a forum where experts representing many technical perspectives can provide independent advice that is factored into the Commission's decision making process. The ACRS meetings are open to the public and any member of the public may request an opportunity to make an oral statement during the committee meetings. Transcripts of the meetings are maintained on the

ACRS website at www.nrc.gov. As is customary, the ACRS assigned one of its subcommittees to the PINGP license renewal application, to gather information, analyze relevant issues and facts, and formulate a proposed position and proposed action for consideration by the full committee. The ACRS subcommittee reviewed the PINGP license renewal application and associated documents during its subcommittee meeting on July 7, 2009, and, at that meeting, had the benefit of discussions with representatives of the NRC technical staff and the applicant. During the 568th meeting of the ACRS, in December 2009, the ACRS completed its review of the PINGP LRA and the Staff SER. During the December meeting, the ACRS members conducted detailed follow-up discussions on the reactor cavity leakage issue with the applicant and NRC technical staff. The discussions were focused on understanding the evaluations performed, actions taken by the applicant, and commitments planned to be performed by the applicant to address the refueling cavity leakage issue. The ACRS subsequently agreed with the Staff's conclusion by letter to the NRC Commissioner, dated December 10, 2009. During the ACRS and subcommittee meetings associated with the PINGP LRA review, no member of the public made any oral statements.

Q11. What did the ACRS conclude with respect to the Staff's SER for PINGP?

A11. By letter to the NRC Commissioner, dated December 10, 2009, the ACRS concluded that 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 CLB for the period of extended operation without undue risk to the health and safety of the public, and that the application for renewal should be approved. NRC Staff Exhibt No. 3.

Q12. The SER included an Open Item related to refueling cavity leakage at PINGP. Please explain why the Staff established refueling cavity leakage as an Open Item in its PINGP SER.

A12. An item is considered open if, in the staff's judgment, it has not been shown to meet all applicable regulatory requirements at the time of issuance of the SER. During the AMP audit, the Staff discovered an ongoing issue with water seepage, from the refueling cavity into the containment sumps. Based on its review of the information provided, the Staff determined that borated water was coming into contact with the containment vessel during refueling outages, and that portions of the containment vessel may remain wetted after refueling outages. At the time of issuance of the SER, the Staff did not have enough information to conclude that NSPM had identified the appropriate actions to effectively manage the effects of aging related to refueling cavity leakage during the period of extended operation for the containment vessel. This resulted in the establishment of the Open Item.

Q13. What does it mean to "close" an Open Item?

A13. The Staff closes an Open Item when the applicant has provided the necessary information to the Staff, which can support a conclusion by the Staff, that all applicable regulatory requirements are met for a particular Open Item.

Q14. The Staff closed the refueling cavity leakage Open Item on what basis?

A14. The Staff closed the refueling cavity leakage Open Item based on the Staff review of the commitments provided by the applicant to address the refueling cavity leakage issue. The Staff concluded that the applicant demonstrated that the effects of aging will be adequately managed so that the intended function(s) of the containment structure will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The Staff provided its analysis supporting closure of the refueling cavity leakage Open Item to the ACRS and its subcommittee.

Q15. What did the ACRS conclude with respect to refueling cavity leakage at PINGP?

A15. With respect to the reactor cavity leakage the ACRS heard presentations and analysis from the Staff and the applicant. The ACRS asked numerous follow-up questions during the presentations. The ACRS reviewed the Open Item closure documented in the Final SER and subsequently agreed with the Staff's conclusion. The ACRS concluded: "The programs established and committed to by NSPM provide reasonable assurance that the 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."

Q16. Did the ACRS or the ACRS subcommittee determine whether PINGP will actually implement the AMPs in the SER?

A16. No.

Q17. Did the Staff determine whether PINGP will actually implement the AMPs in the SER?

A17. The Staff's determination of the adequacy of the AMPs presumes that if a license is issued, the licensee will implement the AMPs in accordance with the renewed licensing basis. Upon issuance of the renewed license, any license commitments the applicant made in connection with license renewal will become part of PINGP's current licensing basis. The plant's adherence to its current licensing basis is routinely examined by regional NRC staff on a current and ongoing basis. Resident inspectors and other Regional Staff routinely conduct inspections and assessments to determine whether plants are in compliance with their current licensing basis and they will continue their inspections and assessments during the period of extended operations. Because the resident inspectors and Regional Staff will be conducting their inspections and assessments during the period of extended operations, there is no need for the Staff to try to determine now whether PINGP will actually implement the AMPs. The inspection and assessment process is designed so that if the AMPs are not implemented, the resident inspectors and Regional Staff will learn about it.

Q18. Does the NRC Staff verify the implementation of aging management programs made in connection with license renewal?

A18. Yes. 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". The Post-Approval inspection will examine a sample of the aging management programs to determine whether the licensee has implemented them.

Richard A. Plasse
Statement of Profession Qualifications

CURRENT POSITION

Project Engineer Division of License Renewal, Office of Nuclear Regulation, U.S. Nuclear Regulatory Commission, Rockville, MD

EDUCATION

B.S., Clarkson University, 1982, Chemical Engineering

SUMMARY

I have 28 years experience applying the theories, principles and practices of engineering to the design, construction and operation of nuclear power plants. This is exemplified by my 23 years of experience working in the NRC Headquarters as a DLR Project Manager, the Fitzpatrick Licensing Manager and Senior NRR Project Manager at the Fitzpatrick Nuclear power plant, and a Region I resident inspector at the Nine Mile Point and Fitzpatrick Nuclear power plants. Prior to working for the NRC and the utility, I worked at Norfolk Naval Shipyard (NNSY) for five years as an Assistant Chief Test Engineer (ACTE) and Shift Test Engineer (STE). To work in these positions, I had to complete formal qualification programs that involved instruction in mechanical plant systems, electrical plant systems, reactor physics, chemistry, radiation protection, maintenance, and integrated plant operations.

EMPLOYMENT

Developmental/Commercial Qualification

- Qualified as Emergency Director Aide for Emergency Plan at the Fitzpatrick Nuclear Power Plant, 1995
- Qualified as an NRC Resident Inspector, 1988
- Qualified as a Shift Test Engineer, 1983

NRC Career History (2008-current)

I have been a license renewal project manager (PM) for the last 2 years with the NRC, responsible for the PINGP and Seabrook projects. A license renewal review is a complex 22-30 month process, requiring coordination and review by multiple headquarters and regional offices. Of primary importance is establishing direct lines of communication with the Regional office and the applicant's site management so that the PM is fully informed on all significant activities regarding the assigned plant. The PM should be aware of all Regional and Headquarters activities concerning the plant that might impact the safety review. Further, the PM is expected to have frequent communication with the applicant's personnel and the NRR technical reviewers regarding activities associated with the safety review.

The PM must be ready at all times to inform management on the status, problems, and progress of all aspects of the review. In addition, the PM is responsible for transmitting information to and from the applicant and technical reviewers in a timely manner; for maintaining liaison with the Office of the General Counsel (OGC) with respect to potential hearings and review of the safety evaluation report (SER).

<u>COMMERCIAL CAREER HISTORY (1995-2008)</u>

Entergy/New York Power Authority (Fitzpatrick Nuclear Power Plant, Oswego, NY)

LICENSING MANAGER/SR NRR PROJECT MANAGER/SR LICENSING ENGINEER

I have 13 years experience working as a manager and a senior staff engineer in the Fitzpatrick licensing department. I was involved in daily decisions made to ensure safe plant operations and compliance with the operating license. I was recognized by senior management and peers for practical and effective methods of identifying and managing risks. As a leader in the licensing department and a former NRC resident inspector at the Fitzpatrick plant I was considered a role model for others regarding plant operational safety. I was a member of the plant on-site review committee (PORC) which reviewed all significant operational events to ensure all potential safety issues were resolved including appropriate corrective actions to prevent reoccurrence. I considered being a PORC member a serious responsibility and demonstrated awareness for all safety issues, accepting responsibility to address difficult issues; standing firm when necessary.

As a senior licensing engineer I was routinely "on call "24 hrs. to support the on-shift licensed SRO in reviewing plant events and issues to ensure appropriate operability and reportability aspects were met (i.e. 10CFR 50.72/50.73). In addition, I was responsible to review the daily condition reports for NRC reportability in the plant electronic database system (PCRS). I was a member of the corrective action review board which reviews all plant identified deficiencies to ensure proper screening for level of review, safety significance, and corrective actions.

I was qualified as an Emergency Director Aide for 13 years as part of the Fitzpatrick Emergency Plan organization. I was responsible to assist the Emergency Director (ED) in fulfillment of his duties during plant events when the site emergency organization is activated. Qualification required formal classroom training and periodic practice drills. I participated in numerous full participation and partial participation emergency drills. As ED Aide I was familiar with all aspects of the emergency plan organization, facilities, and implementing procedures. I provided the ED with changing plant conditions and anticipated changes in emergency action levels, event classification, and associated protective measures as appropriate (i.e. protective action recommendations, PARs). When not performing ED Aide duties I was called upon to perform the role of an observer or a controller to ensure proper drill performance and deficiency identification for continuing improvement. An individual selected to be an ED Aide required demonstration of an understanding of integrated plant operations, emergency operating procedures (EOPs), plant design, engineering principles, radiological conditions, and plant maintenance. ED Aides represented the facility in interaction with local, county, state, and federal government officials.

As the licensing manager I was responsible for maintenance of the operating license, state permits and related licensing basis documents that support NRC requirements for the Fitzpatrick plant. As the NRR project manager for Entergy I was the point of contact to the NRC project manager in NRR. I was responsible for ensuring technical accuracy and completeness of all NRC technical specification submittals docketed to NRC. I resolved any NRC questions or issues by teleconference and follow-up request for additional information as appropriate.

<u>NRC Career History (1987-1995)</u>

I have eight years of experience working as a NRC resident inspector. While working in these assignments, I helped develop, plan, and implement the overall NRC inspection program at the Fitzpatrick and Nine Mile Point nuclear power plants. As such, I have extensive first hand experience with NRC regulatory programs (design, licensing, operation, and regulation). I have successfully completed the formal NRC resident inspector training qualification for a BWR reactor design. As part of the NRC inspection process, I conducted numerous inspections to evaluate the adequacy of structures, systems, and components to verify correct system alignment and operational readiness. Conducting these inspections required me to develop a thorough understanding of the plant's licensing and design bases. During a rotation to the office of NRR, I reviewed staff SER inputs to various plant licensing actions to ensure the documents were high quality and followed NRC policy. I also facilitated the resolution of technical issues by proposing alternative courses of action, conducting brainstorming meetings, and negotiating solutions with stakeholders.

As the Resident Inspector at the Fitzpatrick and Nine Mile Point nuclear power plants, using my knowledge of reactor plant operations, testing and design, I have identified numerous issues that could have been risk-significant if they had not been corrected. Some of these issues resulted in several NRC enforcement conferences and level III violations. They include the following:

-Plant operation with ultimate heat sink temperature greater than FSAR design
-Failure to properly calibrate feedwater flow instruments resulting in operation greater than licensed thermal power
-Failure to meet design requirements for various degraded fire penetration barriers
-Failure to meet design requirements for ATTS relays
-Failure to meet design requirements for emergency service water system

At Fitzpatrick and Nine Mile Point nuclear power plants I have observed and responded to many plant transients including reactor scrams and Engineered Safety Features (ESF) actuations.

After each plant transient, I briefed NRC management and independently performed a post trip review to verify proper operation of plant equipment and systems. I also participated in several diverse multi-disciplinary teams, a number of emergency preparedness exercises, follow-up inspections at other utilities, and identified a variety of safety significant and insightful findings during these inspections.

Norfolk Naval Shipyard Career History (1982-1987)

ASSISTANT CHIEF TEST ENGINEER/SHIFT TEST ENGINEER

As an ACTE, I was assigned to plan and schedule complex overhauls and selected restricted availabilities on nuclear powered submarines. I directed the test engineering activities and served as the shipyard representative of the joint test group. An extensive two year training qualification program was required to learn the various electrical and mechanical plant systems. I successfully completed an eight-hour NAVSEA 08 administered examination. Additionally, I successfully completed an oral board examination administered by senior shipyard managers and representatives from NAVSEA 08 and Knolls Atomic Power Laboratory.

As a STE at NNSY, I was specifically responsible for the direct supervision of shipyard trades, engineers, and ships force personnel during the performance of complex plant evolutions and testing including: reactor startups, power range testing, and primary plant hydrostatic testing. In addition, I was responsible for establishing plant conditions to allow maintenance and testing to proceed in a safe manner, and performed tours of reactor plant and engineering spaces on a daily basis to ensure plant parameters were within specification, and that test equipment and support systems were installed and functioning properly.

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AFFIDAVIT OF RICHARD A. PLASSE

I, Richard A. Plasse, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Executed in Accordance with 10 CFR § 2.304(d)

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Dated at Rockville, Maryland
this 30th day of July, 2010

July 30, 2010

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NRC STAFF TESTIMONY OF
DR. DAN J. NAUS AND ABDUL H. SHEIKH
CONCERNING THE SAFETY CULTURE CONTENTION AND
THE REACTOR REFUELING CAVITY LEAKAGE

Q1. Please state your name, occupation, and by whom you are employed.

A1(a). My name is Dan J. Naus ("Naus").[1] I am employed as a Distinguished Research Staff Member at Oak Ridge National Laboratory ("ORNL") by UT-Battelle, LLC. A statement of my professional qualifications is attached.

A1(b). My name is Abdul H. Sheikh ("Sheikh"). I am employed as a Senior Structural Engineer in the Division of License Renewal ("DLR"), Office of Nuclear Reactor regulation ("NRR"), U.S. Nuclear Regulatory Commission ("NRC"). A statement of my professional qualifications is attached.

Q2. Please describe your current responsibilities.

A2(a). (Naus) I am involved in several areas related to continuing the service of nuclear power plant safety-related structures. I have led several research programs for the U.S. NRC, Division of Engineering, Office of Nuclear Regulatory Research, over the last twenty years. I have had responsibility for conducting concrete materials-related research addressing aging

[1] In this testimony, the sponsors of each numbered response are identified by their last name; no such designation is provided for paragraphs which are sponsored by all witnesses.

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. I am currently involved in license renewal activities for the U.S. NRC, DLR, related to assessments of the safety-related structures. This involves technical support in the civil/structural area and to date has included audit and reviews of aging management programs, aging management reviews, and time-limited aging analyses of license renewal applications at six nuclear power plants. I am also currently leading a program under the Department of Energy, Office of Nuclear Energy, Light Water Reactor Sustainability Program. As part of these duties, I serve 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.

A2(b). (Sheikh) I am responsible for performing safety reviews and onsite audits of nuclear power plant structures, including containment structures and various structural supports for license renewal applications. In the last 15 months, I have performed onsite audits and reviewed the structural sections, including containment structures, of the license renewal applications for 10 nuclear power plants. For the Pressurized Water Reactor (PWR) containments, I have principally reviewed the steel containment, liner plate, and concrete to ensure that they will remain leak tight and maintain their pressure retaining function during the period of extended operation. In addition, I evaluated the effect of borated water leakage on the structural integrity of the reactor refueling cavity and spent fuel pool structures. I have also been involved with the revision of the Generic Aging Lessons Learned (GALL) report related to PWR and boiling water reactor structures. As a part of my duties, I represent the NRC on the American Institute of Steel Construction (AISC) Committee N690 for Specification for Safety-Related Steel Structures for Nuclear Facilities.

Q3. Please explain your duties in connection with the Staff’s review of the License

Renewal Application ("LRA") submitted by Northern States Power Company ("NSP") for the renewal of the Prairie Island Nuclear Generating Plant ("PINGP"), Units 1 and 2, License Nos. DPR-42 and DPR-60.

A3(a) (Naus) As part of my duties, I provided technical support in the area of concrete containment aging to the DLR project team reviewing and evaluating the PINGP operating license renewal application. Since March 2009, I also participated in Advisory Committee on Reactor Safeguard (ACRS) meetings, conference calls, and an on-site audit related to the refueling cavity leakage at PINGP.

A3(b) (Sheikh) As a part of my official duties, I performed an on-site audit, prepared requests for additional information (RAI), and reviewed the license renewal application and additional documentation provided by NSP related to reactor cavity leakage. This included a special report prepared by the NSP concerning the reactor refueling cavity leakage and responses to RAIs. I also participated in several phone calls with PINGP personnel concerning the timeline for completing different commitments made by NSP to resolve the reactor refueling cavity leakage issue.

Q4. What is the purpose of your testimony?

A4. The purpose of this testimony is to summarize the Staff's review of the potential impact of leakage of borated water from the refueling cavity on the steel containment, concrete, and reinforcing steel bars. The Staff concluded that there is reasonable assurance that NSP's aging management program, including commitments to monitor and repair the reactor leakage, is sufficient to ensure that the steel containment vessel and reactor refueling cavity support structures can perform their intended function during the proposed license renewal period.

Q5. Describe the reactor refueling cavity leakage of borated water at PINGP Units 1 and 2.

A5. Intermittent leakage from the reactor refueling cavity has occurred at PINGP Units 1 and 2 since the late 1980's during outages. At those times, the refueling cavity is filled with borated water. Between 1987 and the present, the PINGP units have been refueled 16 times. Leakage indications typically begin 2-4 days after the refueling cavity is flooded and end approximately 3 days after the refueling cavity is drained. The leakage rate was initially on the order of 1 to 2 gallons per hour in the emergency core cooling system sump (Sump B) and ceiling of the regenerative heat exchanger room, which is located below the reactor cavity.

Q6. What is the potential significance of the reactor refueling cavity leakage from a safety perspective?

A6. (Naus) If the borated water reaches the steel containment vessel, it has the potential to initiate corrosion that could result in a reduction of the thickness of the containment vessel that in turn could reduce the structural margins associated with the capacity of the steel containment vessel to resist pressure build up in the unlikely event of an accident. Borated water can also result in erosion of the concrete materials, corrosion of the embedded steel reinforcement, and possibly impact the performance of the affected concrete structures.

Q7. In your experience, is this the only instance of reactor refueling cavity leakage at a pressurized water reactor?

A7. (Naus) No, refueling cavity leakage has occurred at other plants.

Q8. When did NSP first identify the reactor refueling cavity leakage?

A8. (Sheikh) Intermittent leakage in both Units 1 and 2 has been observed since 1987. The leakage was first documented in 1998 during the Unit 2 refueling outage.

Q9. What steps did NSP initially take to address the reactor refueling cavity leakage?

A9. PINGP Personnel have tried several sealing methods over the years to eliminate the leakage. During the 1987-1998 period, PINGP Personnel performed repairs to the Unit 1 reactor refueling cavity stainless steel plate welds. In 1998, the PINGP Personnel performed a non-destructive examination of the Unit 2 reactor refueling cavity liner plate, identified three pinhole leaks, and repaired the welds at these three locations. PINGP Personnel also performed an engineering evaluation in 1998 to determine the effects of borated water on the steel containment and concrete structure. In 2002-2003, PINGP Personnel sprayed a coating on the reactor refueling cavity stainless steel liner plate. Leakage was mitigated when the coating was applied properly. During the period from 2004-2008, PINGP Personnel applied caulk at the reactor internal stand embeds which stopped the leakage when applied properly. In 2006, the PINGP reviewed the 1998 engineering evaluation to assess exposure of containment vessel and structures from 1998 to 2006.

PINGP personnel removed grout from the Unit 1 Sump B to visually inspect the steel containment vessel, the 3 ½ inch thick plate below the grout, for corrosion. Grout from Unit 2 Sump B was removed in 2008 and PINGP personnel performed visual and ultrasonic (UT) examination of the 3 ½ inch thick plate below the grout for corrosion. In addition, the PINGP personnel took over 150 ultrasonic readings of the steel containment vessel thickness in the area of the expected leak path. All readings were found to be within the regulatory requirements with no evidence of corrosion.

Q10. How successful were those steps in halting the reactor refueling cavity leakage?

A10. (Naus) Sealing methods have stopped the leakage, but not consistently during 1998 to 2008.

Q11. How was the reactor refueling cavity leakage considered in the Staff's review of the PINGP LRA?

A11. During the aging management program audit the NRC Staff identified the ongoing issue associated with water seepage from the refueling cavity into the containment sumps. In RAI B2.1.38-2, dated November 5, 2008, the Staff requested that NSP provide information regarding the root cause analysis of the seepage, as well as corrective and preventive actions taken to correct the problem. In its response, dated December 5, 2008, NSP stated that the existing steel containment and structures monitoring aging management programs have taken corrective actions to address the reactor refueling cavity leakage. The Staff reviewed the response and was concerned that leakage from the reactor refueling cavity could potentially accumulate at the bottom of the steel containment and that the area could remain wetted even after the refueling outages.

The staff arranged a public meeting, which was held in Rockville, Maryland on March 2, 2009, to gain additional insight about the reactor refuelling cavity leakage. In this meeting, NSP provided additional information about the reactor refueling cavity leakage. After this meeting, the Staff requested on March 31, 2009, in a follow-up RAI B2.1.38, that NSP discuss its plan for assessing the current condition of the steel containment vessel and to explain how the IWE program, or a plant-specific program, will manage aging of the vessel, especially in inaccessible regions, during the period of extended operation. In a letter dated April 6, 2009, PINGP personnel responded to follow-up RAI B2.1.38 and reiterated earlier proposed corrective actions for permanently fixing the leakage during the October 2009 outage for Unit 1, and April 2010 outage for Unit 2. The response also included two commitments. The applicant committed to remove concrete from the lowest point of the containment vessel bottom head and assess the condition of steel containment, rebar, and concrete. The applicant also committed to visually inspect the areas where reactor cavity leakage has been observed during two consecutive refueling outages after the repairs to the reactor cavity are implemented.

On May 28, 2009, NRC Staff conducted a supplemental plant audit at PINGP related to potential degradation of reinforced concrete and the carbon steel plate of the containment vessel resulting from leakage of the borated water from the refueling cavity. This included discussions with PINGP Personnel and a review of pertinent documentation. As a result of the site audit, several areas were identified by the NRC audit team for further inquiry. Responses to the NRC audit team requests for additional information were provided by PINGP in a letter (L-PI-09-082) dated June 24, 2009, a conference call on July 22, 2009, and a follow-up letter (L-PI-09-092) dated August 7, 2009. The Staff reviewed the PINGP response in the letters dated June 24 and August 7, 2009, and found them acceptable because the applicant committed to remove concrete from Sump C and inspect the steel containment vessel and rebar for degradation. The bottom of Sump C is located at the lowest point of the containment and is a likely place for corrosion to occur as a result of the leakage. PINGP also committed to obtain concrete samples from locations known to have been wetted by borated water and to test them for compressive strength and to perform a petrographic examination

Q12. What documents did the Staff review to reach its conclusions regarding the reactor refueling cavity leakage?

A.12 (Naus) The NRC Staff reviewed the license renewal application, program basis documentation available during the on-site aging management program audit, and documentation provided in response to requests for additional information related to the refueling cavity leakage. The NRC Staff also reviewed documentation provided during an on-site audit related to potential degradation of reinforced concrete and carbon steel plate of the containment vessel resulting from leakage of the borated water from the refueling cavity (e.g., "Evaluation of Effects of Borated Water Leaks on Containment Reinforcing Bars and Carbon Steel Plate of the Containment Vessels at Prairie Island Units 1 and 2" – Report No. R-4448-00-01, March 2009; "Refueling Cavity Leakage, Event Date 1988-2008," Report No. RCE 01160372-01, Volumes 1 and 2; and "Prairie Island Refueling Cavity Leakage High Level Summary," – Summary Presentation dated May 28, 2009.) The NRC Staff also reviewed information provided at public meetings and meetings of the ACRS.

Q13. What aging management programs has NSP identified in the LRA that would manage the effects from the reactor refueling cavity leakage?

A13. (Naus) PINGP has agreed to a number of activities and commitments as outlined in my responses to Questions 18, 19, and 21. PINGP will utilize the Structures Monitoring Program and ASME Section XI, Subsection IWE Program to manage potential aging associated with leakage of borated water from the refueling cavity.

Q14. Did the Staff agree with NSP's assessment of the likely impacts of the reactor refueling cavity leakage, particularly with respect to the pH value of the leak and the length of time the vessel liner and concrete reinforcements were potentially exposed to the leakage?

A14(a). (Naus) The NRC Staff did not agree with the high pH value of the leakage initially provided by NSP in their assessments (i.e., 12.5). A review of the experiments conducted by NSP, that formed the basis of their conclusions regarding the resulting pH of the borated water after contacting concrete, found that those experiments were not representative of what would actually happen on the surface of the concrete. Also, equilibrium calculations performed at Oak Ridge National Laboratory indicated that at equilibrium with an excess of borate the highest pH obtained would be on the order of 8 to 9, which is more acidic than NSP's assessment. Measurements of pH of fluid collected in 1998 by PINGP indicated that the pH value was between 7 and 7.8. The pH value of the leakage has significance from a safety perspective because the more acidic the leakage, the greater the potential for corrosion. The Applicant's inspection and testing programs will address any corrosion resulting from the leakage. Therefore, the NRC's concern regarding the pH of the leakage has been resolved.

A14(b). (Sheikh) The Staff did not agree with the Applicant's assessment regarding the length of time the vessel liner and concrete reinforcements were potentially exposed to the leakage. There is a possibility that water leaked during the refueling outages and collected at the bottom head of the steel containment vessel and continuously wet the inaccessible surface of the steel containment. However, the Staff determined that NSP's commitment to excavate the concrete at the containment lowest point in Sump C and to inspect rebar and steel containment provides assurance that either the vessel has not experienced significant degradation, or that any existing degradation will be documented and reviewed for structural impacts prior to period of extended operation. The previous inspection of steel containment and rebar in Sump B of Units 1 and 2 did not reveal any signs of degradation and so provides assurance that the implementation schedule for containment vessel inspection by August 2013 for Unit 1 and October 2014 for Unit 2 is adequate.

Q15. Did the Staff find that NSP's aging management programs were sufficient to address the impacts of the reactor refueling cavity leakage during the period of extended operation?

A15. (Naus) After addressing this as Open Item 3.0.3.2.17-1 in the Safety Evaluation Report, the NRC Staff, in their presentation to ACRS on the License Renewal Safety Evaluation Report for Prairie Island Nuclear Generating Plant, Units 1 and 2, on December 3, 2009, noted that this item was closed based on PINGP commitments to permanently repair the refueling cavity leakage, remove concrete and perform ultrasonic testing of the containment vessel at a low point of the containment, inspect exposed steel reinforcement for degradation, and remove and test concrete from the wetted area.

Q16. Did the ACRS consider the refueling cavity leakage in its review of the PINGP LRA?

A16. (Sheikh) Yes. On July 7, 2009, an ACRS subcommittee meeting was held in Rockville, Maryland related to the PINGP license renewal submission in which both the applicant and the staff presented information and corrective actions planned to repair and monitor reactor refueling cavity leakage. On December 3, 2009, an ACRS PINGP License Renewal Meeting was held, in Rockville, Maryland. In a letter dated December 10, 2010 (ML093420316) (NRC Staff Exhibit No. 3), the ACRS stated that the Committee agrees with the Staff conclusion that inspections, evaluations, and commitments made by the Applicant are adequate to address the refueling cavity leakage issue.

Q17. What features of the aging management program supported the Staff's conclusion that it would adequately address the impacts from the reactor refueling cavity leakage during the period of extended operation?

A17. (Naus) The absence of indications of degradation to the steel containment vessel in areas where visual and ultrasonic inspections were performed, lack of indications that the steel reinforcement or structural concrete have been impacted, intermittent nature of the leakage, activities over the years to mitigate the problem and commitments of PINGP, support the Staff's conclusion that the Applicant's AMP adequately addresses the reactor refueling cavity leakage.

Q18. What commitments did NSP agree to perform to address the reactor refuelling cavity leakage?

A18. PINGP completed a root cause evaluation in April 2009 that identified the sources of leakage as embedment plates for reactor vessel internals stands and the rod control cluster assembly change fixture. In September/October 2009, PINGP personnel conducted repairs to the embedment plates for reactor vessel internals stands and the rod control cluster assembly change fixture in Unit 1. The PINGP personnel removed existing nuts, replaced them with blind nuts, seal welded the blind nuts to the baseplate, applied seal weld between the baseplate and embedment plate, and examined the weld by non-destructive examination methods.

Q19. How successful were these efforts in addressing the leakage?

A19. Repairs eliminated the embedment plate leakage source, resulted in no evidence of leakage in Sump B, but resulted in minor leakage (i.e., estimated at 0.05 gallons per hour) observed from the regenerative heat exchanger room ceiling after the refueling cavity had been flooded for 14 days. There was no evidence that the leakage was reaching the containment vessel.

In September/October 2009, PINGP Personnel also performed, (1) vacuum box testing on the reactor cavity liner plate seam welds with no leakage identified, (2) non-destructive examinations on fuel transfer tube welds with no indications, and (3) non-destructive examinations of liner to embedment plate fillet welds which identified one porosity indication that will be repaired during next refueling outage. To verify the condition of steel containment at Unit 1, PINGP Personnel again removed grout from Unit 1 Sump B, conducted non-destructive examinations of the containment vessel wall, and found the wall thickness measurements at or above nominal, no corrosion of steel reinforcement or containment vessel, and no evidence of wet areas or leakage. If planned repairs do not completely stop leakage, it will be entered into the Corrective Action Program and evaluated for impact on structural integrity and identification of additional actions that may be warranted.

Q20. Describe any additional tests and inspections NSP has committed to perform to address the impacts of the reactor refueling cavity leakage at Unit 2.

A20. (Naus) In 2010 several corrective actions will be implemented in Unit 2: (1) repair of reactor vessel internals stands and rod control cluster assemblage change fixture plates (2) repair of rod control cluster assemblage guide box embedment plates, (3) non-destructive examination of fuel transfer tube welds, (4) vacuum box testing of reactor cavity liner plate seam welds, (5) non-destructive examination of liner to embedment plate fillet welds, and (6) other repairs and testing resulting from evaluation of 2009 Unit 1 leakage. Additional activities planned by PINGP in 2011 for Unit 1 include: (1) repair of rod control cluster assemblage guide box embedment plates, (2) repair of liner to embedment plate fillet weld porosity indication, and (3) other repairs and testing resulting from evaluation of 2009 Unit 1 and 2010 Unit 2 repair results.

Q21. How will all of these actions ensure that the impacts from the reactor refueling cavity leakage are adequately managed during the period of extended operation?

A21. (Naus) NSP has committed to: (1) during the first refuelling outage following the refueling cavity leak repairs in each unit, remove concrete from sump C to expose an area of the containment vessel bottom head, conduct visual examinations and ultrasonic thickness measurements of exposed portions of the containment vessel, and assess the condition of the concrete and exposed steel reinforcement; (2) during two consecutive refueling outages following the refueling cavity leak repairs in each unit, visually inspect the areas where reactor cavity leakage has been previously observed to confirm that leakage has been resolved, and if not resolved, the issue will be entered into the Corrective Action Program and evaluated to identify additional actions to mitigate leakage and monitor the condition of the containment vessel and internal structures; and (3) during the first refuelling outage following refuelling cavity leak repairs in each unit obtain a concrete sample from a location known to have been wetted by borated water leakage from the refueling cavity and tested for compressive strength and subjected to petrographic examination to assess the presence of degradation, if any, resulting from the borated water exposure, and if degradation is identified it will be entered into the Corrective Action Program and evaluated for impact on structural integrity and identification of additional actions that may be warranted.

Q22. If those tests and inspections indicate that impacts from the reactor refueling cavity leakage are greater than expected, what further steps will NSP take?

A22. (Naus) Visual inspections will be performed of the areas where reactor cavity leakage has been previously observed to confirm that leakage has been resolved, and if not resolved, the issue will be entered into the Corrective Action Program and evaluated to identify additional actions to mitigate leakage and monitor the condition of the containment vessel and internal structures. Also, during the first refueling outage following the refueling cavity leak repairs in each unit, PINGP Personnel will obtain a concrete sample from a location known to have been wetted by borated water leakage from the refuelling cavity. They will test the component for compressive strength and subject it to petrographic examination to assess the presence of degradation, if any, resulting from the borated water exposure. If degradation is identified, it will be entered into the Corrective Action Program and evaluated for impact on structural integrity and identification of additional actions that may be warranted.

Q24. What is the Staff's conclusion regarding whether actions have been identified and have been or will be taken to manage the effects of aging related to refueling cavity leakage during the period of extended operation on the functionality of structures and components identified under 10 C.F.R. § 54.21(a)(1)?

A24 Based on the intermittent nature of the problem, lack of evidence indicating that erosion of the concrete or corrosion of the steel reinforcement is occurring, planned repair and inspection activities, and the commitments from NSP during the license renewal process, neither the containment integrity is compromised nor the structural integrity of the concrete impacted significantly. Actions have been identified to effectively manage the effects of aging related to refueling cavity leakage during the period of extended operation on the functionality of structures and components identified under 10 C.F.R. § 54.21(a)(1).

Abdul H. Sheikh. PE
Statement of Professional Qualifications

CURRENT POSITION

Senior Structural Engineer — Division of License Renewal, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Rockville, MD

EDUCATION

B.S., University of Engineering and Technology, Lahore, Pakistan, 1966, Civil Engineering
M.S., University of Southampton, United Kingdom, 1969, Structural Engineering

SUMMARY

Mr. Sheikh has over 30 years of experience in design, construction, operations support, modification, vulnerability assessment, and license renewal activities of nuclear power plants. He has been involved in structural design and construction of nuclear power plant containments, static/seismic/dynamic stress analysis, aircraft impact on existing and new nuclear power plants, steam generator and reactor head replacements as an engineer, engineering group supervisor, and senior technical specialist at USNRC, Bechtel Corporation, and at Canatom Limited, Montreal Canada. He is a Qualified Reactor Technical Reviewer at USNRC.

EMPLOYMENT

U.S. Nuclear Regulatory Commission, May 2004 - Present

May 2009 to Present – Senior Structural Engineer, Division of License Renewal, Office of Nuclear Reactor Regulation

Responsible for the review and onsite audit of the structural portion of the license renewal applications for the last 15 months. In this capacity, he performed audits and reviewed license renewal applications for 10 nuclear power plants. Specific area of review includes borated water leakage from reactor refueling cavity and spent fuel pools, and containment corrosion and degradation. He was also involved in the Generic Aging Lessons Learned (GALL) update for aging management of containments, structures, and supports.

May 2004 to May 2009 – Senior Structural Engineer, Division of Engineering, Office of Nuclear Regulatory Research

Mr. Sheikh's was primarily involved with the assessment of existing and new reactors for large aircraft impacts. This included detailed analysis and evaluation of spent fuel pool and containment structures for five new reactors and a representative sample of the 104 existing nuclear power plants. In addition, to his work for the aircraft impact, he was the project manager for the Spent Fuel Transportation Cask Performance Study (PPS) and Containment Capacity Studies projects. He also performed analysis and evaluation of a representative

Dan J. Naus
Statement of Professional Qualifications

CURRENT POSITION

Distinguished Research Staff Member	Material Science and Technology Division Oak Ridge National Laboratory (ORNL) Oak Ridge, TN

EDUCATION

B.S.: Civil Engineering from the University of Illinois, Urbana-Champaign, 1966.
M.S.: Theoretical & Applied Mechanics from the University of Illinois, Urbana-Champaign, 1968.
Ph.D.: Theoretical & Applied Mechanics from the University of Illinois, Urbana-Champaign, 1971.

SUMMARY

Dr. Naus has over 40 years' experience in teaching and conducting research related primarily to the performance of materials, with the majority of his research activities related to aging management of nuclear power plant concrete and concrete-related materials and structures. He has been active in international activities related to aging management of nuclear power plant concrete structures sponsored by the International Atomic Energy Agency, the Committee on the Safety of Nuclear Installations of the Nuclear Energy Agency, and the International Union of Laboratories and Experts in Construction Materials, Systems, and Structures (RILEM). Prior to joining ORNL in 1975, he was a materials research engineer for the U.S. Army Corps of Engineers in Champaign, Illinois and an instructor and research associate at the University of Illinois in Urbana-Champaign. A large part of his early professional career was related to materials-related investigations in support of specific applications related to the military. For the last 30+ years he has primarily been involved in research programs supported by the DOE related to development of the High-Temperature Gas-Cooled Reactor, Clinch River Breeder Reactor, and Gas-Cooled Fast-Breeder reactor, and under a series of NRC subcontracts that have addressed fracture-mechanics-related testing of large steel plates, aging management of nuclear power plant concrete structures, inspection of aged/degraded nuclear power plant containments, evaluation of the potential effects of phosphate on cementitious materials, and evaluation of the effects of elevated temperature on concrete materials and structures. Most recently he has been involved in an NRC activity related to evaluation of nuclear power plant license renewal applications in the area of structural components, and a DOE program looking at the longevity of nuclear power plant concrete structures.

EMPLOYMENT

Oak Ridge National Laboratory, 1975 – Present

Progression from Staff Member to Senior Staff Member to Distinguished Research Staff Member

- 2009 – Present Providing technical assistance for civil/structural review of license renewal applications and have been involved with seven nuclear power plants to date (NRC)
- 2009 – Present Providing technical assistance for license renewal civil/structural NUREG report development related to operating experience for spent fuel pools, reactor cavity refueling pool leakage, BWR torus corrosion, and concrete structures (NRC).
- 2009 – Present Leading research program addressing activities (e.g., material performance, inspection, modeling of performance, inspection, assessment, and repair) related to continuing the service of nuclear power plant concrete structures past 60 years (DOE).
- 2005 – 2010 Lead a research program compiling data and information on the response of concrete materials and structures to elevated temperature (NRC).
- 2005 – 2007 Lead a research program investigating the potential degradation of cementitious materials subjected to phosphate environments (NRC).
- 2004 – 2006 Lead a research program developing durability-based design criteria for a quasi-isotropic carbon-fiber-reinforced thermoplastic automotive composite (DOE).
- 2002 – 2005 Lead a research program investigating inspection of aged/degraded containments (NRC).

- 1989 – 2002 Lead a research program investigating aging of nuclear power plant concrete structures and providing guidance for management of their aging (NRC).
- 1986 – 1990 Lead a research study investigating the crack run-arrest behavior of light-water reactor pressure vessel materials (NRC).
- 1979 – 1980 Lead a research program investigating mechanical and physical properties of concrete at elevated temperature in support of development of the Clinch River Breeder Reactor (DOE).
- 1975 – 1985 Lead a series of research activities (e.g., materials development, grouted and non-grouted tendon concrete beam performance, material's response to mechanical and thermal loading, model tests, and instrumentation evaluation) related to development of the High-Temperature Gas-Cooled and Gas-Cooled Fast-Breeder Reactors (DOE).

U.S. Army Construction Engineering Research Laboratory, 1970 – 1975

Materials Research Engineer

- 1970 – 1972 Lead investigation of applicability of linear-elastic fracture mechanics to Portland cement concrete materials.
- 1972 – 1975 Lead a series of investigations related to structural materials evaluations for dome antennas, methods for achieving exposed aggregate concrete finishes, inflation formed concrete and structural plastic domes, and ballistic testing of fibrous concrete dome and plate specimens.

University of Illinois, 1966 – 1971

Research Associate/Instructor

- 1966 – 1968 Lead research looking at parameters affecting the fracture toughness of Portland cement concretes.
- 1968 – 1971 Taught statics, mechanics of deformable bodies, and mechanical behavior of solids courses.

PROFESSIONAL AFFILIATIONS

Sigma Xi (Research Society)
Member, American Concrete Institute Committee 349, Nuclear Structures
Member, American Concrete Institute Committee 365, Service Life Prediction
Member, International Federation of Structural Concrete, Working Group 2 Containment Structures and Working Group 3 Non-Nuclear Prestressed Concrete Pressure Vessels
Member, American Society of Mechanical Engineers Section XI Working Group on Concrete Containments
Past Chair, Technical Committee 160-MLN, Methodology for Life Prediction of Concrete Structures in Nuclear Power Plants, of the International Union of Laboratories and Experts in Construction Materials, Systems, and Structures (RILEM)
Fellow, International Union of Laboratories and Experts in Construction Materials, Systems, and Structures (RILEM)
Fellow, American Society of Civil Engineers
Fellow, American Concrete Institute
Professional Engineer, State of Illinois

SELECTED PUBLICATIONS

Over 300 publications, papers or technical reports have been published. Selected publications are provided below.

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D. J. Naus, C. B. Oland, and G. C. Robinson, "Testing Program for Concrete at Temperatures to 894 K," *Sixth Int. Conf. on Structural Mechanics in Reactor Technology,* Paper H1/5, pp. 91-105, Paris, France (August 1981).

D. J. Naus, *Prestressed Concrete Reactor Vessel Research and Development Studies at the Oak Ridge National Laboratory,* British Nuclear Energy Society Conference on Gas-Cooled Reactors Today, Bristol, United Kingdom (September 20-24, 1982).

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D. J. Naus and G. C. Robinson, "Testing of Small-Scale Cavity Closure Models in Support of Development of a 300 MW(e) Gas-Cooled Fast-Breeder Reactor, *International Journal of Pressure Vessels and Piping* 13, Applied Science Publishers, Vol. 14, No. 1, England (1983).

H. Ashar and D. J. Naus, "Overview of the Use of Prestressed Concrete in U.S. Nuclear Power Plants, *Nucl. Eng. and Design* 75, No. 3 (1983).

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B. R. Ellingwood and D. J. Naus, "Chapter 6, Aging Nuclear Structures," in *Modeling Complex Engineering Structures*, Ed. by R. E. Melchers and R. Hough, pp. 137-170, American Society of Civil Engineers, Reston, Virginia (2007).

D. J. Naus, "Aging Management of Nuclear Power Plant Concrete Structures – Overview and Suggested Research Topics" *Proceedings of CSNI Workshop on Ageing Management of Thick Walled Concrete Structures, Including ISI, Maintenance and Repair – Instrumentation Methods and Safety Assessment in View of Long Term Operation,"* Holiday Inn Congress Centre, Prague, Czech Republic, 1-3 October 2008.

D, J, Naus, "Inspection of Nuclear Power Plant Structures – Overview of Methods and Related Applications," ORNL/TM-2007/191, Oak Ridge National Laboratory, Oak Ridge, Tennessee (May 2009).

D.J. Naus, "Overview of Nuclear Power Plant Concrete Structures Aging and Its Management," *Journal of Materials* **61**(7), pp. 37-43 (July 2009).

D.J. Naus, "Structural Materials: Chapter 95. Concrete," *Comprehensive Nuclear Materials*, Elsevier (2009) (draft).

PRESENTATIONS

Over 185 technical presentations have been made. Selected presentations are provided below.

"Fracture Mechanics Applicability to Portland Cement Concrete," *Army Science Conference,* West Point, NY (June 21, 1972).

"Inflation Formed Fibrous Concrete Dome Structures," *Research in Progress on Plain and Reinforced Concrete,* American Concrete Institute Annual Convention, San Francisco, CA (April 5, 1974).

"Resin-Bound Aggregate Material Systems," *First International Congress on Polymer Concretes,* London, United Kingdom (May 6, 1974).

"Embedment Instrumentation for Prestressed Concrete Pressure Vessels," *Fourth International Conference on Structural Mechanics in Reactor Technology,* San Francisco, CA (August 15-19, 1977).

"An Overview of Experimental Results Obtained Under the Prestressed Concrete Nuclear Pressure Vessel Program at the Oak Ridge National Laboratory, *International Atomic Energy Agency Specialist Meeting on Vessel Concepts for Gas-Cooled Reactors,* Lausanne, Switzerland (October 23, 1978).

"Acoustic Emission Monitoring of Intermediate Pressure Vessels Tested Under the ORNL Heavy-Section Steel Technology Program," *ASTM Symposium on Acoustic Emission Monitoring of Pressurized Systems,* Fort Lauderdale, FL (January 26, l979).

"Grouted and Nongrouted Tendons for Prestressed Concrete Pressure Vessels," *Fifth International Conference on Structural Mechanics in Reactor Technology,* Berlin, Germany (August 13-17, 1979).

"Acoustic Emission Monitoring of Steel and Concrete Structural Elements with Particular Reference to Primary Nuclear Containment Structures," *International Conference on Acoustic Emission,* Anaheim, CA (September 12, 1979).

"Elevated Temperature Concrete Property Determinations in Support of Advanced Energy Systems Concepts," *Seminar,* University of California, Berkeley (February 22, 1982).

"Prestressed Concrete Reactor Vessel Research and Development Studies at the Oak Ridge National Laboratory," *British Nuclear Energy Society International Conference on Gas-Cooled Reactors Today,* Bristol, United Kingdom (September 22, 1982).

"Prestressed Concrete Pressure Vessels and Their Applicability to Advanced Energy Systems Concepts," *ASME 4th National Congress on Pressure Vessels and Piping Technology,* Portland, OR (June 21, 1983).

"CERL/ORNL Research and Development Programs in Support of Prestressed Concrete Reactor Vessel Development," *FIP/CPCI Symposia on Concrete Pressure and Storage Vessels,* Calgary, Alberta, Canada (August 27, 1984).

"Applications of High-Strength Prestressed Concrete Reactor Vessel (PCRV) for HTGR-SC/C Plant," *International Atomic Energy Agency Specialists' Meeting on Design, Criteria and Experience with Prestressed Concrete Vessels for Gas-Cooled Reactors,* Lausanne, Switzerland (December 6, 1984).

"Summary of the Oak Ridge National Laboratory Concrete Program," *ISPRA Establishment-Joint Research Center,* Ispra, Italy (August 14, 1987).

"Aging of Concrete Components and Its Significance Relative to Life Extension of Nuclear Power Plants," *9th International Conference on Structural Mechanics in Reactor Technology,* Lausanne, Switzerland (August 20, 1987).

"Evaluation of Aged Concrete Structures for Continued Service in Nuclear Power Plants," *ANS Topical Meeting on Nuclear Power Plant Life Extension,* Snowbird, UT (July 31-August 3, 1988).

"Considerations in the Evaluation of Concrete Structures for Continued Service in Aged Nuclear Power Plants," *American Power Conference,* Chicago, IL (April 25, 1989).

"Wide-Plate Tests Utilizing Prototypical and Degraded (Simulated) Pressure Vessel Steels," *10th International Conference on Structural Mechanics in Reactor Technology,* Anaheim, CA (August 17, 1989).

"Assessment of the Significance of Aging of Concrete Structures in Nuclear Power Plants," *119th Annual Meeting of the Minerals, Metals & Materials Society,* Anaheim, CA (February 22, 1990).

"Aging Management of Safety-Related Concrete Structures in Nuclear Power Plants (NPPs)," *The 1991 Pressure Vessels and Piping Conference,* San Diego, CA (June 24, 1991).

"Concrete Material Systems," International Atomic Energy Agency Interregional Training Course, *Safety Aspects of Aging and Maintenance in Nuclear Power Plant Operation,* Argonne National Laboratory, Argonne, IL (April 13, 1992).

"An Improved Basis for Evaluating Continued Service of Category I Concrete Structures in Nuclear Power Plants," *Technical Session 16 - Nuclear Plant Systems/Components Aging Management and Life Extension*, ASME Pressure Vessel and Piping Conference, New Orleans, LA (June 22, 1992).

"Towards Assuring the Continued Performance of Safety-Related Concrete Structures in Nuclear Power Plants,' *1993 American Society of Mechanical Engineers Pressure Vessel & Piping Conference,* Denver, CO (July 27, 1993).

"Structural Aging Program to Evaluate Continued Performance of Safety-Related Concrete Structures in Nuclear Power Plants," *BNES/ENS International Conference on Thermal Reactor Safety Assessment*, Ramada Hotel, Manchester, United Kingdom (May 23, 1994).

"Aging Management of Containment Structures in Nuclear Power Plants," Proc. of The Third International Conference on Containment Design and Operation, held October 20–21, 1994 at Toronto Hilton Hotel, Toronto, Ontario, Canada (October 20, 1994).

“Extending the Lifespan of Nuclear Power Plant Concrete Structures,” International Association for Bridge and Structural Engineers Symposium – Extending the Lifespan of Structures, San Francisco, California (August 24, 1995.).

“Management of Age-Related Degradation in Nuclear Power Plant Concrete Structures,” *Specialists Meeting on the Effectiveness of Methods for Detection and Monitoring of Age Related degradation in Nuclear Power Plants,* International Atomic Energy Agency, Bariloche, Argentina (October 19, 1995).

“Aging Management of NPP Reinforced Concrete Structures,” *Proc. of Sixth Symposium on Current Issues Related to Nuclear Power Plant Structures, Equipment, and Piping,* North Carolina State University in Raleigh, North Carolina (December 4, 1996).

“Service Life of Nuclear Power Plant Concrete Structures – From Condition Assessment to Service Life Estimations,” *American Concrete Institute Fall Convention*, Atlanta, Georgia (November 12, 1997).

“Factors Related to Aging Management of Nuclear Power Plant Containment Structures,” *Proc. of 6th International Conference on Nuclear Engineering,* Paper No. ICONE-6135, American Society of Mechanical Engineers/Japanese Society of Mechanical Engineers, San Diego, California (May 13, 1998).

“Aging of Steel Containments and Liners of Reinforced Concrete Containments,” Session 1.1D, American Society of Mechanical Engineers/Japanese Society of Mechanical Engineers Joint Pressure Vessels and Piping Conference, Sheraton San Diego Hotel and Marina, San Diego, California (July 27, 1998.

“Management of Aging of Nuclear Power Plant Containment Structures,” *Fourth International Conference on Engineering Structural Integrity Assessment*, Churchill College, Risley, Warrington, Cheshire, United Kingdom (September 24, 1998).

“An Overview of the Structural Aging Program,” Obayashi Corporation, Tokyo, Japan (April 22, 1999).

“An Investigation of Tendon Corrosion Inhibitor Leakage Into Concrete,” *International Conference on Life Prediction and Aging Management of Concrete Structures*, Bratislava, Slovakia (July 6,1999).

“Detection of Aging of Nuclear Power Plant Structures,” *OECD-NEA Workshop on the Instrumentation and Monitoring of Concrete Structures*, Tractabel Offices, Brussels, Belgium (March 22, 2000).

“Factors Related to Degradation of Nuclear Power Plant Concrete Structures,” *Proc. of International RILEM Workshop on Aging Management and Life Prediction of Concrete Structures*, Cannes, France (October 16, 2000).

“Aging Management of Concrete Structures to Help Assure Continued Safe and Reliable Nuclear Power Plant Operation,” Czech Technical University, Prague, Czech Republic (October 19, 2000).

“Overview of Report of Task Group Reviewing Activities in the Area of Ageing of Structures Used to Construct Nuclear Power Plant Fuel Cycle Facilities,” Nuclear Energy Agency, Organization for Economic Cooperation and Development, Paris, France (May 15, 2001).

“Inspection, Assessment, and Repair of Nuclear Power Plant Concrete Structures,” Proc. of OECD-NEA Workshop on the Evaluation of Defects, Repair Criteria, and Methods of Repair of Concrete Structures of Nuclear Power Plants, DIN Institute, Berlin, Germany (April 10, 2002).

“An Overview of Activities in North America Related to Aging Management of NPP Containments and Other Structures,” *Proc. Of Workshop on the Concrete Containments in the Swedish NPP*, Swedish Nuclear Power Inspectorate, Stockholm, Sweden (April 25, 2002).

“Nuclear Power Plant Concrete Structures – Aging Considerations,” *International Congress Challenges of Concrete Construction*, University of Dundee, Scotland (11 September 2002).

“Task Group Report on Aging of Concrete Structures Used to Construct Nuclear Power Plant Fuel-Cycle Facilities,” Proceedings of CSNI/RILEM Workshop on Use and Performance of Concrete in NPP Fuel Cycle Facilities,” Instituto Ciencias de la Construccíon, “Eduardo Torroja, Madrid, Spain (March 15, 2004).

“Continuing the Service of Nuclear Power Plant Civil Structures – A Review of Activities and Research Needs,”*6th International Conference Global Construction: Ultimate Concrete Opportunities*, University of Dundee, Dundee, Scotland, (July 7, 2005).

“Inspection, Assessment, and Repair of Nuclear Power Plant Concrete Structures,” Regional Technical Meeting on Containment and Civil Structures Ageing Monitoring of Nuclear Power Plants, Rosenergoatom Moscow, Russian Federation (September 15, 2005).

“A Review of the Effects of Elevated Temperature on Concrete Materials and Structures,” 14th International Conference on Nuclear Engineering, Intercontinental Hotel, Miami, Florida (July 17, 2006).

“Aging Management of Nuclear Power Plant Concrete Structures – Overview and Suggested Research Topics” *CSNI Workshop on Ageing Management of Thick Walled Concrete Structures, Including ISI, Maintenance and Repair – Instrumentation Methods and Safety Assessment in View of Long Term Operation,”* Holiday Inn Congress Centre, Prague, Czech Republic (October 1, 2008).

“Aging Management of NPP Concrete Structures: An Overview of Activities and Candidate Research Areas,” *Embedded Topical Meeting – NFSM for NGNR*, ANS Technical Session, San Diego, Ca, June 15, 2010.

sample of nuclear power plant containments for the beyond design basis accidents as member of the NRC's project team for the state of the art consequence analysis of nuclear power plants.

Bechtel Power Corporation, Frederick, Maryland, April 1979 – May 2004

Mr. Sheikh held a variety of assignments during the 25 years employment at Bechtel Power Corporation in the design, construction support, startup, and modifications of the power plants. The details of these assignments are as follows:

February 2002 - May 2004, Engineering Group Supervisor, Reactor Pressure Vessel Head Replacement Projects, Frederick, Maryland

Mr. Sheikh served as the Engineering Group Supervisor for Reactor Pressure Vessel Head Replacement (RPVH) Projects. In this capacity, he was responsible for the design and construction support activities for the North Anna Units 1 and 2, and Surry Units 1 and 2 RPVH projects. The design activities included heavy load drop analysis in accordance with NUREG 0612, rigging and transportation of the reactor pressure vessel heads, finite element analysis for creating a temporary construction opening in the containment and internal concrete structures, and liner plate. Previously, he prepared a detailed design report for the reactor head replacement project for Davis Besse Nuclear Power Plant.

August 1999 - February 2002, Engineering Group Supervisor
Mountainview Power Plant, California, and Athens Power Plant in New York

Mr. Sheikh was responsible for the development of the scope book, design criteria, quantities, detailed geotechnical and civil/structural design of Mountainview and Athens Power Plants. Each plant had a capacity of 1000 MW. I supervised and coordinated the work of a team of engineers and designers located in Frederick, Maryland, and New Delhi, India. The design work was performed round the clock in two separate locations, and included large structural steel turbine buildings, foundations for turbines, Heat Recovery Steam Generators (HRSG), cooling towers, transformers foundations. He was responsible for coordination and approval of all geotechnical, structural, civil, architectural, and mechanical equipment designs from the California Chief Building Official (CBO) in San Bernardino County for the Mountainview Power Plant.

June 1996 - August 1999, Senior Technical Specialist, Chief Civil Engineer's Staff, Gaithersburg, Maryland

Mr. Sheikh worked as a technical specialist in the Chief Civil Engineers Staff /Central Engineering Group. In this assignment, he was also involved with the off project independent review of the structural design calculations for the several nuclear steam generator replacement projects. He was responsible for the design and analysis of complex structures such as prestressed concrete containments, chimney stacks, and turbine foundations. he participated in the development of proposals and estimates for decommissioning of the Main Yankee and Connecticut Yankee Nuclear Power Plants.

February 1996 - June 1996, Resident Engineer, GINNA Nuclear Power Plant, New York

Mr. Sheikh was part of a team of engineers that provided engineering support at the GINNA Nuclear Power Station during steam generator replacement activities. He was the responsible for processing and approval of field changes.

June 1989 - June 1996, Engineering Group Supervisor, Nuclear Operations Group, Gaithersburg, Maryland

Mr. Sheikh served as the Civil/Structural Engineering Group Supervisor for Nuclear Operations Group for six years. This group was responsible for preparing design modification packages and safety evaluation reports for different nuclear power plants, including Farley, Hatch, Vogtle, Perry, Fitzpatrick, Arkansas, Robinson, and Brunswick Nuclear Units. The scope of work included seismic analysis, equipment qualification, containment tendon surveillance, modification and evaluation of nuclear plant structures, design of security/safety barriers for protection against malevolent vehicles, tornado missile impact analysis on safety related structures. In addition, the group also provided support for operability evaluations and construction.

January 1987 - June 1989, Resident Engineer, Vogtle Nuclear Power Plant, Augusta, Georgia

Mr. Sheikh was part of the resident engineering group that prepared design change packages during the plant operations for Unit 1 and startup activities for Unit 2. He also prepared detailed design for low radwaste storage facility, and installation of high-density fuel storage racks in the spent fuel pool. He supported the in service inspection activities for the piping support snubbers during the first outage for Unit 1, and prepared detailed procedures for the surveillance of the prestressed steel containment.

May 1981 - January 1987, Engineering Group Leader/Engineering Group Supervisor, Korea Nuclear Units 7 and 8, in Los Angeles, Seoul Korea, and jobsite at Kwanju Korea

Mr. Sheikh was the Engineering Group Leader for Containment Structures, and the Civil Structural Engineering Group Supervisor for Korea Nuclear Units 7 and 8. In these assignments, he participated in conceptual design, detailed design, procurement, construction support, and startup support in Los Angeles, California and Seoul Korea, including at the remote jobsite in Kwang-Ju, Korea. He participated in the preparation of Preliminary Safety Analysis Report (PSAR) and Final Safety Analysis Report (FSAR), and prepared Section 3.8 and provided input for other sections of the FSAR for the Korea Nuclear Units 7 and 8 power plants. At the job site, he was responsible for the construction support and coordination activities.

April 79 - May 1981, Senior Engineer, Palo Verde Nuclear Power Plant, Los Angeles, California.

Mr. Sheikh prepared the detailed design calculations for the reactor internal concrete structure, and structural steel platforms. He also designed the pipe whip restraints for the high-energy line piping systems utilizing the energy from plastic deformation of stainless steel rods to resist the dynamic loads.

Canatom Limited, Montreal, Canada, February 1975 - April 1979

Mr. Sheikh was responsible for a small team of engineers who prepared detailed design calculations for Wolsung Nuclear Power Plant in Korea, and Gentilly 2 Power Plant in Quebec, Canada. He also prepared detailed design of the containment prestressing system, and Calandria Vault, which houses the reactor. He was also involved with construction support and made site visits to Gentilly, Quebec, and Korea.

Rendel, Palmer and Tritton Consulting Engineers, London, February 1973 - February 1975

Mr. Sheikh prepared detailed design of the heavy prestressed concrete sill beams that support the gates for Thames Barrier Project, London. This included finite element analysis and coordination for hydraulic modeling performed at the Imperial College to establish the design criteria for the unique structure.

Milton Keynes Development Corporation, United Kingdom, March 1970 - February 1973

Mr. Sheikh prepared detailed design calculations and drawings for highway bridges. In addition, he worked at the jobsite as an Assistant Resident Engineer.

PROFESSIONAL AFFILIATIONS

- Registered Professional Civil Engineer in the State of California (Active)
- Chartered Civil Engineer, England (Inactive)
- Member, American Institute of Steel Construction Committee N 690, Specification for Safety-Related Steel Structures for Nuclear Facilities
- Past Member American Concrete Institute Code Committee ACI 351, Foundations for Equipment and Machinery

PUBLICATIONS AND PRESENTATIONS

- Sheikh A., (2007, August), A Simplified Approach for Predicting Containment Performance During a Severe Accident, Proceedings of 19th International Conference on Structural Mechanics in Reactor Technology, Toronto, Canada

- Sheikh A., (1999, April), Design of Axial Exhaust Turbine Foundations, Proceedings of American Power Conference, Chicago, Illinois

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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 COMPANY	)	Docket Nos. 50-282-LR/ 50-306-LR
	)	
(Prairie Island Nuclear Generating Plant,	)	
Units 1 and 2)	)	

AFFIDAVIT OF ABDUL H. SHEIKH

I, Abdul H. Sheikh, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Executed in Accordance with 10 CFR § 2.304(d)

Abdul H. Sheikh
Senior Structural Engineer
Aging Management of Structures,
Electrical, and Systems Branch
Division of License Renewal,
Office of Nuclear Reactor Regulation
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Dated at Rockville, Maryland
this 30th day of July, 2010

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
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NORTHERN STATES POWER COMPANY	)	Docket Nos. 50-282-LR/ 50-306-LR
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(Prairie Island Nuclear Generating Plant,	)	
Units 1 and 2)	)	

AFFIDAVIT OF DR. DAN J. NAUS

I, Dan J. Naus, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Executed in Accordance with 10 CFR § 2.304(d)

Dr. Dan J. Naus
Distinguished Research Staff Member
Oak Ridge National Laboratory
Post Office Box 2008 MS6069
Oak Ridge, TN 37831-6069
(865)-574-6098
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Dated at Oak Ridge, TN
this 30th day of July, 2010

July 30, 2010

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
NORTHERN STATES POWER COMPANY	)	Docket Nos. 50-282-LR/ 50-306-LR
	)	
(Prairie Island Nuclear Generating Plant,	)	
Units 1 and 2)	)	

NRC STAFF TESTIMONY OF JOHN GIESSNER
CONCERNING THE SAFETY CULTURE CONTENTION AND
THE REACTOR OVERSIGHT PROCESS

Q1. Please state your name, occupation, and by whom you are employed.

A1. John (Jack) B. Giessner, a branch chief and supervisor at the United States Nuclear Regulatory Commission (NRC) and am responsible for oversight of inspections at the Prairie Island Nuclear Generating Plant (PINGP). A statement of my professional qualifications is attached.

Q2. Please describe your current responsibilities.

A2. I am a branch chief and supervisor responsible for the day-to-day inspections and assessment at PINGP, Fermi Nuclear Plant, and Palisades Nuclear Plant. My job is to ensure the operators of these plants operate their plants safely in a manner that complies with the NRC regulations and preserves the health and safety of the public and the environment. Each site, including PINGP, has two dedicated inspectors who are assigned to the plant, live in the area, and report to me directly on a daily basis. In addition, there are staff personnel in the regional office of Lisle, Illinois (Region III), who provide inspection and assessment support. Some of these personnel report directly to me, others report to branch chiefs who work in a specialized area. However, I am temporarily assigned to the Office of the Executive Director for Operations as an Executive Technical Assistant.

Q3. Please explain your duties in connection with the Staff's ongoing oversight of the PINGP, Units 1 and 2, operated by Northern States Power Company ("NSP") pursuant to License Nos. DPR-42 and DPR-60.

A3. I am the supervisor responsible for the day-to-day inspections and assessment at PINGP. My job is to ensure the operators of this plant operate the plant safely in a manner that complies with the NRC regulations and preserves the health and safety of the public and the environment. I review all inspections performed by the resident inspectors, including Findings and violations at the site; I also approve, and/or concur in, all inspections and their reports for PINGP. I am in contact with plant management several times a month and meet and discuss performance with them every couple months.

Q4. What is the purpose of your testimony?

A4. The purpose of my testimony is to explain how the reactor oversight process (ROP) works, how the ROP fits in with the NRC's mission to ensure safe and secure operation of the plant while preserving the health and safety of the public and the environment, how we assess PINGP in this process, and some of the findings and our current activities.

Q5. Please describe the ROP?

A5. The ROP is a risk-informed objective process for inspecting and assessing licensee performance.

Q6. What is a risk-informed process?

A6. A risk informed process takes into account the risk of not complying with standards or regulations. Thus all non-compliances are not treated equally under the ROP; the ROP finds some are more significant than others based on the risk posed by those non-compliances. Under the ROP, as explained in greater detail below, the risk of a non-compliance drives the level of response by the NRC.

Q7. Do the licensees have any input into this process?

A7. In addition to inspections, the licensees also provide to the NRC quantitative performance Indicator (PI) information. The PIs provide objective data on conditions at the site and are consistent among plants.

Q8. Please describe what a licensee-provided performance indicator is?

A8. The PI's provided by the licensee are an agreed upon set of indicators all sites voluntarily provide to us to assess plant performance in different performance areas. They are the same for every plant. The inspectors review and validate that the information provided is accurate using a standard procedure. Some of the information is very straightforward; for example, the number of times the plant has to reduce power significantly (>20%). However, some PIs are more complicated and may, for example, assess the impact from the unavailability of certain equipment based on the risk that equipment's unavailability poses to the system it supports.

Q9. Under the ROP, what actions does the NRC take in response to the licensees' performance?

A9. The process gives a graduated series of responses to licensees' performance when our assessment process determines more oversight is warranted. Prairie Island Annual Assessment Meeting (May 20, 2010) (NRC Staff Exhibit No. 5), Slide 14, provides a high level overview of how NRC inspection results and licensee-provided PIs are taken into account in the ROP's assessment of plant performance.

Q10. Why was the ROP developed?

A10. As stated in Inspection Manual Chapter (IMC) 0308, "Reactor Oversight Process (ROP) Basis Document," at ¶ 0501, (June 25, 2004) (NRC Staff Exhibit No.6)

> "On April 2, 2000, the NRC implemented a new ROP at all

> operating commercial nuclear power plants. The objectives of the staff in developing the various components of this new oversight process were to provide tools for inspecting and assessing licensee performance in a manner that was more risk-informed, objective, predictable, and understandable than the previous oversight processes. The ROP was also developed to meet the four agency performance goals to: (1) maintain safety, (2) increase openness, (3) make NRC activities and decisions more effective, efficient, and realistic, and (4) reduce unnecessary regulatory burden."

IMC 0308 provides a history of the NRC's assessment and actions which led to the ROP.

Q11. What is the purpose of the ROP?

A11. The purpose of the ROP is to provide an objective and risk-informed approach to inspecting licensees and assessing licensee performance and to provide a graduated response to issues that arise in licensee performance. Although the old process had elements of risk incorporated in it, it was not as objective and predictable as the new process is. Risk informed is distinguished from risk-based. This is described in detail in IMC 0308, Reactor Oversight Process (ROP) Basis Document," Attachment 6 "Significance Determination Process Basis Document," (Jul. 28, 2005) (NRC Staff Exhibit No. 7):

> The reactor safety [Significance Determination Process] SDP process is considered risk-informed, not risk-based, and supportive of the Commission Policy on Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities (1995). As defined in SRM SECY-98-144 revision 1, dated March 1, 1999, a "risk-based" approach to regulatory decision-making is one in which such decisionmaking is solely based on the numerical results of a risk assessment. Under this definition, the approach taken by the ROP (for both PIs and the SDP, where appropriate) might be considered "risk-based." However, the SDP is considered risk-informed by virtue of the expectation that SDP result bases are sufficiently understood by those technically knowledgeable persons (such as inspectors and technical staff) who are best positioned to critically examine the most influential probabilistic and technical assumptions, as well as by the management decision-makers who ultimately make the decisions. Conversely, if decisions are made without an understanding appropriate to the objectives of the ROP, they are risk-based.

The risk-informed approach, as discussed in the above mentioned SRM, should also consider other factors. Historically, these other factors can include defense in depth, safety margins, and consideration on reliance of operator actions.

Q12. How often are inspections conducted?

A12. First, inspection frequency is determined using a risk-informed baseline inspection program. These risk-informed baseline inspections are the inspections all operating reactors receive. They are detailed in the inspection manual which provides the inspection policy, IMC 2515, Light Water Reactor Inspection Program-Operations Phase (IMC 2515) (NRC Staff Exhibit No. 8). The inspection policy provides the frequency and approximate times inspections take. Some items are required to be done frequently, for example on a daily basis, such as the action to review adverse conditions, called condition reports, the licensee has written. There are many other inspections that assess the licensee's performance in the strategic performance areas and cornerstones.

Strategic performance areas and cornerstones are different areas that must be inspected to ensure all aspects of plant operations are acceptable. The timing of these will vary depending on the program items. The inspection manual, IMC 0305, Operating Reactor Assessment Program (IMC 0305) (NRC Staff Exhibit No. 10) provides the details of the assessment program and provides the strategic performance areas and cornerstones. The strategic performance areas are reactor safety, radiation safety, and safeguards (i.e. security). The cornerstones are initiating events, mitigating systems, barrier integrity, emergency preparedness, public radiation, occupational radiation safety, and security. Essentially, the strategic performance areas and their associated cornerstones are the key areas that the baseline inspection must cover to ensure we have confidence we can make an accurate assessment of plant safety based on the ROP. In addition to baseline inspections, other inspections could be performed if conditions exist. These are also covered in IMC 2515 and include: supplemental inspections (if issues of greater than very low safety significance are assessed); reactive inspections (if an actual event at the site has potential safety consequences); and generic safety inspections (if industry wide issues are identified, inspections at sites may be warranted to evaluate a safety issue). These items are shown in Exhibit 2 to IMC 0305. There are approximately two thousand hours of actual inspection at each licensee power reactor per year.

Q13. How are inspection results documented?

A13. All results are documented in inspection reports that are public records with the exception of security inspection reports, which are part of the Safeguards Strategic Performance Area. Because security inspections reports may contain sensitive information related to security issues, they are not made public. Baseline inspections that are performed by the resident inspector staff are reported in quarterly reports. Baseline inspections that are performed by a team with engineering specialty are reported in separate reports. Reactive, generic, and supplemental reports are issued when the inspections are complete. Reports on specific issues of safety significance are issued when their significance is assessed. There are two letters which are assessment summaries: there is one annual assessment letter and one mid-cycle assessment letter.

Q14. What does the ROP cover?

A14. The ROP covers those baseline inspections necessary to ensure the public health and safety as a result of civilian nuclear reactor power operation. The inspections are grouped in strategic performance areas and cornerstones as shown in Exhibit 1 of IMC 0305 (NRC Staff Exhibit No. 10). Other inspections are discussed in below. In addition to inspections, the NRC has an assessment process to evaluate the performance of licensees as a result of inspections or other activities. The level of oversight (and additional inspections) is based on these assessments in a graded approach.

Q15. Which inspection procedures are under the ROP baseline assessment program?

A15. IMC 2515 Appendix A, “Risk Informed baseline Inspection Program” (NRC Staff Exhibit No. 9), provides the details of the philosophy underlying the program and lists the baseline inspections that need to be performed, and the frequency they need to be performed. Some inspections are performed on an as needed basis; others are performed quarterly, annually, every outage, biennially, and triennially. There are 46 inspection procedures in the ROP.

Q16. How does the ROP baseline inspection program address the licensee’s corrective action program?

A16. There are four requirements that are part of the NRC Inspection Manual, Inspection Procedure 71152, Problem Identification and Resolution, (IP 71152) (NRC Staff Exhibit No. 22) and one other requirement in individual inspection procedures:

1. Routine review – the inspectors review all Condition Reports (CRs) and follow up on significant issues and ensure subsequent action is performed, as needed, using other inspections.

2. Semi-annual trend review – inspectors perform a semiannual review to identify trends (either NRC- or licensee identified) that might indicate the existence of a more significant safety issue.

3. The annual follow-up of selected issues – inspectors ensure that the licensee has planned and/or implemented corrective actions commensurate with the significance of identified issues. This is an in depth assessment in a focused area.

4. Biennial team assessment – inspectors assess the program in general. This inspection is the most in-depth of the inspections; and all aspects of the program are reviewed.

5. Finally each of the inspection modules has a section where the inspectors review the CRs in that specific inspection area.

Q17. How do inspection findings fit within the ROP?

A17. There are two inputs directly into the ROP assessment: one is the result of inspection findings the other is associated with the site's PIs.

Q18. What significance levels may the Staff assign to an inspection finding or PI?

A18. The levels of a Finding are very low safety significance (Green), White (moderate safety significance),Yellow (substantial safety significance) and finally Red (high safety significance) based on IMC 0609, "Significance Determination Process" (NRC Staff Exhibit No. 12).

Q19. Describe the Significance Determination Process?

A19. The significance can be based on qualitative and quantitative factors; with some being more complex than others. For example if a performance deficiency did not impact the function of a safety related piece of equipment, the issue would most likely be Green. If the Finding is related to reactor operations and caused the loss of a safety function, then detailed assessments may need to be done including the use of Probabilistic Risk Assessments (PRA). In cases where a detailed PRA is used, the color of the Findings would be based 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 probability of large early release frequency (LERF). In these cases, specific thresholds correspond to the color (e.g. for CDF greater than $1x10^{-6}$/year (yr) is White, greater than $1x10^{-5}$/yr is Yellow, greater than $1x10^{-4}$/yr is Red and for LERF greater than $1x10^{-7}$/yr is White, greater than $1x10^{-6}$/yr is Yellow, and greater than $1x10^{-5}$/yr is Red).

Q20. What criteria guided the Staff when it established the quantitative thresholds for determining a finding's significance?

A20. The NRC's policy statement on probabilistic risk assessment (PRA) ("Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," *Federal*

Register, Vol. 60, p. 42622 (60 FR 42622), August 16, 1995) encourages greater use of this analysis technique to improve safety decision-making and improve regulatory efficiency. Commission paper SECY-99-007A, dated March 22, 1999, described a method for assigning a probabilistic public health and safety risk characterization to inspection findings related to reactor safety. This risk characterization tool was the first of a set of tools that became central elements of the Significance Determination Process (SDP) to determine reactor inspection finding significance consistent with the thresholds used for the risk-informed plant PIs. The quantitative basis aligns with the Commission's Safety Goal Policy Statement ("Safety Goals for the Operations of Nuclear Power Plants; Policy Statement," *Federal Register*, Vol. 51, p. 30028 (51 FR 30028), August 4, 1986), Regulatory Guide 1.1740 and the SDP process which assign incremental changes in risk a color assessment.

IMC 0308 (NRC Staff Exhibit No. 6) further describes this process:

> In developing the new performance assessment process one of the tasks was to establish risk-informed thresholds for PIs and corresponding thresholds for inspection findings, so that indications of performance degradation obtained from inspection findings and from changes in PI values could be put on equal footing. The basis documents for establishing risk guidelines were Reg Guide 1.174, which bring in the Regulatory Analysis Guidelines, and the Safety Goal Policy Statement. The metrics that have been adopted in RG 1.174 for the characterization of risk are Core damage Frequency (CDF) and Large Early release Frequency (LERF). These are essentially surrogates for health effects, which are the principal metrics in the Safety Goal Policy Statement, and, in addition, they are consistent with the metrics used in the Regulatory Analysis Guidelines. In RG 1.174, acceptance guidelines were established for assessing changes to the licensing basis of a plant. Acceptance is predicated on increases in CDF and LERF implied by the change to the licensing basis being small.

Q21. Once findings are made, how is a plant evaluated?

A21. The NRC uses findings to place the plant in the appropriate column of the Action Matrix. The Action Matrix represents a graded approach to oversight in which the agency actions are based on the assessment inputs. As stated in IMC 0305 (NRC Staff Exhibit No. 10) paragraph 10:

> The Action Matrix (Exhibit 4) identifies the range of NRC and licensee actions and the appropriate level of communication for different levels of licensee performance. The Action Matrix describes a graded approach in addressing performance issues and was developed with the philosophy that, within a certain level of safety performance (e.g., the licensee response band), licensees would address their performance issues without additional NRC engagement beyond the baseline inspection program. Agency action beyond the baseline inspection program will normally occur only if assessment input thresholds are exceeded.

Q22. How and why do plants move from one column of the Action Matrix to another?

A22. All issues are assessed during inspections. If the NRC determines that a regulation or standard is not followed and it was reasonable that the licensee should have known or should have foreseen the issue, this is called a performance deficiency ("PD"). An issue that is more than minor in significance is called a Finding and must be documented. All PDs are evaluated in our significance determination process ("SDP") (IMC 0612, Power Reactor Inspection Reports (NRC Staff Exhibit No. 13) and IMC 0609 (NRC Staff Exhibit No. 13)) to determine the risk associated with the issue. The risk can be addressed in a qualitative or quantitative way, depending on the affected cornerstones and the tools we have to evaluate the issues. In addition to Findings, the licensee's PIs are also assessed. Each PI is linked to one of the seven cornerstones, and each has thresholds which have been pre-determined and are common among all power reactors.

The coding of these thresholds for PIs and assessment for Findings are grouped by a color scheme, but each is independently assessed. A PI that crosses a color threshold would be assigned a color to be evaluated in the Action Matrix. Separately, a Finding that crosses a color threshold would be assigned a color to be evaluated in the Action Matrix.

The PIs in themselves do not impact a Finding and vice versa. In some cases, albeit not often, a significant Finding may exist which was the reason a PI changed from Green to White. In these cases we do not "double count" and assess two White Findings. For example, say the plant had a Finding related to managing certain equipment, and the Finding caused them to shutdown several times, causing the indicator to cross the White threshold. If the Finding is assessed at a White significance based on the SDP, then only one White Finding would count.

The color of the Finding determines the licensee's column in the Action Matrix. As described in IMC 0305 (NRC Staff Exhibit No. 10) if the licensee has no greater than Green findings it would be in Column I (Licensee response – no additional action other than baseline inspection). About 80% of all plants for calendar year 2009 were in Column I. As the Findings become more significant, so does the engagement and inspections by the NRC. IMC 0305 Exhibit 4 – Action Matrix (NRC Staff Exhibit No. 11). If a licensee has one or two White Findings (not in the same cornerstone), the licensee will be in Column II (licensee response). A licensee with one Yellow or two White Findings in the same cornerstone is in Column III (degraded cornerstone). Placing a plant in Column III indicates that there is a moderate impact to safety performance. One Red or multiple degraded cornerstones puts a plant in Column IV and indicates that there is significant degradation in safety performance.

Q23. When will a Plant move to Column V?

A23. According to IMC 0305 (NRC Staff Exhibit No. 10), a plant's performance is unacceptable and the plant will be ordered to shut down when:

> 1. Licensee performance is unacceptable and continued plant operation is not permitted within this column. Unacceptable performance represents situations in which the NRC lacks reasonable assurance that the licensee can or will conduct its activities to ensure protection of public health and safety. Examples of unacceptable performance may include:
>
> (a) Multiple significant violations of the facility's license, technical specifications, regulations, or orders.
>
> (b) Loss of confidence in the licensee's ability to maintain and operate the facility in accordance with the design basis (e.g., multiple safety significant examples where the facility was determined to be outside of its design basis, either due to inappropriate modifications, the unavailability of design basis information, inadequate configuration management, or the demonstrated lack of an effective PI&R).

> (c) A pattern of failure of licensee management controls to effectively address previous significant concerns to prevent recurrence. In general, it is expected, but not required, that entry into the Multiple/Repetitive Degraded Cornerstone column of the Action Matrix and completion of supplemental IP 95003 will precede consideration of whether a plant is in the Unacceptable Performance Column."

Q24. What is the significance of a plant moving from one column to another?

A24. The significance is that the plant is showing degraded safety performance and will require more oversight and additional inspections. These additional inspections are needed to ensure the plant can continue to operate safely. The assessment process is a continuous process which requires looking at issues on a daily, quarterly, semiannual and annual basis. Even in Column IV, the plant can be safely operated because despite the safety performance degradation, additional actions (such as supplemental inspections and perhaps more frequent inspections) will be taken under the ROP to ensure safe operation. For plants in Action Matrix Columns I through IV, the NRC has the reasonable assurance the plant can be operated safely subject to additional inspections and oversight. If the agency determines, at any time, that safety performance is unacceptable, then the plant would be directed to shutdown (if it hasn't already done so) and the licensee would be in Column V. The ROP process is graded in that it requires the agency to respond and become more intrusive to ensure the plant can operate safely.

Q25. Can a plant move across more than one column at a time?

A25. Yes, for example, if a plant was in Column I or II and a Finding resulted in a Red Finding, the Finding would most likely result in moving the plant to Column IV.

Q26. How and why does a licensee move from one Column to another?

A26. This is discussed in detail above, but in short the shift in Column is based on the significance of Findings discovered during inspections or as a result in PI that have crossed a color boundary. IMC 0305 provides a detailed accounting of how long each Finding is "counted" to determine how many “greater than green” Findings exist at any one time.

Q27. How long will the NRC rely on an inspection Finding or PI to determine the licensee’s column in the Action Matrix?

A27. The inspection guidance provides a detailed accounting of how long each Finding is “counted” to determine how many “greater than green” Findings exist at any one time. But generally the Findings are assessed on a quarterly basis. But if a Finding has been finalized greater than Green in the middle of the quarter, then the Column shift occurs when the Finding was first introduced without waiting for the quarter to end. In general a Finding stays “on the books” (is being counted for in the Action Matrix) for one year or until the NRC supplemental inspection has cleared the Finding – whichever is longer. The Finding is cleared when the supplemental inspection team has verified the licensee has properly evaluated the cause, taken appropriate corrective actions to prevent recurrence, and evaluated the extent to which the problem could exist elsewhere at the site.

Q28. What is the significance of a white finding on a substantive-cross cutting human performance issue?

A28. After a Finding has been established, the inspectors will evaluate the likely cause for the issue. A cross-cutting aspect is a performance characteristic that is the most significant contributor to a performance deficiency that resulted in a finding. Cross-cutting aspects are so-called because they impact all the cornerstones. Not all Findings have a cross-cutting aspect. For example, the issue may not be indicative of current performance. The aspects that are assigned will be in one of the three cross-cutting areas. Exhibit 1 of IMC 0305 (NRC Staff Exhibit No. 10) identifies the three cross-cutting areas: human performance, problem identification and resolution, and safety conscious work environment.

Cross-cutting aspects are not Findings themselves, but rather are the most significant contributors to an issue. As such they do not change a White Finding in significance. A White

Finding is still a White Finding regardless if there is a cross-cutting aspect.

Q29. How does the NRC determine whether to assign a crosscutting aspect?

A29. After a Finding has been established, the inspectors will evaluate the likely cause for the issue. This assessment is discussed in IMC 0612 (NRC Staff Exhibit No. 13). A cross-cutting aspect is a performance characteristic that is the most significant contributor to a performance deficiency that resulted in a finding. A cross-cutting aspect is a characteristic of a Finding, it is not a Finding itself. If the cause of finding is reflective of current performance (the inspectors ask the question: did the performance characteristic described by this potential cross-cutting aspect occur within the last three years) and aligns with one of the aspects listed in IMC 0310, "Components Within the Cross-cutting Areas" (NRC Staff Exhibit No. 19), then the Finding is assigned a cross-cutting aspect.

Q30. How does the NRC determine whether a Substantive Cross-Cutting Issue ("SCCI") exists?

A30. The agency recognized that the cross-cutting areas of human performance, problem identification and resolution, and safety conscious work environment manifest themselves in the causes of issues. The components of these three areas are attributes of the safety culture. In general, the process works as follows: after a Finding is determined, the inspectors performing the inspection assess whether a cross-cutting aspect should be assigned. During an assessment period if there four or more Findings related to the same cross-cutting aspect, a theme is developed. If the licensee's actions to date have not been effective in addressing the NRC concerns, the licensee is then determined to have a SCCI. The SCCI does not change the Action Matrix Column, nor does it change a Finding's risk determination. As stated in IMC 0308, (NRC Staff Exhibit No. 6) section 05.05, a SCCI means the NRC "has a significant level of concern with the licensee's performance in the cross-cutting area." A licensee may be in Column I and have a SCCI.

A SCCI cannot move a plant in columns. The ROP is built on the philosophy that inspection findings move a plant through the columns of the Action Matrix, as discussed above, because they are indicative of degraded performance. In contrast, SCCIs are potential leading indicators of degraded performance but they do not actually indicate degraded 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.

Q31. Does NRC identification of a SCCI-Human performance indicate an inadequate safety culture?

A31. No, it indicates we have significant concerns regarding some of the aspects of the safety culture, but it does not indicate that the NRC believes the safety culture as a whole is inadequate. It should be noted if a plant is in an SCCI for three consecutive six month periods, IMC 0305 then directs the Staff to request that the licensee perform a safety culture assessment. After three periods in an SCCI, the region is directed to work with the Executive Director's office on what actions to take. The basis is clear: if actions to improve the site are not being effective, other actions (including deviations from the Action matrix) can be considered to ensure the aspects of safety culture are addressed.

Q32. Please explain the basis for the staff's identification of a SCCI-Human performance at PINGP?

A32. During the assessment period there were 4 aspects where there were more than four Findings in the same cross-cutting aspect. A theme was developed. The licensee's actions to date have not been effective in addressing the NRC concerns. The Agency determined the licensee has a Substantive Cross-Cutting Issue (SCCI).

Q33. In the specific case of PINGP, does Staff identification of a SCCI-Human Performance indicate inadequate safety culture?

A33. No, the Agency has significant concerns on certain aspects of the safety culture, but I would not conclude the safety culture is inadequate based on the current information available.

Q34. In your opinion, do findings that resulting in PINGP Units 1 and 2 being placed in Column II indicate poor safety culture at PINGP?

A34. No. IMC 0305, (NRC Staff Exhibit No. 10) defines safety culture as “That assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance.” Being in Column II does not imply that a plant’s safety culture is poor. In addition, although the site is in a SCCI, which implies we have concerns with attributes of their safety culture, I would not say the assembly of attributes, attitudes, and characteristics of PINGP indicate that safety culture is poor. The NRC has concerns that the site needs to address. Specifically the site needs to implement a strategic plan that addresses the aspects in human performance in the area causing the Findings. The site’s plan needs to reduce the number of Findings with those aspects and the site needs to create a program to ensure there is sustainability for future operations. The NRC will conduct follow-on inspections to validate whether this does or does not occur.

Q35. Describe the Staff’s concerns with the Corrective Action Program (“CAP”) at PINGP, Units 1 and 2.

A35. The NRC has had concerns with some aspects of the CAP process over the last few years. These items were assessed in the last two Problem Identification and Resolution (PIR) Inspections, with the most recent report being documented September 25, 2009 (NRC Staff Exhibit No. 50) and the previous being documented December 21, 2007 (NRC Staff Exhibit No. 59). In all cases we noted problems in the CAP process, but concluded it was functioning, and found actions were needed to improve the process. The September 25, 2009 report states in summary: "On the basis of the information reviewed, the team concluded that the corrective action (CA) program at Prairie Island was functional, but implementation was lacking in rigor resulting in inconsistent and undesirable results. In general, the licensee had a low threshold [that is, the licensee generally tended to be conservative and put items in the process] for identifying problems (issue reports called CAPs) and entering them in the CA program; however, some significant issues went unrecognized and therefore CAPs were not issued for these."

Q36. When and how did these concerns originate?

A36. Some items were documented in the PIR inspection report dated December 21, 2007 (NRC Staff Exhibit No. 59), and others in the recent inspection report dated September 25, 2009 (NRC Staff Exhibit No. 50).

Q37. What regulatory provisions govern the CAP?

A37. The corrective action process is required, in part by 10 CFR 50 Appendix B, notably Criterion XVI:

> Corrective Action Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

Q38. How does the CAP relate to the ROP?

A38. At its most fundamental level the CAP is the site's process to implement the cross-cutting area called problem identification and resolution. So while the CAP may be in a cross-cutting area, problems in these areas do not cause the licensee to change position on the Action Matrix. A licensee's position in the Action Matrix is based on a Finding's safety significance as determined by the significance determination process (IMC 0609 (NRC Staff

Exhibit No. 12)). Problems in the CAP process may be the cause of some performance deficiencies that result in Findings.

Q39. Did the Staff's concerns with the CAP at PINGP lead to any inspection Finding, and, if so, what level of Findings?

A39. The last PIR report (September 25, 2009) (NRC Staff Exhibit No. 50) had three Findings, all of which screened as Green. One was a direct violation of 10 CFR 50 Appendix B, criterion XVI, Corrective Action.

Q40. Why did the Staff classify the findings regarding the PINGP CAP as Green Findings?

A40. Because when it was reviewed using our process (IMC 0609 (NRC Staff Exhibit No. 12)), it was determined to be of very low safety significance. The risk assessment determined that because there was no loss of safety function, the issue had very low safety significance.

Q41. PIIC asserts that PINGP has failed to promptly and effectively correct deficient conditions, using the licensing response to leakage from the reactor refueling cavity as an illustration. Do you agree?

A41. No. I would say we have concerns and there have been concerns in the past of failing to identify and correct certain issues. But I would not characterize the failure as systemic and thus indicative of a CAP process that is not functioning. Typically the CAP process functions and ensures the issues that are important to safety are identified promptly and correctly commensurate with their safety significance.

With respect to the example of leakage from the refueling cavity, I am not a structural expert, but when I found out about the issue we had our structural personnel review the issue to determine if there was an immediate safety issue. That is, our expert looked for an impact to the structure of the containment, liner, or other required supports. The assessment concluded there was not. The leakage needs to be corrected, but it is not a safety significant item at this time. If the licensee does not take action promptly, it could become a more significant issue. The NRC's most recent inspection findings on this issue are discussed in NRC Inspection Report No. 0500282/2010003; 05000306/2010003 (July 26, 2010) (Staff Exhibit No. 51). This report states that the NRC did not identify any findings of significance, and notes that previous evaluations have not revealed any degradation of the containment pressure vessel, concrete, or rebar due to the refueling cavity leakage. *Id*. at 18.

So the leakage needs to be addressed, but this item does not show me there is an inadequate corrective action process. The licensee has taken action, albeit not totally effective. The NRC has reviewed the issue and has determined there is no Finding at this time. Follow-up action is required by the licensee for license renewal, and the NRC resident is following these issues during outages as well.

Q42. Describe the condition that led to the Staff's inspection finding in the fourth quarter of 2008 regarding the 11 Turbine Driven Auxiliary Feedwater Pump (TDAFWP).

A42. A valve that was required to be open for the pump to operate to perform its safety function was found out of position. This resulted in the safety related component not being able to perform its safety function to mitigate events.

Q43. How did the Staff first become aware of this condition?

A43. The condition self-revealed. In other words, when the TDAFWP was running, it shutdown due to this PD (failure to have the valve in its required position).

Q44. What regulatory provisions did this condition violate?

A44. The site's Technical Specification 3.7.5.B requires, in part, that if one Auxiliary Feedwater train is inoperable in Modes 1, 2, and 3, the affected train shall be restored to operable status within 72 hours or the plant placed in Mode 3 within 6 hours and Mode 4 within 12 hours. Specifically, the pump was inoperable for greater than 12 hrs (approximately 138 days) due to the discharge low pressure switch being isolated and no actions were taken to restore the pump to operable status or to place the plant in Mode 3 or 4.

Q45. What significance level did the Staff ultimately assign this finding?

A45. White – low to moderate safety significance.

Q46. Why did the Staff classify the failure to adequately control the position of a normally open valve used to isolate the 11 TDAFWP as a White Finding?

A46. The NRC performed a detailed quantitative PRA. The licensee provided information to us which we considered and agreed with, in part. In summary, the NRC considered the licensee's information in the final significance determination with some exceptions. The NRC analysis using the licensee's information, with the modifications, resulted in a change in core damage frequency of approximately $2x10^{-6}$/yr. The dominant core damage sequence was a control room fire which results in abandonment of the control room, followed by the failure of the 11 TDAFWP, and a failure of the operator to recover the pump. With a change in CDF of $2x10^{-6}$/yr, this is a White Finding. The NRC's analysis supporting this conclusion is documented in NRC Inspection Report No. 05000282/2008008, (January 27, 2009) (NRC Staff Exhibit No. 52) and NRC Special Inspection Report 05000282/2008008; 05000306/2008008 (November 7, 2008) (NRC Staff Exhibit No. 53).

Q47. In your opinion, does Finding indicate poor safety culture at PINGP?

A47. No. There are aspects of the safety culture that concern me and that they need to address promptly (which they have), but I do not see this Finding as indicative of a weak safety culture.

Q48. Describe the condition that led to the Staff's inspection finding regarding the radioactive material shipment sent on October 29, 2009.

A48. A transportation shipment of low level waste had a radiation reading on the outside of the packaging that was above the Department of Transportation (DOT) limits. When the package arrived at its destination, the detected radiation levels exceeded Nuclear Regulatory Commission (NRC) regulations, which invoke the Department of Transportation requirements limiting the radiation level on the surface of a package shipped in an open transport vehicle to 200 millirem per hour.

Q49. How did the Staff first become aware of this condition?

A49. The NRC was informed by the site who received a call from the recipient of the container informing them of such a condition.

Q50. What regulatory provisions did this condition violate?

A50. This violated DOT rules for shipping waste. Title 10 CFR 71.5, "Transportation of Licensed Material," requires licensees to comply with the Department of Transportation (DOT) regulations in Title 49 CFR parts 170 through 189 relative to the transportation of licensed material. Specifically,

1) Title 49 CFR 173.441(a) requires that each package of radioactive material offered for transportation must be designed and prepared for shipment, so that under conditions normally incident to transportation, the radiation level does not exceed 2 millisievert per hour (200 millirem per hour) at any point on the external surface of the package.

Contrary to the above, on October 29, 2008, the licensee shipped a package containing radioactive material that was not designed or prepared to assure that, under conditions normally incident to transportation, the radiation level on the external surface of the package would not exceed 200 millirem per hour.

2) Title 49 CFR 172.704, "Training Requirements," requires that individuals involved in the transport of hazardous materials receive function specific training relative to their specific tasks, and that these individuals receive recurrent training at least once every three years.

Contrary to the above, as of October 29, 2008, five people involved in preparing a package for radioactive shipment and transport had not received the required function-specific training.

Q51. What significance level did the Staff ultimately assign this finding?

A51. White.

Q52. Why did the Staff classify PINGP's failure to comply with applicable Department of Transportation regulations when shipping the radioactive material on October 29, 2009, as a White Finding?

A52. The NRC did a qualitative risk assessment using technical assessments. The NRC used the results of the measurements obtained at the receipt of the package and the relative risk from the point radiation source to develop the significance of the finding. Both radiation detection instruments measured radiation levels that exceeded the regulatory limit, which provides a level of protection to a member of the public that may come into contact with the shipment. Although no exposures to the public resulted from the shipment, the potential consequences could have been greater under less favorable circumstances. Any shipment with radiation levels that exceed regulatory limits can be potentially significant, and in this case the risk was more than minimal. Based on this assessment and after considering the information developed during the inspection, the information provided at the regulatory conference by the site, and supplemental information, the NRC has concluded that the finding is appropriately characterized as White, a finding with low to moderate increased importance to safety that may require additional NRC inspections. The NRC's analysis supporting this conclusion is documented in NRC Inspection Report No. 05000282/2009008; 05000306/2009008, (May 6, 2009) (NRC Staff Exhibit No. 54) and NRC Inspection Report 05000282/2008009; 05000306/2008009 (February 10, 2009) (NRC Staff Exhibit No. 55).

Q53. In your opinion, does this finding indicate a weak safety culture at PINGP?

A53. No. There are aspects of the safety culture that concerned me and they need to address promptly (which they have), but I do not see this as indicative of a weak safety culture.

Q54. Describe the condition that led to the Staff's inspection finding in July of 2009 regarding the design of the PINGP Unit 2 component cooling water (CCW) system.

A54. This White Finding is associated with the licensee's failure to design the component cooling water system such that it would be protected from the impact of a high-energy line break, seismic, or tornado events.

Q55. How did the Staff first become aware of this condition?

A55. The NRC found out when the site wrote a CAP document indicating that while they were performing a walkdown of CCW piping in response to a previous CAP, they discovered this vulnerability.

Q56. What regulatory provisions did this condition violate?

A56. This violated 10 CFR 50, Appendix B, Criteria III which requires that the design basis of safety components be adequately translated into configuration represented in the plant. In this case, the piping was vulnerable to design basis events that it should have been protected from.

Specifically, Title 10 of the Code of Federal Regulations, Part 50, Appendix B, criterion III, "Design Control," requires, in part, that measures be established to assure that the design basis for safety-related functions of structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. Further, Criterion III requires that the design control measures provide for verifying or checking the adequacy of designs.

Contrary to the above, as of July 29, 2008, the licensee failed to implement design control measures to ensure that the design basis for the component cooling water system was correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to ensure that the safety-related function of the component cooling water system was maintained following a high-energy line break, seismic, or tornado events in the turbine building.

Q57. What significance level did the Staff ultimately assign this finding?

A57. White.

Q58. Why did the Staff classify the inadequate design of the component cooling water system for PINGP Unit 2 as a White Finding?

A58. The NRC performed a detailed quantitative PRA. The NRC analysis bounded the resulting change in CDF in being less than $1x10^{-5}$/yr and greater than $1x10^{-6}$/yr. This corresponds to a White Finding. The NRC's analysis supporting this conclusion is documented in NRC Inspection Report No. 05000306/2009013 (September 3, 2009) (NRC Staff Exhibit No. 56) and NRC Inspection Report 05000282/2008010; 05000306/2008010 (August 5, 2009) (NRC Staff Exhibit No. 57).

Q59. Does the Finding indicate weak safety culture at PINGP?

A59. No. There are aspects of the safety culture that concerned me and they need to address promptly, but I do not see this as indicative of a weak safety culture.

Q60. As a result of the findings at PINGP, discussed above, have PINGP Units 1 or 2 moved columns in the Action Matrix?

A60. Both units were in Column II for a period of time. The current assessment is Unit 2 is in Column II and Unit 1 has returned to Column I. The transportation and TDAFWP Findings are no longer considered in the Action Matrix. The Findings have been inspected, were assigned in the matrix for a year, and are now removed.

Q61. What Findings are currently open for PINGP Units 1 and 2?

A61. The only Greater-than-Green Finding, which is still open, is the high-energy line break/CCW issue. One other Finding, that is preliminary Greater- than- Green, is failure to protect safety related piping from effects of flooding. No final decision by the agency has been made. A Regulatory Conference was held on July 13, 2010.

Q62. Based on PINGP's position in the Action Matrix, does the Staff currently have reasonable assurance that NSP will operate PINGP in accordance with its licensing basis.

A62. Yes.

Q63. What are the grounds for this conclusion?

A63. The Staff conducts continuous, quarterly, mid cycle and annual assessments. Although the agency has some concerns with site performance; overall we have reasonable assurance the site will continue to operate in accordance with their licensing and design basis. If additional issues occur, the NRC will take action in accordance with the oversight process. The NRC has, and will continue to have, increased oversight until performance is shown to improve and the licensee returns to Column I.

John (Jack) B. Giessner

Statement of Professional Qualifications

CURRENT POSITION

Branch Chief — Division of Reactor Projects, Region III,
United States Nuclear Regulatory Commission

EDUCATION

B.S., Naval Academy, 1983, Physics (Top 1% of class)

M.A., Naval War College,1993, National Security Affairs

SUMMARY

Five years of experience as an NRC inspector, preceded by 9 years of experience in the nuclear power industry and 4 years in the nuclear navy.

Employment

NRC CAREER HISTORY (2004 – current)

- As Branch Chief for Prairie Island Nuclear Generating Plant, Palisades Nuclear Plant and Fermi since October 2008, I provide assessment and leadership in implementing the inspection program at these sites.

- As a qualified reactor engineer for Point Beach Nuclear Plant participated in the Problem Identification and Resolution inspection, Confirmatory Action Letter follow-up and closeout inspection; supported the final Inspection (95003) assessment of the site.

- Was a team member for Kewaunee Power Station inspection related to auxiliary feedwater and flooding issues in 2005. Assisted in providing inspection activities for the modifications being put in place; implemented the Significance Determination Process (SDP) for preliminary greater than green issues. In addition, drafted the Significance and Enforcement review Panel (SERP) package, assisted the Senior Reactor Analyst (SRA), and drafted the report for the branch. Assisted in the technical support for the Yellow Finding during the Regulatory Conference and final SERP package.

- While inspecting at Kewaunee Power Station, identified a potential issue of flooding that could impact both trains of heat removal. Although the licensee did not agree with the position, a Technical Interface Agreement (TIA) was written (TIA 2005-11). Initial response by NRR to the TIA validated there was potential concern and the licensee took prompt action to correct the vulnerability.

- While a RI at Palisades Nuclear Plant, identified several findings including one selected as a Value Added Finding (VAF) (2006-05) for 2006 with licensee's failure to declare piping inoperable for a through wall leak, which provided a containment barrier.

- While RI at Palisades, identified two preconditioning items related to high risk components on the emergency Diesel Generators and the Turbine Driven Auxiliary feedwater pumps.

- While RI at Palisades in November 2006, identified all three Auxiliary Feedwater (AFW) pumps were inoperable during start-up. A Special Inspection Team was needed to address operations' performance issues. A VAF was also written.

- Supported security inspections and outage activities.

- Acted as the Senior Resident Inspector (SRI) at Duane Arnold Energy Center (2006); included the normal duties as SRI

    - While SRI at Duane Arnold, discovered and investigated the improper assessment for equipment availability during a surveillance test for a risk important component. This was selected as a VAF (2006-24).

- Acted as SRI at Calvert Cliffs Nuclear Power Plant (2006); included the normal duties as SRI

    - Processed and developed the SERP package for a Preliminary White issue for 1A Emergency Diesel Generator (EDG) unavailability at Calvert Cliffs Unit 1; assisted in the presentation for the SERP; presented the exit meeting; supervised the submission of the report and choice letter; completed the final White finding
    - Oversaw the visit by the new NRC Chairman at the site
    - Received cash award and VAF for issue concerning AFW testing without high energy line break protection

- Acted as SRI Indian Point Nuclear Generating (2007); included the normal duties as SRI

    - reviewed issues of siren performance; tritium leakage and participated in key briefings to local, state, and federal officials
    - Received cash award for backshift tour and discovery of inattentive security officer in August of 2007

- Team member for Special Investigation Team (SIT) team at Point Beach Nuclear Plant (2007) for failure of AFW pumps; resulted in several findings

- Team member for 95002 assessment for Kewaunee Power Station's Yellow and White finding; resulted in several findings

- Received VAF for deficiencies found during a containment closeout inspection at Palisades Nuclear Plant in 2007 (VAF 2008-006)

- Participated in the receipt, assessment, investigation and closeout for allegations at Point Beach Nuclear Plant and Palisades Nuclear Plant

COMMERCIAL CAREER HISTORY

AMERICAN ELECTRIC POWER (Donald C Cook Nuclear Plant), Bridgman, MI, 2000-2004

Team leader and director responsible for strategic vision, equipment performance, human performance and safety of DC Cook; reporting to the Chief Nuclear Officer.

-Plant/Site Engineering Director responsible for all site engineering functions

-Senior leader (Emergency Director) on the Emergency Response Organization (ERO) at DC Cook

-Director of information technology upgrade at DC Cook responsible for upgrade of work management, document management and site data migration

-Assistant Operations Manager for DC Cook responsible for day-to-day operations of the plant, technical issue resolution and operating procedure upgrades during the restart of DC Cook in 2000 and post-restart; managed to successful conclusion many operational and technical issues during the restart effort of Cook; ensured the operating license personnel and plant systems could meet the design and licensed basis of DC Cook.

- As manager of the procedure upgrade project ensured adequate technical talent and processes were in place to upgrade hundreds of Emergency, Abnormal and Normal Operating procedures
- Addressed Operability Determinations (GL 91-18) and Operator Workarounds to ensure required items were addressed prior to plant restart
- Primary liaison on all engineering and technical issues
- Evaluated Licensed and Non-licensed Operator Performance on shift and in training; and ensured that the right standards were being promulgated and enforced

PUBLIC SERVICE ELECTRIC AND GAS (Salem Nuclear Generating Station), Salem NJ, 1995-2000

Team member and leader who provided operational and technical guidance to Operations and Plant Senior management during restart of both Salem Units; reporting to Assistant Operations Manager and Operations Manger.

-Supervisor for Operations during extended shutdown at Salem in 1996-1997; responsible for technical issue resolution in Operations; responsible for the procedure upgrade at the site; primary link between Operations, Design and Plant Engineering for issues.

- Provided resolution for significant Operability Determinations (GL 91-18) issues
- Provided resolution for long-standing operator workarounds
- Team leader in the effort to upgrade Normal Operation, Abnormal Operation

and Emergency Operations procedures; over 1000 revisions were completed; the plan included validation, verification and training

- Established teams to address significant technical issues including the Appendix R program, switchgear, control room and auxiliary ventilation issues.
- Managed the 50.59 process for procedures ensuring proper reviews were complete
- Provided review of root causes and performed effectiveness reviews to ensure issues were resolved to prevent recurrence

UNITED STATES NAVAL OFFICER

Officer USN (1983-1995), Honorably Discharged 1995
Qualified for Command at Sea in Navy (1992)
Qualified as Engineer Officer by Naval Reactors (1987)
Qualified as Joint Staff Officer (1993)
US Navy Nuclear Submarine Service and Joint Staff Officer
Navy Nuclear training pipeline (1984)
Submarine Officer Basic (1985) and Advanced Training (1988)
Midshipman US Naval Academy (1979-1983)
Held the highest security clearances in the Department of Defense (including Top Secret)

October 1993 - November 1995, Commissioned Officer US Navy (O-4 Lieutenant Commander)
Staff Officer at Unified Military Command: US Strategic Command (Omaha, Nebraska)

- Certified Joint Staff Officer in Armed Forces
- Managed day-to-day operations of US intercontinental and submarine ballistic missiles; and developed contingency planning for the forces
- Provided policy guidelines for future strategic forces and analyzed impact on the forces
- Provided detailed briefs to senior military and civilian leadership in Washington on several key nuclear initiatives

May 1989 - July 1992, Engineer Officer on US Navy Nuclear-powered Submarine responsible for the operation, maintenance and supervision of all reactor plant, auxiliary and propulsion systems

- Supervised 10 mid level supervisors and 50 operators on the submarine
- Primary liaison with the Puget Sound Shipyard during complex refueling overhaul. The $120M Overhaul was completed ahead of schedule.
- Supervised reactor plant, ship testing and operations without incident
- Developed the qualification standard and administered the ship's Quality Assurance Program
- Primary member of the Joint Test group responsible for approval and verification off all nuclear procedures and testing
- Certifying Officer for over 100 qualifications. Devised and implemented a comprehensive training program to prepare the crew transition from maintenance to an operational environment
- Qualified for Command at Sea

October 1987 - October 1988, Weapons Officer and Engineer Officer on US Nuclear-powered Submarine responsible for operation and maintenance of all weapons and sensor systems onboard

- Qualified as Engineer Officer and performed duties as Engineer Officer as a Junior Officer. This is rarely approved by the cognizant reactor authority (Naval Reactors).
- Qualified as Weapons Officer in charge of all nuclear and non-nuclear weapons.
- Supervised 4 midlevel and 25 technicians in day-to-day inspections and operations.

May 1985 - October 1987, Division Officer onboard US Nuclear-powered Submarine performing duties as a front-line supervisor in several reactor plant and ship system departments

- Supervised Chemistry and Radiological Controls department, and Main Propulsion Assistant to the Engineer Officer
- Supervised sensor, weapons and communication equipment during major deployments important to the defense of this country

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
NORTHERN STATES POWER COMPANY	)	Docket Nos. 50-282-LR/ 50-306-LR
	)	
(Prairie Island Nuclear Generating Plant,	)	
Units 1 and 2)	)	

AFFIDAVIT OF JOHN (JACK) B. GIESSNER

I, John B. Giessner, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Executed in Accordance with 10 CFR § 2.304(d)

John B. Giessner
Branch Chief, Division of Reactor Projects
U.S. Nuclear Regulatory Commission, Region III
2443 Warrenville Road, Suite 210
Lisle, IL 65032-4352
(630) 829-9619
john.geissner@nrc.gov

Dated at Rockville, Maryland
this 30th day of July, 2010

July 30, 2010

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
NORTHERN STATES POWER COMPANY	)	Docket Nos. 50-282-LR/ 50-306-LR
	)	
(Prairie Island Nuclear Generating Plant,	)	
Units 1 and 2)	)	

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

Q1. Please state your name, occupation, and by whom you are employed.

A1(a). My name is Dr. Valerie Barnes ("Barnes"). [1] I am employed by the U.S. Nuclear Regulatory Commission ("NRC") as Senior Technical Advisor in Human Factors, Office of Nuclear Regulatory Research. A statement of my professional qualifications is attached.

A1(b). My name is June Cai ("Cai"). I am employed by the NRC as the Senior Safety Culture Program Manager in the Office of Enforcement. A statement of my professional qualifications is attached.

A1(c). My name is Molly Jean Keefe ("Keefe"). I am employed by the NRC as a Human Factors Specialist in the Health Physics and Human Performance Branch of the Division of Inspection and Regional Support in the Office of Nuclear Reactor Regulation ("NRR"). A statement of my professional qualifications is attached.

A1(d). My name is Audrey L. Klett ("Klett"). I am employed by the NRC as a Reactor Operations Engineer in the Performance Assessment Branch of the Division of Inspection and

[1] In this testimony, the sponsors of each numbered response are identified by their last name; no such designation is provided for paragraphs which are sponsored by all witnesses.

Regional Support in NRR. A statement of my professional qualifications is attached.

Q2. Please describe your current responsibilities as relevant to safety culture.

A2(a). (Barnes) I am currently involved in the following activities related to the agency's safety culture initiatives:

- 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.

A2(b). (Cai) Currently I have the staff lead in advising, developing, and implementing activities related to supporting and improving the NRC's internal safety culture. In addition, I advise and participate in the NRC's external safety culture activities in the oversight of licensees. I am also leading a variety of continuous learning and improvement efforts on safety culture, including benchmarking other agencies and organizations, learning from operating experience from other industries, and training and development of new staff.

A2(c). (Keefe) I am the contact for safety culture in NRR. My current responsibilities involve developing a new definition and traits of safety culture for the NRC's draft safety culture policy statement and working through issues associated with implementation of the policy statement that is under development. I am also involved in stakeholder outreach and present on the policy statement at various conferences and workshops. Prior to my NRR work, I worked in the NRC's Office of Nuclear Reactor Research and was involved in the Reactor Oversight Process ("ROP") Safety Culture Working Group and the development of the safety culture enhancements to the ROP and the inspector training program in 2006. Additionally, I have participated in safety culture and safety conscious work environment assessments at plants throughout the country including Davis-Besse, Salem and Hope Creek, Duane Arnold, and Palo Verde.

A2(d). (Klett) I have the staff lead for the ROP's operating reactor assessment program and am responsible for the oversight of the program's implementation. The ROP operating reactor assessment program describes the NRC's oversight of an operating reactor licensee's safety culture and the process for identifying substantive cross-cutting issues.

Q3. What is the purpose of your testimony?

A3(a). (Barnes) The purpose of my testimony is to describe theory, research and practice related to safety culture analysis.

A3(b). (Cai) The purpose of my testimony is to describe safety culture and the Commission's safety culture policy, including the development of that policy.

A3(c). (Keefe) The purpose of my testimony is to describe the Commission's implementation of its safety culture policy through the ROP, and opine on the current status of safety culture at Prairie Island and the ability of the ROP to verify safety culture at Prairie Island.

A3(d). (Klett) The purpose of my testimony is to describe the oversight of safety culture in the Reactor Oversight Process and the ability of the ROP to verify adequate safety culture at Prairie Island.

Safety Culture Generally

Q4. What is safety culture?

A4. (Barnes) The term, "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. There have been many definitions of safety culture published in the research literature (e.g., E.H. SHEIN, ORGANIZATIONAL CULTURE AND LEADERSHIP (2d ed. 1992); Guldenmund, 2000 (NRC Staff Exhibit 26), cites 16 definitions; Mearns, et al, 2003 (NRC Staff Exhibit 27); Von Thaden and Gibbons, 2008 (NRC Staff Exhibit 31)), but no consensus exists on a "best" definition. Within the nuclear domain, numerous regulatory bodies, including the U.S. Nuclear Regulatory Commission, and other entities have developed working definitions of safety culture to communicate the necessity of maintaining an over-arching commitment to the protection of people and the environment in nuclear operations ("Policy Statement on the Conduct of Nuclear Power Plant Operations," 54 FR 3424, January 24, 1989 (NRC Staff Exhibit 35); "Freedom of Employees in the Nuclear Industry to Raise Safety Concerns Without Fear of Retaliation," 61 FR 24336, May 14, 1996 (NRC Staff Exhibit 36); "Draft Safety Culture Policy Statement: Request for Public Comments" 74 Fed. Reg. 57525, November 6, 2009 (NRC Staff Exhibit 40); INSAG, 1991 (Staff Exhibit 32); INPO, 2004 (NRC Staff Exhibit 29). For the purposes of the ROP, the Office of Nuclear Reactor Regulation has adopted the INSAG (1996) definition of safety culture, which is "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." NRC Staff Exhibit 33. As Ms. Cai will discuss in greater detail, the NRC staff is working with stakeholders to develop a revised definition that will apply across the range of activities and organizations that the NRC regulates.

"Safety climate" is a related concept that has been used interchangeably with "safety culture." As currently defined by most researchers and theoreticians, however, safety climate is considered to be an organization's members' shared perceptions of and attitudes towards the state of safety within the organization at a given time. Climate is more easily affected by recent events and conditions both within and external to the organization than culture, and is therefore considered to be more transitory than culture. In his book, The Corporate Culture Survival Guide (1999), Dr. Edgar Schein suggests that climate is a "surface feature" or "artifact" of the underlying organizational safety culture. Some authors have distinguished culture from climate by characterizing culture as analogous to the "personality traits" of an organization whereas climate is analogous to the current "mood or state" of the organization (*cf*. Weigmann, et al, 2002 (NRC Staff Exhibit 30)).

For simplicity of communication, the NRC uses the term, "safety culture," to refer to the concepts included within the definitions of both safety culture and safety climate.

Q5. How can safety culture be assessed or evaluated?

A5. (Barnes) There is general agreement that safety culture can be assessed and evaluated, but some disagreement as to the most appropriate methods. Experts from the disciplines of anthropology and sociology suggest that ethnographic methods, particularly extended participant observation and interviews, are most useful for understanding an organization's safety culture. Organizational psychologists see additional value in workforce surveys, although it is recognized that survey results are more likely to provide information about the organization's current safety climate than its culture.

When evaluating a licensee's self- or third-party safety culture assessment (or when performing an independent safety culture assessment), the NRC staff prefers that a combination of methods be used (*cf.* NRC Inspection Procedure ("IP") 95003 (NRC Staff Exhibit 24)). For example, a survey may be useful for identifying shared perceptions, beliefs and attitudes among members of the organization with respect to the importance the organization places on nuclear safety, as well as differences in perceptions, beliefs and attitudes between departments or work groups. However, survey results do not provide sufficient information to understand the sources or causes of the perceptions, beliefs and attitudes, or whether those opinions are transitory or represent deeply held and enduring beliefs. Therefore, the staff prefers that an assessment also include historical investigation, typically through document reviews and interviews; case studies, for example, of any work groups that may have expressed more negative views than others in survey results or of events that were meaningful to the workforce; observations of meetings and the performance of work at the site; and interviews with site personnel from all levels of the organization. Use of multiple methods in a safety culture assessment strengthens the likelihood that the conclusions drawn from the assessment are valid and that they will be useful to inform the regulatory review of any licensee corrective action plans to address identified safety culture weaknesses.

Q6. How can safety culture be improved?

A6. (Barnes) There are many ways to improve an organization's safety culture, which may range in scope from an organization-wide change effort to very small changes, such as improving the usability and accuracy of a procedure to ensure it can be implemented in the circumstances for which it was written. There is an extensive research and practice literature available to aid in the design and conduct of organization-wide change efforts, which also applies to safety culture improvement efforts (*cf.* W.W. BURKE, ORGANIZATION CHANGE: THEORY AND PRACTICE (FOUNDATIONS FOR ORGANIZATIONAL SCIENCE) (2002)).

In response to the results of a safety culture assessment, interventions to improve safety culture should be tailored to the specific nature and scope of the weaknesses identified. For example, assessment results might show that the majority of the members of a particular work group are reluctant to raise safety concerns to their supervisor. Interventions could include evaluation of the supervisor's leadership style and behaviors and then coaching, additional training, or possible replacement of the supervisor; facilitated problem-solving meetings involving the supervisor and group members; management meetings with the group members to reassure them that raising concerns is valued behavior; prompt action to resolve safety concerns; and on-going monitoring and follow-up by management or their representatives to verify that the interventions have been effective. As another example, safety culture assessment results might indicate that the organization's work practices have fallen below industry standards. In this case, interventions could include initiating benchmarking trips to other sites and implementing the lessons learned from those trips; funding staff attendance at conferences and participation in industry working groups; creating incentives for personnel to seek continuing education; and bringing in representatives of industry groups or from other sites for training sessions or workshops.

Q7. How can a positive safety culture be maintained?

A7. (Barnes) There is a consensus among experts as well as empirical support for the key role that an organization's leaders play in sustaining a positive safety culture (*cf.* Yule and Flin; 2007 (NRC Staff Exhibit 58)). Research and practical experience indicate that management commitment to safety is fundamental to maintaining a positive safety culture. A commitment to safety is demonstrated both directly in the oral and written messages that managers communicate to the organization's members, as well as indirectly through the decisions they make and the behavior they model. As an example, a senior management decision to lengthen a refueling outage to make discretionary repairs (i.e., those not required by regulation) to safety-related equipment would demonstrate management commitment to safety over the competing goal of maximizing production. Of course, a single management communication or action is insufficient to ensure that a positive safety culture is maintained. A pattern of consistent management emphasis on safety over time is necessary to shape the culture.

Like managers, supervisors also have a central role in maintaining a positive safety culture through their patterns of communication and behavior. Supervisors have the additional responsibility of translating management's commitment to safety into specific expectations for how work is to be performed in their work groups. Supervisors are also responsible for reinforcing those expectations daily.

Informal leaders and individual contributors in an organization similarly contribute to maintaining a positive safety culture. These individuals demonstrate a commitment to safety by, for example, ensuring that their day-to-day work activities and products meet high standards, commensurate with the potential impacts of their work on safety; stopping work to resolve unexpected conditions, uncertainties, or unsafe circumstances; peer-checking one another's work; and holding one another accountable for safety behaviors on the job, such as following procedures.

As stated by the Institute of Nuclear Power Operations in its *Principles for a Strong Nuclear Safety Culture* (2004) (NRC Staff Exhibit 29), "everyone is personally responsible for nuclear safety." The exercise of this responsibility by the organization's members maintains a positive safety culture.

Q8. How quickly can safety culture change (improve or decline)?

A8. (Barnes) Theory and practice suggest that managing culture change may be difficult. Organizational change practitioners suggest that a wide-scope, significant culture change effort may require up to 5 years or more to complete (*cf*. Schneider, et al, 1996 (NRC Staff Exhibit 28)).

As previously discussed, safety climate is more transitory than safety culture. Experience in the nuclear industry has shown that single, highly visible events within an organization or work group have the potential to rapidly and adversely affect the safety climate. An example might be an incident in which is senior manager is perceived by the workforce to have retaliated against an individual for raising a safety concern. Overcoming those perceptions and re-establishing trust may require continued active interventions over 18 months or more.

In general, addressing localized safety climate issues or specific weaknesses in an area related to safety culture can be achieved in shorter time periods than attempts to implement a wholesale safety culture change.

Q9. Is it possible to predict future safety culture based on current performance?

A9. (Barnes) Predicting an organization's future safety culture is difficult because there are many unpredictable external and internal factors that will change the safety culture. The extent to which an organization's safety culture will remain stable depends on the external and internal influences to which the organization will be subject over time. External factors that can impact an organization's culture may include changes in the wider economy, corporate-level mergers and acquisition, or regulatory pressures. Internal influences may include leadership changes as well as changes in the workforce itself. For example, the nuclear power plant workforce has aged and is retiring. As a result, there is currently an influx of new personnel entering the industry. Although new personnel become acculturated to the organization they join, they also change it. Because the probability and nature of these types of pressures to change are difficult to foresee, it is unlikely that an organization's current safety culture predicts its future performance, particularly over the long term (e.g., beyond 5 years into the future).

NRC Safety Culture Policy Development

Q10. Why did the NRC first become concerned about safety culture?

A10. (Cai) The NRC has been concerned with elements related to safety culture since the 1979 Three Mile Island accident, although the term "safety culture" was not in use then. In 1989, in response to an incident at the Peach Bottom Nuclear Power Plant involving operators sleeping in the control room, the NRC issued a policy statement on the conduct of operations which describes the NRC's expectation that licensees place appropriate emphasis on safety in the operations of nuclear power plants ("Policy Statement on the Conduct of Nuclear Power Plant Operations," 54 FR 3424, January 24, 1989 (NRC Staff Exhibit 35)). In 1996, following an incident at the Millstone Nuclear Power Station in which workers were retaliated against for whistle-blowing, the Commission issued another policy statement ("Freedom of Employees in the Nuclear Industry to Raise Safety Concerns Without Fear of Retaliation," 61 FR 24336, May 14, 1996 (NRC Staff Exhibit 36)). That policy statement describes the agency's expectations that licensees establish and maintain a safety conscious work environment ("SCWE"), which is an environment in which employees are encouraged to raise safety concerns both to their own management and to the NRC without fear of retaliation. SCWE is an important element of a positive safety culture.

Q11. Do the Commission's regulations require licensees to maintain a positive safety culture?

A11. (Cai) The NRC does not have a specific regulation for safety culture. However, many of the proposed characteristics/traits being developed for the draft safety culture policy statement (see additional details regarding the characteristics/traits in A17) are embedded in NRC's regulations. For example, provisions protecting employees from discrimination for engaging in protected activities (related to supporting a SCWE) are in 10 CFR 50.7, and requirements for quality assurance programs are in Appendix B of Part 50 (related to identification and resolution of problems). In addition, elements of safety culture are addressed in the NRC's oversight of reactor licensees (see A19 about the ROP's treatment of safety culture).

Q12. What is the NRC's current policy on safety culture?

A12. (Cai) As stated in A10, the NRC has issued two policy statements in the past that are related to safety culture. The 1989 policy statement, "Policy Statement on the Conduct of Nuclear Power Plant Operations," places an emphasis on the personal dedication and accountability of all individuals engaged in any activity which has a bearing on the safety of nuclear power plants. The policy statement reads:

> The Commission has decided to issue this policy statement to help foster the development and maintenance of a safety culture at every facility licensed by the NRC, and to make clear its expectations of utility management and licensed operators in fulfilling NRC regulations and prior guidance regarding the conduct of control room operations.
>
> …
>
> Management has a duty and obligation to foster the development of a "safety culture" at each facility and to provide a professional working environment, in the control room and throughout the facility, that assures safe operations. Management must provide the leadership that nurtures and perpetuates the safety culture.

NRC Staff Exhibit 35 at 2.

The 1996 policy statement, "Freedom of Employees in the Nuclear Industry to Raise Safety Concerns Without Fear of Retaliation," describes the NRC's expectation that that licensees establish and maintain a SCWE. The policy statement states:

> The purpose of this Statement of Policy is to set forth the Nuclear Regulatory Commission's expectation that licensees and other employers subject to NRC authority will establish and maintain a safety-conscious work environment in which employees feel free to raise concerns both to their own management and the NRC without fear of retaliation. A safety-conscious work environment is critical to a licensee's ability to safely carry out licensed activities.

NRC Staff Exhibit 36 at 24337.

In response to Commission direction in SRM COMGBJ-08-0001, "A Commission Policy Statement on Safety Culture," (February 25, 2008) (NRC Staff Exhibit 37), the staff is currently in the process of updating the Commission's policy on safety culture (see A14 and A15).

Q13. How was the NRC's current policy developed?

A13. The 1989 policy statement, "Policy Statement on the Conduct of Nuclear Power Plant Operations," reads that it "is being issued to make clear the Commission's expectations of utility management and licensed operations with respect to the conduct of nuclear power plant operations." NRC Staff Exhibit 35 at 1. It describes how the NRC had received reports of operator inattentiveness and unprofessional behavior in the control room. It references several regulations and regulatory guidance where the Commission previously addressed expectations for operator conduct. It also provides "endorsement of industry initiatives to enhance professionalism by both management and plant operators." *Id.* The policy statement states, "The Commission has decided to issue this policy statement to help foster the development and maintenance of a safety culture at every facility licensed by the NRC, and to make clear its expectations of utility management and licensed operators in fulfilling NRC regulations and prior guidance regarding the conduct of control room operations" *Id*. at 2.

The 1996 policy statement, "Freedom of Employees in the Nuclear Industry to Raise Safety Concerns Without Fear of Retaliation," contains the following details about the development process:

> As a result of questions raised about NRC's efforts to address retaliation against individuals who raise health and safety concerns, the Commission established a review team in 1993 to reassess the NRC's program for protecting allegers against retaliation. In its report (NUREG–1499, 'Reassessment of the NRC's Program for Protecting Allegers Against Retaliation,' January 7, 1994) the review team made numerous recommendations, including several recommendations involving issuing a policy statement to address the need to encourage responsible licensee action with regard to fostering a quality conscious environment in which employees are free to raise safety concerns without fear of retribution (recommendations II.A–1, II.A–2, and II.A–4). On February 8, 1995, the Commission after considering those recommendations and the bases for them published for comment a proposed policy statement, 'Freedom of Employees in the Nuclear Industry to Raise Safety Concerns Without Fear of

> Retaliation,' in the Federal Register (60 FR 7592, February 8, 1995).
>
> The proposed policy statement generated comments from private citizens and representatives of the industry concerning both the policy statement and NRC and Department of Labor (DOL) performance.
>
> …
>
> In addition, two public meetings were held with representatives of the Nuclear Energy Institute (NEI) to discuss the proposed policy statement. Summaries of these meetings along with a revised policy statement proposed by NEI were included with the comments to the policy statement filed in the Public Document Room (PDR).
>
> This policy statement is being issued after considering the public comments and coordination with the Department of Labor.

NRC Staff Exhibit 36 at 24336-24337.

In developing the current draft policy statement (SECY 09-0075 (NRC Staff Exhibit 38)), the NRC staff considered a wide variety of information sources, including NRC lessons learned, domestic and international documents, organizational science literature, and other high reliability industry approaches. The staff also sought input and feedback from a wide range of stakeholders through presentations at stakeholder organization meetings, a *Federal Register* notice, a public workshop, and teleconferences with the Organization of Agreement State representatives and the Conference for Radiation Control Program Directors. The draft policy statement contains a proposed definition of safety culture and a set of characteristics of a positive safety culture (see additional details in A16 and A17).

After the Commission provided additional guidance in response to the draft policy statement the staff submitted in May 2009 (SRM-SECY 09-0075 (NRC Staff Exhibit 39)), the staff continued to engage in outreach activities. The NRC published the draft Safety Culture Policy Statement formally for public comment from November 6, 2009, through March 1, 2010, in the *Federal Register* (74 FR 5752, November 6, 2009; and 75 FR 1656, January 12, 2010). In February 2010, the staff held a public workshop to: (1) to develop a common definition of safety culture and a common set of descriptions/traits of what constitutes a strong safety culture and (2) to solicit input on the draft policy statement that had been published in the *Federal Register*. The workshop was a collaborative effort – the panelists and other participants represented a wide range of Agreement and non-Agreement State materials users, including reactor licensees, fuel cycle licensees, certificate holders, both medical and industrial materials users, a member of an Indian Tribe, and members of the public. The NRC and the Organization of Agreement States, as co-regulators, took a less active role during the workshop and allowed the workshop panelists to reach alignment with input from other meeting attendees on a high level definition of safety culture and a set of descriptions/traits. This process allowed staff to gain a fuller understanding of what is important to the various stakeholders as they endeavored to develop the terminology. Additionally, on the last day of the workshop, participants were able to provide their comments on the published draft policy statement.

Based on the public comments received on the draft policy statement, the products from the February 2010 workshop, and the additional input from other stakeholder outreach efforts, as well as consideration of the Commission guidance in the October 2009 SRM to SECY 09-0075 (NRC Staff Exhibit 38), the staff will develop a final draft policy statement and provide it to the Commission in early 2011. Included in the final draft policy statement will be a final definition of safety culture and set of characteristics/traits, which the staff is currently developing by taking into consideration the terminology in the draft policy statement and from the February 2010 workshop.

Q14. Has the NRC's policy on safety culture changed over time?

A14. (Cai) The previous two policy statements related to safety culture are described in A13. In February 2008, the Commissioners issued direction SRM COMGBJ-08-0001, "A Commission Policy Statement on Safety Culture," (NRC Staff Exhibit 37) regarding expanding the Commission's policy of safety culture. This followed the Davis Besse reactor vessel head degradation event in 2002, which led to the subsequent 2006 revisions to the ROP to better address safety culture (see additional details in A20 and A21). Specifically, the Commission's primary direction to the staff was (1) to expand the Commission safety culture policy to address the unique aspects of security; and (2) to ensure that the resulting policy would be applicable to all licensees and certificate holders. The focus on security followed the events of September 11, 2001, which had a significant impact on the way the NRC approaches regulation in the area of security. The addition of nuclear security to the safety culture policy statement emphasizes the importance the NRC places on security in the current climate. The focus on expanding the scope of the policy statement to apply to all NRC regulated activities provides clear recognition that safety culture applies to more than power reactors. In May 2009, the staff submitted its proposed draft Safety Culture Policy Statement, SECY 09-0075 (NRC Staff Exhibit 38), to the Commission. In response, the Commission provided additional guidance in October 2009 (SRM-SECY 09-0075) (NRC Staff Exhibit 39) (see more discussion in A15).

Q15. Is the NRC's policy on safety culture likely to change in the future?

A15. (Cai) As described A14, the Commission provided direction in February 2008 to expand the Commission's policy on safety culture. In response to this direction, in May 2009, the staff provided a draft policy statement to the Commission (SECY 09-0075 (NRC Staff Exhibit 38)), which contained the following key messages:

- Licensees and certificate holders bear the primary responsibility for the safe handling and securing of radioactive materials; therefore, it is each licensee's and certificate holder's responsibility to develop and maintain a positive safety culture in their organizations and among individuals who are overseeing or performing regulated activities. In this respect:
    - The draft policy statement addresses what is important in a positive safety culture, but does not address how licensees should implement the NRC's expectations of safety culture in their organization.
    - NRC encourages proactive initiatives by industry in this area.
- The NRC, as a regulator, has an independent oversight role, for example, through inspection and assessment processes.

In October 2009, the Commission approved publication of the draft Safety Culture Policy Statement in the *Federal Register* for public comment. In its SRM to SECY 09-0075, the

Commission provided additional direction to the staff on the content of the policy statement, including:

- The staff should consider incorporating suppliers and vendors of safety related components into the Safety Culture Policy Statement; and
- The staff should seek opportunities to comport NRC terminology, where possible, with that of existing standards and references maintained by those that NRC regulates.

NRC Staff Exhibit 39.

As part of this effort, the staff has been working, with stakeholder input, to develop a definition of safety culture and a set of descriptions/traits of what constitutes a positive safety culture that could be contained in the final policy statement. Additional details on the proposed safety culture terminology under development are described in A16 and A17.

Q16. How does the NRC currently define safety culture?

A16. (Cai) The 1989 policy statement, "Policy Statement on the Conduct of Nuclear Power Plant Operation," states, "the phrase 'safety culture' refers to a very general matter, the personal dedication and accountability of all individuals engaged in any activity which has a bearing on the safety of nuclear power plants." NRC Staff Exhibit 35

The proposed definition in the proposed draft Safety Culture Policy Statement (SECY 09-0075) is: "Safety Culture is that assembly of characteristics, attitudes, and behaviors in organizations and individuals which establishes that as an overriding priority, nuclear safety and security issues receive the attention warranted by their significance." NRC Staff Exhibit 38.

As part of the staff's efforts to further engage all NRC-regulated entities in developing the final Safety Culture Policy Statement, the NRC held a large public workshop in February 2010 (see additional details about the workshop in A13). The panelists, representing a diverse range of stakeholders, developed the following definition of safety culture: "Nuclear safety culture is the core values and behaviors resulting from a collective commitment by leaders and individuals to emphasize safety over competing goals to ensure protection of people and the environment."

The staff is currently in the process of evaluating the draft policy statement and workshop definitions, as well as sets of characteristics/traits (described in more detail in A17), with comments received during the public comment period, to develop a final set of terminology to propose to the Commission.

Q17. What are the elements of safety culture?

A17. (Cai) The draft Safety Culture Policy Statement (SECY 09-0075) includes the following characteristics of a positive safety culture:

- personnel demonstrate ownership for nuclear safety and security in their day-to-day activities;
- processes for planning and controlling work ensure that individual contributors, supervisors, and work groups communicate, coordinate, and execute their work in a manner that supports safety and security;
- the organization maintains a safety conscious work environment in which personnel feel free to raise safety and security concerns without fear of retaliation;
- the organization ensures that issues potentially impacting safety or security are promptly identified, fully evaluated, and promptly addressed and corrected commensurate with their significance; the organization ensures that the personnel, equipment, tools, procedures, and other resources needed to ensure safety and security are available; the organization's decisions ensure that safety and security are maintained;
- roles, responsibilities, and authorities for safety and security are clearly defined and reinforced; and
- the organization maintains a continuous learning environment in which opportunities to improve safety and security are sought out and implemented.

NRC Staff Exhibit 38

The panelists at the February 2010 workshop described in the response to A13 developed the following set of traits of a positive safety culture:

- the organization ensures that issues potentially impacting safety or security are promptly identified, fully evaluated, and promptly addressed and corrected commensurate with their significance;
- everyone is personally responsible for nuclear safety;
- processes for planning and controlling work activities are implemented such that safety is maintained;

- organizational learning is embraced;
- leaders demonstrate commitment to safety;
- effective communication is essential to maintain focus on safety;
- the organization maintains a safety conscious work environment in which personnel feel free to raise concerns without fear of retaliation; and
- trust and respect permeate the organization.

The staff is currently in the process of evaluating both sets of characteristics and traits, along with comments received during the public comment period, to develop a final set of terminology to propose to the Commission.

Q18. How is the NRC's policy on safety culture implemented for operating reactors?

A18. (Cai) Oversight of an operating reactor licensee's safety culture is implemented by the ROP. See A19 for the implementation details. Once the final Safety Culture Policy Statement is approved by the Commission and published, the staff will evaluate the agency's oversight programs, including the ROP, to identify if any changes would be needed to implement the expectations contained in the policy statement as appropriate for that type of licensee/certificate holder.

Implementation of NRC Safety Culture For Operating Reactors

Q19. How does the Reactor Oversight Process (ROP) capture safety culture?

A19(a). (Keefe) The Reactor Oversight Process (ROP) may use multiple inputs as indications into safety culture at NRC licensed facilities. These inputs include: NRC assessment and inspection findings and reports, licensee event reports (LERs) and root cause evaluations. In some cases, if more information about the health of the site's safety culture is needed, the NRC will also review licensee self and independent safety culture assessments. There are nine safety culture components and corresponding attributes which are located under the cross-cutting areas of human performance, safety conscious work environment, and problem identification and resolution to guide the identification of safety culture inputs. The ROP may use these cross-cutting areas to guide the identification and evaluation of safety culture issues.

A19(b). (Klett) The ROP is a risk-informed and performance-based oversight process, meaning that as licensee performance declines, the NRC increases its oversight, including a more in-depth review of safety culture.

The ROP provides for the oversight of a licensee's safety culture in four ways. First, the ROP provides for the review of a licensee's safety culture in a graded manner when that licensee has significant performance issues. The level of the staff's oversight is determined by the safety significance of the performance issues. This review and evaluation is described in the ROP's supplemental inspection program in Inspection Procedures ("IP") 95001 (NRC Staff Exhibit 22), IP 95002 (NRC Staff Exhibit 23), and IP 95003 (NRC Staff Exhibit 24). An IP 95001 inspection is usually performed when a licensee enters the Regulatory Response Column of the ROP Action Matrix (see NRC Staff Exhibit 25). This procedure requires NRC staff to verify that the licensee's root cause evaluation appropriately considered safety culture components. An IP 95002 inspection is usually performed when a licensee enters the Degraded Cornerstone Column of the ROP Action Matrix. This procedure requires the NRC staff to independently determine that the licensee appropriately considered whether any safety culture component caused or significantly contributed to any risk-significant performance issue. If a weakness in any safety culture component did cause or significantly contributed to such an issue, and the licensee's evaluation did not recognize this, then the NRC will request the licensee to perform an independent safety culture assessment. An IP 95003 inspection is performed when a licensee enters the Multiple/Repetitive Degraded Cornerstone Column of the ROP Action Matrix. When this occurs, the NRC expects the licensees to perform a third-party safety culture assessment. The staff will review the results of the assessment and perform sample evaluations to verify the results.

Second, the ROP's reactive inspection program evaluates a licensee's response to an event, including consideration of contributing causes related to the safety culture components, to fully understand the circumstances surrounding an event and its probable causes.

Third, the ROP provides continuous oversight of licensee performance as inspectors evaluate inspection findings for cross-cutting aspects. Cross-cutting aspects are aspects of licensee performance that can potentially affect multiple facets of plant operations and usually manifest themselves as the root causes of performance problems. A cross-cutting aspect is a performance characteristic that is the most significant contributor to a performance deficiency that resulted in an inspection finding. Inspectors will review available causal information to determine if the cause of an inspection finding relates to one of the cross-cutting aspects. A declining trend in licensee performance involving a cross-cutting aspect would warrant the identification of a substantive cross-cutting issue if the licensee is having difficulty correcting the trend.

Fourth, the ROP provides for the review of a licensee's safety culture if that licensee has difficulty correcting long-standing substantive cross-cutting issues. In these cases, the NRC will request the licensee to perform a safety culture assessment, and the NRC Staff will evaluate the results and the licensee's response to the results.

Q20. How was the ROP changed in 2006 to enhance treatment of safety culture?

A20. (Keefe & Klett) In 2004, the Staff received SRM-SECY 04-0111, "Recommended Staff Actions Regarding Agency Guidance in the Areas of Safety Conscious Work Environment and Safety Culture," (August 30, 2004) (NRC Staff Exhibit 42). In 2005, the Staff received SRM-SECY 05-0187, "Status of Safety Culture Initiatives and Schedule for Near-Term Deliverables," (December 21, 2005) (NRC Staff Exhibit 43). These SRMS directed Staff to:

- Enhance the Reactor Oversight Process (ROP) treatment of cross-cutting issues to more fully address Safety Culture,
- Ensure inspectors are properly trained,

- Develop a process for determining the need for a specific safety culture evaluation of plants with degraded performance, and
- Ensure modifications to the ROP are consistent with ROP development principles

Additionally, the SRMs directed the Staff to continue to monitor the industry's efforts. As a result, the Staff started a working group chartered with defining safety culture and developing the ROP changes. With input from internal and external safety culture experts, international research, and industry operating experience, the working group developed thirteen components of safety culture, nine of which are tied directly to the ROP cross-cutting areas. Of these thirteen components, nine are used in the baseline inspection program, and all thirteen are used in the supplemental inspection program. The cross-cutting components are reviewed on a semi-annual basis to determine if there's a trend in a licensee's performance in an aspect of the component. If the licensee has trouble correcting the trend, and if the trend persists for more than eighteen months, the NRC would typically request the licensee to perform a safety culture assessment. The components are discussed further in later responses.

Under the 2006 changes, the Staff's oversight of safety culture is applied in a graded manner, depending on the safety significance of licensees' performance issues, as described in the responses to A19. In 2006, the Staff updated several baseline and supplemental inspection procedures to incorporate the changes and the components, and enhanced the agency's inspector training program to ensure inspectors would be able to appropriately use the new components. The staff also revised reactive inspection procedures, which are performed to evaluate the licensee's response to events, to direct inspectors to consider contributing causes related to the safety culture components as part of their efforts to fully understand the circumstances surrounding an event and its probable causes.

Initial implementation of the ROP safety culture changes took eighteen months. After slight modifications based on stakeholder feedback, the final ROP changes were implemented in 2006.

Q21. Why was the ROP changed in 2006 to enhance treatment of safety culture?

A21. (Keefe & Klett) In 2002, severe boric acid corrosion of the Davis-Besse reactor vessel head was discovered, and a weak safety culture was found to have contributed in large part to this significant safety issue. The NRC's Davis-Besse lessons-learned task force and the General Accounting Office ("GAO") (now called the Government Accountability Office) conducted assessments of the environment (i.e., procedures, processes, programs, etc.) in which this significant degraded condition developed at Davis-Besse. The task force recommended that the NRC review its baseline inspections and plant assessment processes to determine whether they were sufficient to identify and appropriately disposition the types of problems experienced at Davis-Besse. Additionally, the task force recommended that the NRC provide more structured and focused inspections to assess licensees' employee concerns programs and safety conscious work environment. The GAO recommended that the NRC develop a methodology to assess licensee safety culture. Accordingly, the Commission directed the staff to:

- Include inspection requirements to evaluate a licensee's safety culture for plants with significant performance issues;
- Enhance the ROP's treatment of cross-cutting issues to more fully address safety culture; and
- Ensure that the safety culture enhancements were consistent with regulatory principles that guided the development of the ROP (i.e., that they be transparent, understandable, objective, predictable, risk-informed and performance-based).

In June 2006, the ROP was revised to more fully address safety culture. Regulatory Issue Summary 2006-13, "Information on the Changes Made to the Reactor Oversight Process to More Fully Address Safety Culture," dated July 31, 2006 (NRC Staff Exhibit 44), describes the changes made to the ROP to address safety culture, which were summarized in my responses to A19 and A20.

Q22. Is the ROP's treatment of safety culture likely to change in the future?

A22. (Keefe) The NRC has issued a draft safety culture policy statement which describes the agency's expectation that all NRC licensees and certificate holders maintain a positive safety culture. The final policy statement is expected to be issued in 2011. One expected outcome of the final policy statement initiative is the adoption of a common safety culture definition and terminology throughout the nuclear industry and the NRC. The subsequent implementation of the common definition and terminology is the responsibility of the NRC program offices. It is anticipated that NRR will adopt and replace the current safety culture definition in the ROP with the common definition and evaluate whether the components will need to be modified as a result of the final safety culture policy statement. The effective impacts will likely be in the area of inspector training and changes to the inspection manual chapters since the common definition and terminology will probably not change the intent of the definition and components in the current ROP.

Q23. Which ROP programs address safety culture?

A23. (Keefe & Klett) The ROP is comprised of several programs, including the assessment program (Inspection Manuel Chapter ("IMC") 0305 (NRC Staff Exhibit 10)), the performance indicator program (IMC 0608 (NRC Staff Exhibit 14)), the inspection program (IMC 2515 (NRC Staff Exhibit 8), IMC 0612 (NRC Staff 13)), and inspection procedures, the significance determination process (IMC 0609 (NRC Staff Exhibit 12)), the ROP self-assessment process (IMC 0307 (NRC Staff Exhibit 15)), inspector training program (IMC 1245 (NRC Staff Exhibit 17)), and the industry trends program (IMC 0313 (NRC Staff Exhibit 18)). The ROP addresses safety culture in its inspection, assessment, training, and self-assessment programs.

The ROP is a risk-informed and performance-based oversight process, meaning that as licensee performance declines, the NRC increases its oversight, including a more in-depth review of safety culture. The ROP framework describes three cross-cutting areas that contain the nine safety culture components. These cross-cutting areas, identified as human performance, problem identification and resolution, and safety conscious work environment, slice across each cornerstone of the ROP, meaning they can affect every aspect of reactor safety, radiation safety, and safeguards. The relationship between the three cross-cutting areas and the nine safety culture components is depicted in the chart below:

Cross-Cutting Area	Cross-Cutting Components
Problem Identification and Resolution	• Corrective action program • Self- and independent assessments • Operating experience
Human Performance	• Decision-making • Resources • Work control • Work practices
Safety Conscious Work Environment	• Environment for raising safety concerns • Preventing, detecting, and mitigating perceptions of retaliation

The ROP baseline inspection program requires inspectors to evaluate inspection findings to determine if any aspect of the nine cross-cutting components is applicable. The ROP supplemental and reactive inspection programs consider the nine safety culture components above and four additional safety culture components, which are: (1) accountability, (2) continuous learning environment, (3) organizational change management, and (4) safety policies. All thirteen components and their corresponding aspects are described in IMC 0310, "Components within the Cross-Cutting Areas" (NRC Staff Exhibit 19). In addition to considering all thirteen safety culture components, the supplemental inspection program also provides for the review of independent or third-party safety culture assessments obtained by the licensee.

The ROP assessment process looks at long-standing substantive cross-cutting issues to determine if safety culture assessments need to be performed and reviewed. The ROP training program provides safety culture-related training for inspectors. The ROP self-assessment program evaluates the perceived effectiveness of ROP safety culture enhancements. The NRC also may draw insights into a licensee's safety culture from the agency's allegation program.

Q24. How often does the ROP evaluate a plant's safety culture?

A24. (Klett) The ROP allows for continuous oversight of licensee performance and indications that may relate to safety culture. By evaluating baseline, reactive, and supplemental inspection findings for cross-cutting aspects and trending those aspects every six months, the NRC can determine if the licensee is effectively correcting any performance issues related to the safety culture areas and components. The NRC requests a licensee to perform a safety culture assessment if it has difficulty correcting a long-standing (i.e., 18 months) substantive cross-cutting issue. The NRC will evaluate the results of this assessment to determine that the licensee is taking adequate corrective actions to address the results. The NRC also monitors allegations, which may provide insights to a site's safety conscious work environment. Additional information is provided in the response to A25.

The ROP provides for non-routine reviews as well. Through its supplemental inspection program, the NRC will perform a more in-depth review of a licensee's safety culture if licensee performance declines. The NRC also looks into the safety culture components as they relate to the causes of events (e.g., an unexpected plant shutdown).

Q25. Which ROP baseline inspections are directed at safety culture? How often are they performed?

A25. (Klett) The ROP does not have a routine baseline inspection procedure for developing a conclusion about a licensee's overall safety culture. The supplemental inspection procedure that directs this type of evaluation, IP 95003 (NRC Staff Exhibit 24), is performed when a licensee has safety-significant performance issues and enters the Multiple/Repetitive Degraded Cornerstone Column of the ROP Action Matrix. Inspectors may also use aspects of this procedure to evaluate the results from safety culture assessments performed in response to long-standing SCCIs.

The NRC's baseline inspection program provides for continual oversight of cross-cutting aspects. Although it is not intended to provide a conclusion about a licensee's overall safety culture, the ROP baseline inspection procedure, IP 71152, "Problem Identification and Resolution," (NRC Staff Exhibit 21) evaluates licensees' corrective action programs, employee concerns programs, safety conscious work environments, and licensees' progress in addressing any cross-cutting themes. This procedure has four types of reviews: routine (daily) reviews, semi-annual trend reviews, annual sampling, and biennial team inspections. The routine, semi-annual trend, and annual sample reviews allow for the NRC to monitor a licensee's progress in addressing cross-cutting themes. The biennial team inspection looks at licensees' employee concerns programs, safety conscious work environments, cross-cutting themes, and periodic self-initiated and NRC-requested safety culture assessments.

Q26. How does the NRC determine if a licensee has a safety culture issue?

A26. (Keefe) The staff would typically rely on the results of a safety culture assessment to provide a conclusion about overall safety culture. The staff will either perform or request the licensee to perform a safety culture assessment in response to long-standing substantive cross-cutting issues ("SCCI") (i.e., older than 18 months) or safety-significant performance issues.

Q27. What is an SCCI?

A27. (Klett) An SCCI is defined in IMC 0305 (NRC Staff Exhibit 10) as a cross-cutting theme, about which the NRC staff has a concern with the licensee's scope of efforts or progress in addressing the cross-cutting theme. For the problem identification and human performance cross-cutting areas, a cross-cutting theme exists when multiple inspection findings (i.e., four or more) are assigned the same cross-cutting aspect within a one-year assessment period. A cross-cutting theme exists in the safety conscious work environment (SCWE) area if at least one of the following three conditions exists: (1) a finding has a documented cross-cutting aspect in SCWE and the impact on SCWE was not isolated, or (2) the licensee has received a chilling effect letter, or (3) the licensee has received correspondence from the NRC that transmitted an enforcement action with a Severity Level of I, II, or III, and that involved discrimination or a confirmatory order that involved discrimination.

Q28. What is an SCCI in the area of human performance?

A28. (Klett) A substantive cross-cutting issue in the area of human performance is a human performance-related cross-cutting theme, about which the NRC staff has a concern with the licensee's scope of efforts or progress in addressing the cross-cutting theme. In evaluating whether the NRC staff has a concern with the licensee's scope of efforts or progress in addressing the cross-cutting theme, the staff considers if any of the following situations exists:

- The licensee had not identified or recognized the cross-cutting theme(s) affected other areas and had not taken actions to address the theme(s).
- The licensee recognized the cross-cutting theme(s) affected other areas but failed to schedule or take appropriate corrective action.
- The licensee recognized the cross-cutting theme(s) affected other areas but did not implement timely corrective actions commensurate with the significance of the issue(s).
- The licensee has implemented a range of actions to address the crosscutting theme(s); however, these actions have not yet proven effective in substantially mitigating the cross-cutting theme(s) even though a reasonable duration of time has passed.

Q29. What is the significance of the Staff identifying an SCCI in the area of human performance?

A29. (Klett) The purpose of identifying an SCCI in the area of human performance is to inform the licensee on the docket that the NRC has a significant level of concern with the licensee's performance in the human performance cross-cutting area. SCCIs are not assigned a risk significance characterization, nor does the NRC implement enforcement actions for SCCIs. However, SCCIs are considered during the ROP assessment process. SCCIs are identified in publicly available assessment letters to licensees, when applicable. While the ROP Action Matrix does not prescribe NRC regulatory actions for SCCIs, they can influence the range of actions taken when Action Matrix thresholds are crossed. For example, the NRC may adjust the scope of a supplemental inspection performed in response to a safety-significant inspection finding to focus inspection efforts on the SCCI. The NRC may also focus baseline problem identification and resolution inspection samples on SCCIs. The NRC will monitor SCCIs to determine if licensees are correcting the human performance issues and whether the human performance issues continue to be a causal factor in inspection findings.

Q30. Does Staff identification of a SCCI in area of human performance indicate inadequate safety culture?

A30. (Keefe & Klett) The staff's identification of an SCCI in the area of human performance, in and of itself, does not indicate an overall inadequate safety culture. A human performance SCCI indicates a weakness in the licensee's scope of efforts to address a trend of performance deficiencies involving one of the human performance cross-cutting aspects. Because an SCCI focuses on only one cross-cutting area, and usually only one aspect in that area, the staff would not rely on an SCCI alone to provide an indication of an overall weak or inadequate safety culture. The staff would request a licensee to perform a safety culture assessment when that licensee has difficulty correcting a long-standing SCCI (i.e., an SCCI that is older than 18 months). The staff would typically rely on the results of this safety culture assessment to develop a conclusion about the licensee's overall safety culture.

Q31. How is a SCCI in the area of human performance addressed by the NRC?

A31. (Klett) If the NRC identifies an SCCI in the area of human performance, the NRC will issue a publicly available assessment letter to the licensee that summarizes the SCCI, how the staff will monitor the SCCI, and the criteria that must be met to close the SCCI. The staff will monitor the SCCI during its baseline problem identification and resolution inspections performed in accordance with IP 71152 (NRC Staff Exhibit 21) and at subsequent assessment reviews. At the next assessment review, the regional office will determine to close the SCCI or hold it open based on whether the licensee met the closure criteria. Examples of closure criteria may include fewer findings with the same aspect and increased NRC confidence in the licensee's ability to correct the SCCI.

In the second consecutive assessment letter identifying the same SCCI with the same cross-cutting theme, the NRC may request that: (1) the licensee provide a response at the next annual public meeting, (2) the licensee provide a written response to the SCCIs, or (3) a separate meeting be held with the licensee.

In the third consecutive assessment letter identifying the same SCCI with the same cross-cutting theme, the NRC may request that the licensee perform an assessment of safety culture. The regional office, in consultation with the NRR Division of Inspection and Regional Support Health Physics and Human Performance Branch staff, would review the safety culture assessment results and the licensee's response to the results. The NRC would document its conclusions regarding whether the SCCI closure criteria were met in next assessment letter.

Q32. How is an SCCI in the area of human performance addressed by the licensee?

A32. (Keefe & Klett) When the NRC identifies a substantive cross-cutting issue in the mid-cycle or annual assessment letter, the licensee would typically place this issue into its corrective action program or some other problem identification and resolution program (e.g., the employee concerns program), perform an analysis of causes of the issue, and develop appropriate corrective actions.

Q33. What does the NRC do to verify that safety culture weaknesses are addressed?

A33. (Keefe & Klett) The objective of safety culture in the ROP is to promote the early identification and correction of potential safety culture issues at a plant in order to prevent any further decline in the licensee's overall performance. When the NRC identifies a substantive cross-cutting issue in the mid-cycle or annual assessment letter, the licensee should place this issue into its corrective action program, perform an analysis of causes of the issue, and develop appropriate corrective actions. If a licensee conducts a safety culture assessment, the results would typically be added into the corrective action program to ensure that safety culture issues are identified and corrected. The NRC inspects and assesses the licensee's corrective action programs to verify use and effectiveness through the bi-annual IP 71152 Problem Identification and Resolution inspection (NRC Staff Exhibit 21).

The NRC will typically verify the effectiveness and sustainability of a licensee's corrective actions in response to identified safety culture weaknesses; however, the NRC's level of oversight to verify that licensees address safety culture weaknesses will depend on the licensee's unique situation and which aspect of safety culture is affected. The NRC will typically use IP 71152 (NRC Staff Exhibit 21) to verify that licensees are addressing safety culture-related weaknesses identified in safety culture assessments and long-standing cross-cutting issues. For licensees with safety-significant (i.e., greater-than-green) performance issues, the NRC's supplemental inspection program (e.g., IP 95003 (NRC Staff Exhibit 24)) is used to verify licensees are addressing any safety culture weaknesses identified in safety culture assessments.

Q34. Under the ROP when does the NRC require licensees to perform an independent safety culture assessment?

A34. (Keefe & Klett) The NRC does not require licensees to perform safety culture assessments. The NRC would request a licensee to perform an independent safety culture assessment under the following two circumstances:

- The NRC identified during an IP 95002 inspection (and the licensee did not recognize) that one or more safety culture component deficiencies caused or significantly contributed to the risk-significant performance issues, and
- A third consecutive assessment letter identified the same substantive cross-cutting component with the same cross-cutting area.

If the NRC requests a licensee to take an action, and the licensee refuses, the Agency can perform that action (i.e., the safety culture assessment) for them. This would mean that the NRC may perform that aspect of IP 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs or One Red Input" (NRC Staff Exhibit 24).

Q35. Under the ROP when does the NRC require licensees to obtain a third-party safety culture assessment?

A35. (Klett) The NRC does not require licensees to perform safety culture assessments. A licensee third-party safety culture assessment is performed by qualified individuals who are not members of the licensee's organization or utility operators of the plant. The NRC will request a licensee to perform a third-party safety culture assessment when it enters the Multiple/Repetitive Degraded Cornerstone or Unacceptable Performance Columns of the ROP Action Matrix and in preparation for the IP 95003 inspection.

Q36. Are there any examples whereby the ROP addressed indications of a declining safety culture?

A36. (Keefe) The ROP inspection process provides insights into aspects of safety culture at the nuclear power plants. These insights help to inform the inspection process of possible weaknesses in a licensee's safety culture. Safety culture is the responsibility of the licensee. The ROP may be useful in identifying cross-cutting issues that may reflect underlying organizational issues (including safety conscious work environment and perhaps even safety culture issues), however the ROP does not correct those issues. The NRC requires that licensee's have a corrective action program which ensures that conditions adverse to quality are promptly identified and corrected (10 CFR 50 Appendix B). If a licensee conducts a safety culture assessment, the results should be added into the corrective action program to ensure that safety culture issues are identified and corrected. The NRC inspects and assesses the licensees' corrective action programs to verify their use and effectiveness.

The NRC conducted an IP 95003 inspection at Palo Verde Nuclear Power Station in 2007. The inspection was the first one used after the changes to enhance safety culture were added to the ROP. This inspection was in response to repeated performance deficiencies at the site and was the first one conducted with the newly adopted safety culture components and aspects. The revised 95003 allowed the NRC to look more deeply into safety culture at a plant than ever before. The inspection staff conducted intensive interviews and focus groups with twenty percent of licensee personnel, participated in behavioral observations in the control rooms, and inspected the licensee's training program, corrective action program, and employee concerns program to gain a better understanding of the problems facing the site. The inspection report (NRC Staff Exhibit 45) provides more details about the results of the inspection.

Q37. Has the effectiveness of the ROP in addressing safety culture been assessed?

A37. (Klett) The ROP has a self-assessment process, which is described in IMC 0307 (NRC Staff Exhibit 15), to determine the ROP's effectiveness in achieving its goals of being objective, risk-informed, understandable, and predictable, as well as the applicable agency performance goals in the NRC's Strategic Plan. This process is also used to develop improvements to the ROP and to inform the Commission, NRC senior management, and the public of the results of the self-assessment. The staff continuously performs this assessment, and issues an annual Commission SECY paper. The ROP's safety culture oversight is one of the topics discussed in this paper.

The NRC reviews feedback from internal and external surveys – each type of survey is issued every other year – to determine if the ROP met its self-assessment process metric for the perceived effectiveness of the ROP safety culture enhancements (IMC 0307, App A (NRC Staff Exhibit 16)). The ROP also has a feedback form process that enables NRC staff to generate comments, questions, and recommendations for improving ROP guidance. The NRC's regional offices conduct ROP reliability initiatives to use cross-regional experience to identify best practices and any needed changes to the ROP, including those related to the SCCI process.

Prairie Island's Safety Culture

Q38. Would you agree that, as PIIC asserts, placement of Prairie Island Unit 1 in the Regulatory Response Column due to a White Finding in the Public Radiation Safety cornerstone in the first quarter of 2009 and a White Finding in the Mitigating Systems cornerstone in the fourth quarter of 2008 indicates that PINGP's safety culture is inadequate?

A38(a). (Keefe) No. I do not agree. Placement of a licensee into a Column of the Action Matrix is determined by inspection findings and performance indicators and is not necessarily indicative of inadequate safety culture. Safety culture is addressed through the use of cross-cutting issues which do not relate to the Action Matrix column that a plant may be placed in. More information, such as self and independent safety culture assessments, is needed for the staff to make an assessment as to the health of Prairie Island's safety culture.

A38(b). (Klett) I do not agree with PIIC's assertion. Movement across the ROP Action Matrix results only from safety-significant inspection findings and/or performance indicators, not from safety culture assessment results or SCCIs. Therefore, conclusions about a licensee's safety culture cannot be made based only on ROP Action Matrix movement. Movement to the Regulatory Response Column of the ROP Action Matrix triggers an IP 95001 inspection, which requires that inspectors verify that the licensee evaluated whether any of the safety culture components contributed to the performance issues. The NRC would not expect a licensee to have performed an independent or third-party safety culture assessment in preparation for this inspection.

The NRC supplemental inspection report 05000282/2009011 (NRC Staff Exhibit 46) documents the IP 95001 inspection performed for the fourth quarter 2008 White Mitigating Systems Cornerstone Finding. The inspectors found that the licensee identified concerns in three cross-cutting aspects, and the licensee assigned corrective actions to resolve these concerns. The inspectors determined that the licensee's evaluations included a proper consideration of whether a weakness in any safety culture component was a root cause or a significant contributing cause of the issue. The inspectors also documented that the licensee initiated a human performance improvement plan to address human performance issues at the plant.

The NRC supplemental inspection report 05000282(306)/2009015 (NRC Staff Exhibit 47) documents the IP 95001 inspection performed for the first quarter 2009 White Public Radiation Safety Cornerstone Finding. The inspectors found that the licensee identified concerns in five cross-cutting aspects, and the licensee assigned corrective actions to address these concerns. The inspectors identified one more cross-cutting aspect that had contributed to the issue; however, the licensee had taken corrective actions to address this aspect. The inspectors determined that the licensee's evaluations included a proper consideration of whether a weakness in any safety culture component was a root cause or a significant contributing cause of the issue. The inspectors also documented that the licensee initiated a human performance improvement plan to address human performance issues at the plant.

I do not agree with PIIC's assertion because the ROP Action Matrix Column designation is not dependent on the results of a safety culture assessment and the results of the above-mentioned supplemental inspection reports do not conclude that the licensee's safety culture was inadequate.

Q39. Would you agree that, as PIIC asserts, placement of Prairie Island Unit 2 in the Regulatory Response Column due to a White Finding in the Public Radiation Safety cornerstone in the first quarter of 2009 and a White Finding on Mitigating Systems cornerstone in the third quarter of 2009, indicates that PINGP's safety culture is inadequate?

A39(a). (Keefe) No, I do not agree. See A38(a).

A39(b). (Klett) I do not agree with PIIC's assertion. My response is similar to that to A38(b) regarding the white Public Radiation Safety Cornerstone finding. At the time of this written testimony, I have not had an opportunity to review the results of the supplemental inspection performed for the third quarter 2009 White Mitigating Systems Cornerstone Finding because the report has not been issued. I cannot agree with PIIC's assertion because the ROP Action Matrix Column designation is not dependent on the results of a safety culture assessment and the results of the above-mentioned supplemental inspection report does not conclude that the licensee's safety culture is inadequate.

Q40. Would you agree that, as PIIC asserts, that the NRC's identification of a substantive cross-cutting issue in the area of human performance in the mid-cycle performance review indicates that safety culture at Prairie Island Units 1 and 2 is inadequate (NRC Staff

Exhibit 48)?

A40(a). (Keefe) No, I do not agree. There is not enough information available in the inspection reports to make an adequate judgment on the overall health of Prairie Island's safety culture.

A40(b). (Klett) I do not agree with PIIC's assertion. As discussed in my response to A30, the staff's identification of an SCCI in the area of human performance, in and of itself, does not indicate an overall weak safety culture. A human performance SCCI indicates a weakness in the licensee's scope of efforts to address a trend of performance deficiencies involving one of the human performance cross-cutting aspects. Because an SCCI focuses on only one cross-cutting area, and usually on one component in that area, the Staff would not rely on an SCCI alone to provide an indication of an overall weak safety culture. The Staff would request a licensee to perform a safety culture assessment when that licensee has difficulty correcting a long-standing SCCI (i.e., an SCCI that is older than 18 months). The Staff would typically rely on the results of this safety culture assessment to develop a conclusion about the licensee's safety culture. Therefore, I do not agree with the PIIC's assertion because an SCCI alone would not provide enough information to draw a conclusion about a licensee's overall safety culture.

Q41. Would you agree that, as PIIC asserts, that concerns raised in the NRC's Biennial Problem Identification and Resolution Inspection Report (NRC Staff Exhibit 49) indicate inadequate safety culture at PINGP?

A41(a). (Keefe) There is not enough information available in the inspection reports to make an adequate judgment on the overall health of Prairie Island's safety culture.

A41(b). (Klett). I do not agree with PIIC's assertion. As previously discussed in my response A25, the biennial problem identification and resolution inspections do not provide an overall assessment of a licensee's safety culture. Inspection Report 05000282(306)/2009009 (NRC Staff Exhibit 49) documents the inspectors' conclusions about the licensee's corrective action program and safety conscious work environment. The inspectors documented concerns and findings related to the licensee's implementation of its corrective action program. The inspectors also concluded that "the licensee maintains an accessible, functioning [employee concerns] program, promotes a safety conscious work environment to employees, and periodically assesses employee attitudes though email surveys and a safety culture assessment by an outside team from the Utilities Service Alliance. Based on the [corrective action program documents] generated at the plant, discussions with employees, and survey results, the SCWE at the plant appeared adequate and no concerns were identified by the inspectors."

This report indicated weaknesses with the licensee's problem identification and resolution, which is one of the cross-cutting areas. If the licensee had difficulty correcting a long-standing substantive cross-cutting issue regarding an aspect of problem identification and resolution, then the NRC would request the licensee to perform a safety culture assessment. The staff's review of the results of this assessment would allow the staff to draw conclusions about the overall safety culture at Prairie Island.

Q42. In your opinion, is the ROP adequate to verify adequate safety culture at PINGP during the requested period of extended operation?

A42. (Klett) The ROP assesses current performance through inspection findings and performance indicators to verify that nuclear power plants are operated in a manner that provides adequate protection of public health and safety and the environment, and protection against radiological sabotage and the theft or diversion of special nuclear materials. As licensee performance declines, or if the licensee has difficulty in addressing long-standing substantive cross-cutting issues, the ROP provides for a more in-depth review of a licensee's safety culture. Therefore, consistent with the NRC's risk-informed and performance-based regulatory approach, the ROP is adequate to verify that licensees are ensuring an adequate safety culture. Because the ROP is continuously improving through its self-assessment and feedback programs, the ROP will remain adequate to verify that this licensee ensures an adequate safety culture during the period of extended operation and to respond accordingly if the licensee’s performance indicates otherwise.

Valerie E. Barnes, Ph. D
Statement of Professional Qualifications

CURRENT POSITION

Senior Technical Advisor for Human Factors	Division of Risk Assessment, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Rockville, MD

EDUCATION

B.A., Whitman College, 1977, Psychology
M.S., The University of Washington, 1981, Organizational Psychology
Ph.D., The University of Washington, 1985, Social Psychology

SUMMARY

Dr. Barnes has over 25 years' experience in consulting and conducting research related to human performance in complex technologies, with the majority of her work focused on the nuclear sector. Her primary areas of expertise are the human factors of judgment and decision-making at both the individual and organizational levels, emphasizing the enhancement of operational safety. Prior to joining the NRC in 2005, she managed or played a key technical role in numerous projects for the U.S. Department of Energy (DOE), the U.S. Nuclear Regulatory Commission (NRC), and other private sector and government sponsors undertaken to enhance the reliability of the human contribution to system performance. A large part of her early professional efforts were dedicated to identifying practices and problems with procedures for operations and maintenance tasks and developing standards and guidance for improving them. She has applied her human factors expertise in more than 100 audits, inspections and event investigations that have addressed a variety of human performance issues, including management and organization and safety management assessments. She has also developed and provided training to accident investigators to assess the organizational and institutional contribution to events. Dr. Barnes has also worked on site at nuclear power plants, providing technical assistance, mentoring and training to plant personnel. Since joining the NRC, a large portion of her work has focused on the agency's safety culture initiatives.

EMPLOYMENT

U.S. Nuclear Regulatory Commission, 2005 - Present

12/09/2008 to Present – Senior Technical Advisor for Human Factors, Division of Risk Assessment, Office of Nuclear Regulatory Research

- 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

- Advising the Office of Nuclear Reactor Regulation (NRR) on the scientific and technical validity of an industry-proposed approach to assessing and monitoring safety culture at commercial reactor sites
- Providing technical, scientific and policy analysis input to the Office of Nuclear Security and Incident Response and NRR on issues related to 10 CFR 26, “Fitness for Duty Programs”
- Primary author of Regulatory Guide 5.73, “Fatigue Management for Nuclear Power Plant Personnel”
- Acting as a technical resource to new/advanced reactor human factors research program in the Human Factors and Reliability Branch, DRA
- Representing the NRC on the OECD/NEA/CSNI Working Group on Human and Organizational Factors

11/06/2005 - 12/09/2008 – Senior Human Factors Analyst, Division of Risk Assessment, Office of Nuclear Regulatory Research

- Led the NRC’s first independent safety culture assessment at Palo Verde Nuclear Generating station under Inspection Procedure 95003
- Provided technical/scientific expertise to the internal Safety Culture Working Group, which was chartered to enhance NRR’s Reactor Oversight Process (ROP) by incorporating safety culture
- Provided input on lessons learned to revise the safety culture enhancements to the ROP following the first 18 months’ of implementation
- Primary author of 10 CFR Part 26, “Fitness for Duty Programs”

Performance, Safety and Health Associates, Inc (PSHA). President and Senior Research Associate, 1991-2005. Dr. Barnes founded and managed the operations of PSHA over a 14-year period while concurrently leading or providing technical expertise on a variety of research and consulting contracts to government and private-sector entities related to improving human safety performance.

Battelle Memorial Institute; Senior Research Scientist (1991-2005); Program Area Leader, Systems Performance Group (1989-1991); Research Scientist (1985-1991); Research Assistant (1982-1985). Dr. Barnes joined Battelle’s Human Affairs Research Centers in 1982 as a research assistant while completing her graduate studies, then was hired as a permanent research scientist after she completed her doctorate, assuming increasing responsibilities over time. She maintained her long-standing relationship with Battelle as a visiting senior research scientist after leaving that organization to form PSHA in 1991.

University of Washington, Organizational Research Laboratory. Research Assistant (1979-1985). Dr. Barnes assisted in and conducted research related to leadership effectiveness.

Central Breakthrough Methadone Maintenance Program, Seattle, WA. Assistant Director (1977-1980). This position entailed managing the administrative services provided by a methadone maintenance clinic as well as providing individual counseling and group therapy to opiate-abusers seeking recovery.

SELECTED PROFESSIONAL EXPERIENCE BY TOPIC AREA PRIOR TO 2005

Safety Culture, Organization and Management

Davis-Besse Nuclear Generating Station Organizational Improvement. Following discovery of significant boric acid corrosion on the reactor pressure vessel head, Davis-Besse management contracted with Dr. Barnes and her associates to (1) conduct the first safety culture assessment performed in the U.S. nuclear power industry; (2) improve the Station's corrective action program; and (3) investigate the factors that led to licensed operators' failure to take ownership of the plant safety in the events leading up to discovery of the corrosion.

Millstone Station's Employee Concerns Oversight Panel (ECOP). Dr. Barnes served as a consultant to and voting member of the ECOP, which was charged with evaluating the effectiveness of the Employee Concerns Program and monitoring the safety culture and workplace environment for raising concerns at the Station. She assisted the Panel in developing techniques for conducting targeted employee surveys and focus groups for identifying problem areas within the workforce and methods for assessing "customer satisfaction" of concerned individuals who had taken a concern to the ECP.

Management Assessments in Nuclear Power Plants. Dr. Barnes provided inspection support for the NRC in assessing the readiness to restart of Salem Nuclear Generating Station's Unit 2 in 1996 as a member of the management and organization subteam. In 1995, she served on the management subteam for a 21-member inspection team that was tasked with assessing the root causes for the cyclical performance noted at Dresden Nuclear Station. On that team, she assessed corporate financial resources, the Employee Concerns Program, bargaining unit relations and the licensee's plans for improving leadership performance.

Carolina Power and Light Engineering Assessment. Dr. Barnes performed an organizational analysis of the utility's corporate Chief engineering function and its relation to the Site Engineering organizations at three nuclear power plants. Recommendations were provided regarding staffing and staff training needs, as well as for enhancing the Chief Engineering function's effectiveness.

Pennsylvania State University Change Management. Dr. Barnes presented a series of workshops and provided change management consulting to PSU's Council of Deans and the Pattee Library staff. Faced with downsizing, relocations, reduced budgets and accelerated technology changes, overcoming resistance to change in the statewide Library system presented a significant organizational challenge.

Fermilab Tiger Team. As a member of the Management Subteam, Dr. Barnes assessed human resources programs at Fermilab for supporting environment, safety, and health activities at the lab. During this month-long, on-site assessment, Dr. Barnes evaluated Fermilab's recruiting program, performance evaluation system, staff qualifications, hiring procedures, training programs, staff attrition patterns, salary schedules, overtime records, performance incentive programs, human resources planning efforts, the Employee Assistance and Fitness-for-Duty Programs, and the lab's public affairs program. She also assisted in developing the Management Subteam's key findings and in identifying root causes for the problems identified. Follow-up activities included reviews of the lab's corrective action plan.

Leadership Theory and Cockpit Management. This project was conducted for NASA's Aviation Safety Reporting System (ASRS), with Dr. Barnes as project manager. It involved the preparation of a critical literature review that discussed the issues raised in a separate research paper summarizing ASRS reports pertaining to failures in cockpit resource management.

Evaluate Readiness of Bryan Mound Strategic Petroleum Reserve Site for Restart. As a member of a three-person team from DOE Headquarters, Dr. Barnes assisted in evaluating Bryan Mound's readiness for restart with an emphasis on leadership and organizational factors.

Human Factors

Provide Support for Nuclear Regulatory Commission Fitness-for-Duty and Access Authorization Rulemakings. Following the events of 9/11, the NRC determined that it was necessary to update its requirements for ensuring that personnel who may have access to nuclear materials are trustworthy and reliable. As project manager, Dr. Barnes provided technical assistance in conducting the background research required for the rulemakings, drafted rule text for 10 CFR Part 26 and 73.56, and provided support for numerous public meetings with stakeholders.

Update the NRC’s Human Performance Investigation Process (HPIP). This project was performed for the NRC’s Office of Nuclear Regulatory Research. The goal of the project was to update and refocus the process as guidelines for evaluating licensee event investigations and root cause analyses of operationally significant human errors. The guidelines describe typical causes of human errors in accidents (e.g., procedures, training, management, supervision) and provide a summary of the relevant research literature pertaining to each causal factor.

<u>DOE Accident Investigation Program</u>**.** Dr. Barnes has supported DOE's Office of Oversight Accident Investigation Program for over seven years. Her work entailed developing training materials for DOE Accident Investigation Board chairpersons, members and site points-of-contact. This included developing a curriculum, lesson plans, performance based scenarios, and testing and evaluation materials. She also served as an instructor and small-group facilitator for the program. She developed a white paper on methods to minimize post-traumatic stress disorders among witnesses to or participants in accidents, and served as a root cause analyst and consultant to DOE Type A Accident Investigation Boards.

<u>U.S. Chemical Safety and Hazard Investigation Board (CSB) Investigation Protocol</u>. Served as technical lead for a project to develop an incident investigation protocol and supporting procedures for the CSB.

<u>Nuclear Power Plant Event Investigations</u>. Conducted for the NRC's Office of Nuclear Reactor Regulation (NRR), this project involved two types of activities. The first was to develop a set of criteria for selecting events at nuclear power plants involving human performance deficiencies to be investigated by NRC personnel. The second was to provide human factors support for those event investigations. To-date, the events investigated have involved procedures, management communication practices, and staffing as causal or contributory factors in the events.

<u>Verbal and Written Communications Errors</u>. This project for the NRC's Office of Nuclear Regulatory Research entailed reviewing a large sample of Licensee Event Reports (LERs) and inspection reports to identify and characterize events resulting from verbal and written communications errors. The products of this project were a summary of lessons learned and guidelines for evaluating corrective action plans for communications errors.

<u>Develop a Conduct of Operations Inspection Protocol</u>. Conducted for the NRC's Office of Nuclear Reactor Regulation, this project involved developing inspection guidance and a protocol for evaluating operating crew command, control and communication practices over extended periods of observation. Originally developed for a special inspection at Washington Nuclear Project-2, the protocol was so well received by the users that PSHA was asked to revise it for generic application.

<u>Nuclear Power Plant Shift Staffing Levels</u>. As a subcontractor to Brookhaven National Laboratory, Dr. Barnes and her PSHA colleagues assisted in an assessment of the NRC's rule pertaining to minimum shift staffing levels for nuclear power plants. The work entailed an empirical assessment of different shift-staffing levels in simulator-based research, development and validation of a MicroSaint crew-size assessment model, and evaluation of licensee practices for staffing non-licensed operators. In a follow-on to this project, Dr. Barnes worked with the staff of Alion, Inc. to develop guidance for the review of licensee requests for exemptions from the NRC's shift-staffing requirements that the

NRC staff anticipated they would receive in applications for certification of new reactor designs.

Update the NRC's Knowledge and Abilities Catalog for Nuclear Power Plant Operators. Dr. Barnes led and played a key technical role in two projects, separated by 10 years, to update the existing catalogs of knowledge and abilities (K&As) used by NRC licensing examiners to develop reactor operator and senior reactor operator licensing examinations.

Advanced Digital Man-Machine Interface Analysis. Dr. Barnes assisted in an evaluation of the impact of digital control systems (DCSs) on operator performance, sponsored by Ryan Nuclear, Inc. This project involved a review of existing interface standards and a comparison of operator performance using graphical interfaces versus traditional, hard-panel displays and controls.

Developing Techniques to Evaluate Fitness-for-Duty Program Effectiveness. This project entailed providing technical assistance to the NRC in developing a fitness-for-duty rule for individuals at commercial nuclear power plants who are granted unescorted access to protected areas of the plants and conducting original research to identify and validate indicators of the effectiveness of the fitness-for-duty programs implemented at plants. Dr. Barnes managed this on-going project for two years while the NRC's first regulation on this topic was under development, drafted significant portions of the rule and associated guidance documents, and oversaw a team of 25 individuals who were responsible for analyzing and developing responses, in a five-week period, to the over 3,000 public comments received on the proposed rule.

Human Factors Affecting the Reliability and Safety of Liquefied Natural Gas Facilities: Computer/CRT Control and Display Systems. Dr. Barnes coordinated this project conducted for the Gas Research Institute. Assembled and edited a major review of the literature pertaining to Human-Computer Interfaces and developed human factors guidelines for computer workstation design, data entry, operator-computer dialogue, data display, and system hardware and software documentation.

Environmental Influences on Human Performance. This project for the NRC's Office of Regulatory Research involved an extensive review of the research literature pertaining to the effects on human performance of environmental stressors, such as excess heat or cold, vibration, low levels of illumination, and noise. The project resulted in a handbook for NRC inspectors to aid in assessing the potential safety implications of the exposures encountered by nuclear power plant personnel.

Aircraft Cockpit Distractions. This project was conducted for NASA's Aviation Safety Reporting System (ASRS), with Dr. Barnes as project manager. It involved a statistical analysis of a sample of ASRS reports in order to assess the impact of the FAA's Sterile Cockpit Rule, promulgated in 1983. Recommendations for further decreasing the frequency of incidents resulting from "attention mismanagement" in the cockpit were provided.

Procedures

Evaluation of Emergency Management Plans for a Large Chemical Manufacturing Facility. This project entailed a detailed assessment of the emergency response plans and procedures for a large chemical plant located in an urban area. The work involved site visits to conduct a hazards survey, document reviews and the development of enhanced training and emergency response procedures for the organization.

Emergency Operating Procedures (EOP) Inspections. Sponsored by the NRC's Office of Nuclear Reactor Regulation, Dr. Barnes and her PSHA colleagues provided human factors expertise to Regional inspection teams conducting follow-up assessments of licensees' EOP programs and the usability of their EOPs.

Preliminary Guidelines for the Review of Computer-Based Procedures. Conducted for the NRC's Office of Nuclear Reactor Regulation, this project entailed developing guidelines for staff reviews of computer-based procedure systems introduced into existing plants or incorporated in designs for advanced control rooms.

Develop Guidance for Upgrading Procedures in DOE Nuclear Facilities. This project involved developing guidance for procedure writers in the DOE complex to assist them in preparing both text- and graphic-format procedures for all types of facility work activities. This work resulted in a writers' guide for technical procedures, two draft documents describing the principles to be followed in presenting text and graphics in procedures, and a draft graphics procedure writers' guide. She also drafted a writers' guide for management system procedures.

Guidance for the Review of Procedure Upgrade Programs. This project represented the last in a series of studies conducted for the NRC regarding problems associated with operating procedures used at U.S. nuclear power plants. The purpose of the project initially was to develop the technical bases for review guidance for NRC inspectors to evaluate licensee programs to upgrade normal and abnormal operating procedures. However, the project scope was revised also to address procedures for low-power and shutdown operations. With Dr. Barnes as Principal Investigator, the project was conducted for the Human Factors Branch of the Office of Nuclear Regulatory Research.

Value-Impact Assessment of Upgrading Nuclear Power Plant Operating Procedures. This project involved conducting a cost-benefit analysis for the NRC's Office of Nuclear Regulatory Research to assess whether the NRC should implement regulatory action to specify requirements for the preparation of acceptable normal and abnormal operating procedures in U.S. nuclear power plants.

Procedure Violations. This project was conducted for the NRC's Office of Nuclear Regulatory Research to identify the extent, nature, causes and consequences of procedure violations in U.S. nuclear power plants, as a follow-up to the accident at Chernobyl in the U.S.S.R. Managed by Dr. Barnes, the work involved an extensive statistical analysis of existing event reports from U.S. plants pertaining to procedure-related incidents.

Techniques for the Preparation of Flowchart-Format Emergency Operating Procedures. With Dr. Barnes as Principal Investigator, this project involved the development of guidelines for presenting EOPs in flowchart format. Development of the guidelines required an extensive review of the research literature in several fields, the evaluation of existing flowchart-format EOPs, and the organization of the guidance into usable, handbook form. This project resulted in two volumes; one that set out specific guidance for preparing flowchart-format EOPs and a second that described the technical bases for the guidance presented in the first.

Assessing and Upgrading Operating Procedures for Nuclear Power Plants. Dr. Barnes assessed current problems with operating procedures in the U.S. commercial nuclear power industry for the NRC's Office of Nuclear Reactor Regulation. Project tasks included a review of the human factors literature relevant to procedures, structured interviews with nuclear power plant personnel, and the collection and evaluation of a large sample of written procedures.

Develop Recommended Interfaces Among Plant Operating Procedures. Based upon the operating experience and expert judgment of representatives of various industry groups (e.g., NRC Training Center staff in Chattanooga, utility and plant-level personnel, nuclear steam system supply (NSSS) vendors), Dr. Barnes oversaw the development of guidance for improving the usability of operating procedures by improving the interfaces between and among them.

Audit of Procedures Generation Package Implementation for Operating Reactors. As a technical contributor and then project manager for this NRC effort, Dr. Barnes conducted human factors reviews of selected licensees' EOPs and procedures programs to assess the effectiveness of the NRC's EOP upgrade requirements, as described in NUREG-0899, Guidelines for the Preparation of Emergency Operating Procedures.

Emergency Operating Procedures Program--Review of Operating Plants and Near-Term Operating Licensees. This project entailed the review of Procedures Generation Packages submitted by operating plants and near-term licensees to the NRC to evaluate their compliance with NRC regulations and guidelines. Dr. Barnes managed the team of technical communicators and human factors specialists responsible for conducting these reviews.

The Development, Use and Control of Maintenance Procedures in Nuclear Power Plants: Problems and Recommendations. Dr. Barnes provided leadership in conducting site visits to commercial power plants to gather data about maintenance procedures and summarized the results for the project final report.

Improving the Usability of Palo Verde Procedures. Managed by Dr. Barnes, this large project involved performing a systematic assessment of the Palo Verde operating procedures program, including administrative controls, and developing recommendations for improving it. Based on the assessment, the project team developed a training program for Palo Verde procedure writers, prepared writers' guides and a training handbook, and provided human factors assistance in the validation and verification of the new EOPs.

Improving the Usability of Crystal River Operating Procedures. This project entailed an evaluation of the Crystal River operating procedures program, the revision of their writers' guide, and the development and delivery of a training program for their operating procedure writers. As project manager and technical contributor, Dr. Barnes supervised the assessment of their program and procedures, oversaw the development of the training handbook, and participated in the delivery of the short-course.

Evaluating Palisades Emergency Operating Procedures. In response to the recommendations of an NRC inspection team, the Palisades emergency operating procedures writers' guide and the procedures themselves were reviewed from a human factors perspective, and recommendations for enhancements were provided.

Training Needs Assessment for Procedure Writers at DOE Nuclear Facilities. Conducted for the DOE's former Office of Policy and Standards, this project involved an assessment of the training needs of procedure writers and managers in DOE nuclear facilities as well as evaluating various alternative methodologies for training delivery.

PROFESSIONAL AFFILIATIONS

Past Chair, Human Factors Division of the American Nuclear Society
Member, IEEE Standards Committee SC-5, Human Factors

SELECTED PUBLICATIONS AND PRESENTATIONS

Barnes, V., Haagensen, B. & O'Hara, J. (2002, April) The Human Performance Evaluation Process: A Resource for Reviewing the Identification and Resolution of Human Performance Problems. NUREG/CR-6751. Rockville, MD: U.S. Nuclear Regulatory Commission.

Wieringa, D., Moore, C., & Barnes, V. (1999; 1993) Preparing Text Procedures: Principles and Practices. Columbus, OH: Battelle Press.

Echeverria, D., Barnes, V., Bittner, A., Fawcett-Long, J., Moore, C., Terrill, B., Westra, C., Wieringa, D. & Wilson, R. (1994) The Impact of Environmental Exposures on Human Performance, Vols. I & II. NUREG/CR-5680; PNL-7675; BHARC-700/91/005. Springfield, VA: National Technical Information Service.

Barnes, V., Desmond, P., Moore, C., O'Hara, J. (1996, September) Preliminary Review Criteria for Evaluating Computer-Based Procedures. Technical Report E2090-T4-2-9/96. Upton, NY: Brookhaven National Laboratory.

Barnes, V. & Haagensen, B. (1996, June) Draft guidelines for extended observation of control room command, control and communications. Prepared for the USNRC, June 30, 1996.

Barnes, V., Mumaw, R. & Schoenfeld, I. (1996, May) Communication errors in nuclear power plants. Proceedings of the 1996 American Nuclear Society International Topical Meeting on Nuclear Plant Instrumentation, Control and Human-Machine Interface Technologies. American Nuclear Society: La Grange Park, IL, (1), 671-678.

Barnes, V., Mumaw, R., DeBor, J. & Moore, C. (1995, September) Guidelines for Evaluating Corrective Action Plans for Communications-Related Events. Prepared for the USNRC, September 5, 1995.

Wieringa, D., Hill, S., Reese, W., Moore, C. & Barnes, V. (1992, December) DOE Standard: Writer's Guide for Technical Procedures. DOE-STD-1029-92. Germantown, MD: U.S. Department of Energy.

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Wieringa, D., Hill, S., Reese, W., Moore, C. & Barnes, V. (1991, December) DOE Writer's Guide for Technical Procedures. Germantown, MD: U.S. Department of Energy.

Barnes, V., Hill, S., Bogner, M., Degani, A., & Smillie, R. (1991, September) Task instructions and new technologies: Still a problem. Proceedings of the Human Factors Society 35th Annual Meeting. Santa Monica, CA: Human Factors Society, (2), 1199-1200.

Kantowitz, B., Triggs, T., & Barnes, V. (1990) Stimulus-response compatibility and human factors. In R. Proctor and T. Reeve (Eds.), Stimulus-Response Compatibility. Amsterdam, The Netherlands: Elsevier.

Grant T., Harris, M., Barnes, V., Larson, L. Thurman, A., & Weakley, S. (1989, November) Value Impact Assessment for a Candidate Operating Procedure Upgrade Program. NUREG/CR-5458; PNL-7123; BHARC-800/89/047. Springfield, VA: National Technical Information Service.

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Radford, L., Rankin, W., Barnes, V., McGuire, M. & Hope A. (1983) Drug and Alcohol Abuse: The Bases for Employee Assistance Programs in the Nuclear Utility Industry. NUREG/CR-3196, PNL-4679. Springfield, VA: National Technical Information Service.

Nealey, S., Pelto, P., Southwick, L. & Barnes, V. (1982) Factors Affecting Public Perception of Risks from Nuclear Power. Seattle, WA: Battelle Human Affairs Research Centers.

June Cai
Statement of Qualifications

Current Position

Senior Safety Culture Program Manager
Office of Enforcement

Education

B.S. Applied Psychology, 2002, Georgia Institute of Technology
M.A. Psychology – Human Factors & Applied Cognition, 2006, George Mason University
Additional course work in organizational psychology

Relevant Experience

Office of Enforcement **October 2009 - present**
Senior Safety Culture Program Manager

- Staff lead in advising, developing, and implementing activities for supporting and improving NRC's internal safety culture.
    - Leading implementation of Internal Safety Culture Task Force (2008-2009) recommendations.
    - Communicating and coordinating with a range of agency offices, groups, managers, and staff on internal safety culture activities.
    - Advising agency management and staff on issues related to internal safety culture to ensure consistency and quality of implementation.
    - Developing diverse training and communications products and providing briefings to various audiences.
    - Conducted detailed review and analysis of Office of Inspector General's (OIG) 2009 Safety Culture and Climate Survey results.
- Leading variety of continuous learning and improvement efforts on safety culture, including benchmarking other agencies and organizations, learning from operating experience from other industries, and training and development of new staff.
- Advising and participating in initiative to develop a Commission Safety Culture Policy Statement and common safety culture terminology.

Office of Enforcement **June 2007 – October 2009**
Enforcement Specialist (two temporary promotions: August – November 2008 to Task Force Team Lead, and May – August 2009 to Safety Culture Program Manager)

- Led Internal Safety Culture Task Force as Assistant Team Lead.
    - Designed data collection approach, made assignments, and conducted multiple data collection tasks.

    - Developed and implemented strategies and project plans, consolidated and reviewed results, and wrote the final report.
    - Facilitated task force meetings and discussions on controversial topics.
    - Briefed Commission and senior agency management on results and recommendations.
- Developed policy 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 Fuel Cycle Oversight Program and the Reactor Oversight Program.
- Led revisions to the safety culture portions of Inspection Procedure 95003.
- Conducted safety culture assessment activities and review of corrective action plans for the Palo Verde 95003 inspection in Fall 2007.
- Participated in Alternative Dispute Resolution mediation session with a licensee with safety culture issues, and conducted follow-up inspection activities to evaluate progress in implementation of agreement terms.
- Conducted analysis of treatment of safety culture by current fuel cycle inspection and oversight program and identified options for enhancement.
- Reviewed allegation data from multiple sources to identify trends. Developed input for the annual report.

Office of Nuclear Reactor Regulation **April 2003 – June 2007**
Human Factors Analyst

- Performed multiple onsite inspections of safety culture, safety conscious work environment, and human performance at nuclear power plants. Conducted on-site inspection, assessment, and documentation activities.
- Led agency efforts to develop multi-stage training for inspectors on safety culture. Activities included computer-based training, seminars, modifications to courses, and just-in-time training.
- Improved agency treatment of safety culture and safety conscious work environment for operating reactors by enhancing inspection and oversight program.
- Conducted evaluation of Institute of Nuclear Power Operations (INPO)'s safety culture assessment approach.
- Led group performing periodic operating experience reviews on human performance issues in the nuclear industry.
- Evaluated propose licensed amendments in human factors areas.
- Improved organizational effectiveness for office during rotational assignment.
    - Led projects in knowledge management area.
    - Initiated and coordinated seminars for supervisors and staff.
    - Improved internal communications.
- Evaluated OIG Safety Culture and Climate survey results and conducted focus groups to identify improvements. Implemented improvement activities.
- Upgraded the agency tracking system for human performance issues to improve functionality and usability. Improved contract requirements for the system maintenance.

Conducted data analysis in response to specific requests.

Office of Research
Risk and Reliability Analyst **May 2002 - April 2003**

- Developed database to track information on Human Reliability Analysis and conducted research to identify resources.
- Participated on research projects to identify best practices for conducting Human Reliability Analysis.

Molly J. Keefe
Statement of Professional Qualifications

Current Position:

Human Factors Specialist	Health Physics and Human Performance Branch Division of Inspections and Regional Support Office of Nuclear Reactor Regulation

Education:

B.A. Monmouth College, Monmouth, IL, 2000, Sociology and Anthropology
Associated Colleges of the Midwest Washington Semester Program, Washington, DC
Fall 1998, Anthropology and Environmental Studies
M.A. The American University, Washington, D.C. 2002, Sociology, Comprehensive Exam
Designations: Ph.D. pass

Experience in Safety Culture

U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation March 2010 -- present
Human Factors Specialist

- Serve as the Division expert for safety culture and participate in the NRC working group on safety culture policy
- Recommend changes to the Reactor Oversight Process (ROP) to reflect changes to NRC's safety culture policy and lessons learned through implementation of the ROP
- Present speeches at workshops and conferences on NRC activities in the area of safety culture.
- Advise staff in NRC regional offices on whether inspection findings are indicative of safety culture weaknesses

U.S. Nuclear Regulatory Commission, Office of New Reactors August 2007 – March 2010
Human Factors Analyst

- Assisted with development of new construction inspection program with an emphasis on safety culture.
- Participated in a task force to modify existing safety culture performance measures and inspection procedures for use with new reactor construction
- Assessed the applicability of internal safety culture best practices at other organizations for applicability to the agency

U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research *Human Factors Analyst* **October 2003 - August 2007**

- Served as project manager for contract to develop techniques to assess and mitigate insider threat
- Developed Guidance for the *Fitness for Duty/Managing Fatigue in Control Rooms* Regulation:
  Title 10 CFR 50 Part 26
- Researched international safety culture practices and programs to support the agency during development of Reactor Oversight Process changes enhancing safety culture.
- Designed survey instruments for assessing safety culture
- Served on Safety Culture Working Group to enhance and modify inspection process and procedures
- Identified measurable safety culture indicators for use in the inspection program
- Developed safety culture training for agency inspectors
- Developed and delivered training to commission emergency response employees in NRC's allegation process
- Led focus groups with commission employees as part of NRC's internal organizational culture assessments
- Conducted interviews with first-line supervisors and senior management teams during NRC's organizational culture assessments
- Inspected nuclear power plant human performance, organizational management, and safety culture at power plants in Ohio, Arizona, New Jersey, and Iowa:
    - Conducted focus groups and interviews with plant staff and management to understand and assess site organizational culture
    - Observed training and daily plant operations to assess conduct of business and identify indications of the organizations' safety culture
    - Analyzed data inputs and wrote organizational culture evaluations and inspection reports.

Audrey L. Klett

CURRENT POSITION

Reactor Operations Engineer, Performance Assessment Branch, Division of Inspection and Regional Support, Office of Nuclear Reactor Regulation, US Nuclear Regulatory Commission

Education

University of Illinois at Urbana-Champaign, Urbana, IL
B.S. in Electrical Engineering
Minor in Mathematics
Graduation: December 2002

Professional Experience

U.S. Nuclear Regulatory Commission (NRC) **2003-Present**

Reactor Operations Engineer, May 2008 to present
Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation

- Serve as the lead for the Reactor Oversight Process (ROP) Operating Reactor Assessment Program
- Served as the lead for the problem identification and resolution and supplemental inspection procedures from May 2008 to December 2009
- Served as the division lead for safety culture and substantive cross-cutting issues from September 2008 to December 2009
- Presented at NRC and industry workshops and to the Advisory Committee on Reactor Safeguards on implementation of safety culture in the ROP
- Assisted in the development of the draft NRC Safety Culture Policy Statement
- Developed ROP safety culture implementation training for inspectors

Electrical Engineer, October 2006 to May 2008
Division of Engineering
Office of Nuclear Reactor Regulation

- Developed safety evaluations for license amendments and backfit determinations
- Developed inspection procedures related to emergency diesel generator testing
- Served as the technical lead for emergency diesel generator and cable testing issues

Reactor Engineer Inspector, January 2003 to October 2006
U.S. Nuclear Regulatory Commission Region III Office
Division of Reactor Safety

- Performed inspections in the areas of problem identification and resolution, plant design, modifications, and fire protection
- Served on a rotation to the Point Beach Nuclear Plant Resident Inspector Office

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
NORTHERN STATES POWER COMPANY	)	Docket Nos. 50-282-LR/ 50-306-LR
	)	
(Prairie Island Nuclear Generating Plant,	)	
Units 1 and 2)	)	

AFFIDAVIT OF DR. VALERIE E. BARNES

I, Valerie E. Barnes, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Executed in Accordance with 10 CFR § 2.304(d)

Dr. Valerie E. Barnes
Senior Technical Advisor for Human Factors
Division of Risk Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Mailstop C-4A07M
Washington, DC 20555-0001
(301) 251-7585
Valerie.barnes@nrc.gov

Dated at Rockville, Maryland
this 30th day of July, 2010

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
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(Prairie Island Nuclear Generating Plant,	)	
Units 1 and 2)	)	

AFFIDAVIT OF JUNE CAI

I, June Cai, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Executed in Accordance with 10 CFR § 2.304(d)

June Cai
Senior Safety Culture Program Manager
Office of Enforcement
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Dated at Rockville, Maryland
this 30th day of July, 2010

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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	)	
(Prairie Island Nuclear Generating Plant,	)	
Units 1 and 2)	)	

AFFIDAVIT OF MOLLY JEAN KEEFE

I, Molly Jean Keefe, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Executed in Accordance with 10 CFR § 2.304(d)

Molly Jean Keefe
Human Factors Specialist
Health Physics and Human Performance Branch
Division of Inspection and Regional Support
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U.S. Nuclear Regulatory Commission
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Washington, DC 20555-0001
(301) 415-5717
Molly.keefe@nrc.gov

Dated at Rockville, Maryland
this 30th day of July, 2010

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
NORTHERN STATES POWER COMPANY	)	Docket Nos. 50-282-LR/ 50-306-LR
	)	
(Prairie Island Nuclear Generating Plant,	)	
Units 1 and 2)	)	

AFFIDAVIT OF AUDREY L. KLETT

I, Audrey L Klett, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Executed in Accordance with 10 CFR § 2.304(d)

Audrey L. Klett
Reactor Operations Engineer
Performance Assessment Branch
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Dated at Rockville, Maryland
this 30th day of July, 2010