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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)	)	

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) (Applicant Exhibit No. NSP000013), 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).