July 29, 2010

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)		
	)	Docket Nos.	50-282-LR
Northern States Power Co.	)		50-306-LR
	)		
(Prairie Island Nuclear Generating Plant,	)	ASLBP No.	08-871-01-LR
Units 1 and 2)	)		

TESTIMONY OF STEVEN C. SKOYEN ON SAFETY CULTURE CONTENTION

I. WITNESS BACKGROUND

Q1. Please state your full name.

A1. My name is Steven C. Skoyen.

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

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

Q3. Please summarize your educational and professional qualifications.

A3. I have approximately twenty years of professional experience in the identification of, and response to, equipment problems in nuclear power plants. In my current assignment, I am responsible for the technical oversight, strategic planning and improvement of ASME, NRC and INPO required programs with respect to plant equipment. I am also responsible for the identification and scheduling of component maintenance, testing and inspection

activities and for the coordination of the engineering response to equipment problems. I have performed similar duties in my previous employment with, among others, Westinghouse Electric Corporation, Nuclear Management Company, and the Formrite Tube Company. I received a Bachelor of Science Degree in Industrial Engineering from the University of Wisconsin in 1989.

A copy my resume is provided as Exhibit 1 (NSP000001) to this testimony.

II. PURPOSE OF TESTIMONY

Q4. What is the purpose of your testimony?

A4. The purpose of my testimony is to address the Safety Culture Contention submitted by the Prairie Island Indian Community ("PIIC") in this proceeding. As admitted by the Atomic Safety and Licensing Board ("Board"), the Safety Culture Contention reads:

> PINGP's safety culture is not adequate to provide the reasonable assurance required by 10 C.F.R. § 54.29(a)(1) that PINGP can manage the effects of aging during the requested period of extended operation.
>
> Order (Narrowing and Admitting PIIC's Safety Culture Contention) (Jan. 28, 2010), slip op. at 14.

Q5. What aspects of the Safety Culture Contention will you address in your testimony?

A5. I will address an aspect of NSPM's performance that has been cited by PIIC as indicative of the existence of a weak safety culture at the plant: the identification and correction of leakage

from the refueling cavity inside the containments of both PINGP units.

Specifically, I will summarize the history of observed refueling cavity leakage at PINGP and will describe NSPM's corrective actions, inspections, and repairs undertaken in response to such leakage. Next, I will explain NSPM's current understanding of the causes of the leakage and the inherent difficulty in pinpointing its exact sources. In addition, I will describe the results of both independent and internal engineering evaluations that have analyzed the safety significance of the leakage with respect to the integrity of the containment vessel and its structural components. Further, I will discuss NSPM's planned actions to resolve this issue by eliminating any further leakage in the future. Finally, I will describe NSPM's assessment of the safety culture implications of its response to the refueling cavity leakage issue and will respond to the issues raised in the safety culture contention submitted by PIIC.

Q6. What has been your involvement with the refueling cavity leakage issue?

A6. I first became involved with the refueling cavity leakage issue, with direct responsibility for its resolution, in December 2006. At the time, I was the Engineering Supervisor of the group that was tasked with investigating leakage into the Unit 2 Sump B during the 2R24 refueling outage. From that time forward, I have been involved in overseeing the causal investigations and the corrective actions taken towards eliminating the refueling cavity leakage.

III. HISTORY OF REFUELING CAVITY LEAKAGE ISSUE

Q7. Please describe the physical configuration of the containment of the PINGP units and the location of the refueling cavity.

A7. The containment for each unit consists of two systems: 1) a primary containment consisting of a free-standing, low-leakage steel vessel (including its penetrations, isolation systems, and heat removal systems) designed to withstand the internal pressure accompanying a loss-of-coolant accident ("LOCA"), and 2) a concrete Shield Building surrounding the primary containment. See schematic representation of the PINGP containment, attached hereto as Exhibit 2 (NSP000002). The primary containment, also referred to as the Reactor Containment Vessel ("RCV"), has steel cylinder walls, a hemispherical dome, and an ellipsoidal bottom. A five-foot-wide annular space exists between the RCV walls and those of the Shield Building, and a seven-foot clearance exists between the top of the vessel and the Shield Building roof dome, permitting in-service inspection and collection of any containment pressure vessel out-leakage of gases, which is permitted by and managed through filtration and ventilation in accordance with the NRC's regulations under 50 CFR Part 50 Appendix J. Exhibit 3 (NSP000003) (PINGP License Renewal Application) at 2.4-36.

The RCV internal structure is for the most part made out of reinforced concrete. It includes concrete floor slabs and compartments that support and protect the reactor pressure vessel ("RPV") and other components, and it provides the primary biological shield for the RPV. At various levels, the concrete slabs are supported by structural steel framing which is supported off the central concrete core and peripheral steel columns. The internal structure is supported by reinforced concrete placed in the bottom and knuckle region (the section of the containment between the containment vessel walls and the ellipsoidal bottom of the RCV). Id.

The refueling cavity is an area within the RCV surrounding the RPV. See Exhibit 2 (NSP000002). It contains the fuel transfer area and the reactor internals storage areas, and is located between the RPV and the spent fuel pool. Its function is to allow for refueling by transferring spent fuel from the RPV through the fuel transfer tube into the spent fuel pool in the auxiliary building, followed by transfer of fresh fuel assemblies from the auxiliary building back into the RPV. The cavity is approximately 60 ft. long by 22 ft. wide and is composed of concrete walls, inside of which is a stainless steel liner of approximately ¼ inch thickness. It contains the refueling cavity water during refueling operations. The liner in the refueling cavity is a non-safety-related Type III system, unlike the liner for the spent fuel pool. Because the liner was installed in segments, there are weld joints connecting the steel panels at the seams. In the upper region of the cavity, the reactor vessel flange seals the RPV from the refueling cavity through a segmented rubber seal at the head of the RPV. In the lower region of the cavity, the upper and lower reactor internals stands provide storage for the internals, and the transfer area contains components including the fuel transfer tube, the rod control cluster ("RCC") change fixture (a component located above the fuel transfer tube used for changing the control rods in the refueling cavity if such changing is not done in the spent fuel pool), and the RCC change fixture guide tube along the cavity wall.

The refueling cavity is filled with borated water during the refueling process, so that spent fuel may be transferred underwater from the RPV to the spent fuel pool, followed by transfer of new fuel back into the RPV, and then it is drained after refueling is completed. Typically, the refueling cavity remains flooded for 10-14 days during refueling. It is kept dry during the rest of the

unit's approximately 18-month operating cycle but cannot be accessed during regular plant operation because of the heat and radiation levels in that area of the RPV. It is not designed or evaluated to be flooded during regular plant operation. For these reasons, it is not possible to look for leakage from the cavity save immediately before the refueling cavity is flooded (approximately three days), or after the refueling operation is completed and the cavity is drained, but before plant operations resume (approximately seven days). These periods occur only once every approximately 18 months.

Q8. Please summarize the history of the refueling cavity leakage issue.

A8. PINGP operating personnel first observed indications of refueling cavity leakage in the containments of both PINGP units in 1987 and 1988. Leakage typically would begin two to four days after the refueling cavity was flooded and would end about three days after the cavity was drained. The leaking substance was determined to be borated water, such as is used to flood the refueling cavity. Over time, leaks were observed in sump B (the residual heat removal ("RHR") system LOCA recirculation sump), sump C (the sump under the RPV at the bottom of the containment), the regenerative heat exchanger room (located underneath the refueling cavity), the reactor coolant drain tank ("RCDT") space, and various walls, floors, and vaults. These findings suggested that the leakage was originating in the refueling cavity, although the leakage path was not immediately identified. Exhibit 4 (NSP000004) (Root Cause Evaluation Report 01160372-01) at 3; Exhibit 8 (NSP000008) (Dominion Engineering, Inc. Evaluation R-4448-00-01) at 3-1 to 3-2.

For leakage observed in sump C, possible leak source points in the upper refueling cavity included the reactor vessel cavity seal

(which seals the reactor vessel flange from the upper cavity floor), the sandplug covers (to allow access to the nozzles connecting to the reactor vessel), or the Nuclear Instrumentation System ("NIS") detector well cover plates. For leakage observed in sump B and the regenerative heat exchanger room, possible leak source points in the lower refueling cavity included the seams of the refueling cavity liner and embedment plates for various equipment supports. The estimated leak rate was, typically, one to two gallons per hour (gph). Exhibit 4 (NSP000004) at 3-6; Exhibit 8 (NSP000008) at 3-1 to 3-2.

In 1988, following the initial indications of refueling cavity leakage, we completed weld repairs on the Unit 1 cavity at the suspected locations of the leakage, that is, the cavity liner welds, which testing had shown to be leaking. Exhibit 4 (NSP000004) at 6.

Since 1988, plant records document that pumping of sump B and sump C in both units has been conducted, as needed, during refueling outages to remove any water that reached the sumps. Id. Pumping continued over the years as leakage was observed. During the 1998 Unit 2 outage, leakage was observed in sump B in the area outside the RHR penetration sleeve at the rate of 0.5 gph. A non-conformance report ("NCR") (an element of PINGP's Corrective Action Program ("CAP")) was prepared and issued to address this problem, consistent with plant operating procedures. Testing of the leaking water showed it to have levels of boron concentration indicative of refueling water. PINGP personnel promptly initiated a series of efforts during this outage to pinpoint the source of the leakage, including vacuum box testing of accessible seams and fasteners as well as dye penetrant testing of suspect areas that could not be vacuum box tested. Testing

revealed leakage from some of the sandplug cover cap screws. In addition, three small discontinuities in the liner plate seams were weld repaired. Id.

During this outage, NSPM evaluated the condition of the containment vessel wall by partially removing the grout (a thin mortar used for structural shaping purposes) around the penetration. The vessel showed no signs of degradation. To ensure that leakage had not adversely affected the integrity of the containment structures, NSPM commissioned Automated Engineering Services ("AES") to perform an evaluation of the effects of borated water on concrete, the reinforcing steel bars (rebar), and the RCV steel plate. Id. This evaluation, *Report on the Effect of Borated Water Leaks on Containment Concrete, Reinforcing Steel, and Containment Steel Plate*, issued December 16, 1998 (Exhibit 5 (NSP000005)), concluded that the effects of the leakage on the containment structures would be minimal and would have no safety significance. Exhibit 5 (NSP000005) at 4.

During the 1999 Unit 1 refueling outage, NSPM personnel observed similar leakage in sump B as well as cavity leakage through the ceiling of the regenerative heat exchanger room at the rate of 1.25 gph. Vacuum box testing of the refueling cavity in Unit 1 revealed no leakage indications other than in the sandplug cover screws. Exhibit 4 (NSP000004) at 6.

Concurrent with its procedures to diagnose the source of observed leakage and evaluate its potential effects, PINGP employed various methods to prevent the leakage, including installation of a strippable liner (brand name InstaCote) to the refueling cavity, starting during the 2000 Unit 2 outage and continuing through the 2003 Unit 2 outage. These applications had inconsistent results –

sometimes they succeeded in preventing leakage from occurring, while other times the leakage occurred despite the coating application. Id. at 6-7.

Starting in 2004, we began caulking suspected leak paths at the baseplates and fasteners of the internals stands and the rod control cluster ("RCC") change fixture. Id. at 7. The baseplates for this assembly (which floods along with the cavity) have supports that lie on the floor of the transfer canal. These baseplates bear most of the force of the water pressure on the structures in the transfer canal and refueling cavity.

Our caulking process consisted of applying caulk underneath the nuts to seal the points at which the anchor bolts penetrate the embedment plates, and at the joints of the baseplates and embedment plates. After similar caulking steps were performed during the 2005 Unit 2 outage, the 2006 Unit 1 outage, and the 2008 Unit 1 outage, no leakage from the refueling cavity was observed. However, during the 2006 Unit 2 outage, leakage was observed through the grout in sump B. In the next Unit 2 outage in 2008, we repeated the caulking procedure but did not remove the nuts due to the risk of their galling (locking up); instead, we caulked around the tops and edges of the nuts. During this outage, leakage was reported in the ceiling of the regenerative heat exchanger room, the 22 vault (where steam generators and reactor coolant pumps are located), and sump B. Id. at 7-8.

In 2006, PINGP requested that AES re-review the cause and potential safety significance of the Unit 2 sump B leakage relative to its earlier evaluation performed in 1998. The resulting evaluation (Exhibit 6 (NSP000006)) confirmed that the basis and conclusions of the 1998 report remained valid, and that the

integrity of the concrete, rebar, and containment shell had not been compromised. Exhibit 6 (NSP000006) at 1-2.

Because of the limited success of the measures implemented until then to address the leakage, a corrective action document identifying the problem, CAP 1160372 (Exhibit 7 (NSP000007)), was initiated in the fall of 2008 recommending performance of a Root Cause Evaluation ("RCE") to evaluate the leakage problem and develop a permanent solution to it. A comprehensive evaluation was made in 2009, and a Root Cause Evaluation Report, RCE 01160372-01, *Refueling Cavity Leakage, Event Date 1988-2008* (Exhibit 4 (NSP000004)) was prepared and issued.

In addition, PINGP further updated the 1998 AES evaluation of the potential effects of borated water on the containment vessel by commissioning a second outside expert study from Dominion Engineering, Inc. ("DEI"), which was completed in February 2009. The DEI evaluation, R-4448-00-01, *Evaluation of Effect of Borated Water Leaks on Concrete, Reinforcing Bars, and Carbon Steel Plate of the Containment Vessels at Prairie Island Units 1 and 2* (Exhibit 8 (NSP000008)), concluded that there was no evidence of significant corrosion of the steel containment vessel, that there was no evidence of significant degradation of concrete, and that any corrosion of rebar in the concrete would be minor. Exhibit 8 (NSP000008) at 2-2 to 2-3.

The RCE and two subsequent engineering evaluations (EC 14139 and EC 15044) were commissioned to identify the sources of the leakage and assess the potential effects of borated water leakage on the containment vessel, concrete, and concrete reinforcing bar. The evaluations involved a wide-ranging analysis of the history of observed leakage; the likely leakage source(s); the nuclear safety

significance of the leakage, including the impact of continued leakage; an analysis of relevant operating experience; and recommendations for corrective actions, including effectiveness reviews. Exhibit 4 (NSP000004) at 3-5, 30. The RCE, completed in April 2009 and updated in February 2010 to reflect results of fall 2009 Unit 1 repairs, determined that the most likely sources of lower cavity leakage were the floor embedment plates for the reactor vessel internals stands and the RCC assembly change fixture, as well as the RCC change fixture guide tube supports located on the cavity wall. Id. at 5.

In the fall of 2009, NSPM completed permanent repairs to the Unit 1 floor embedment plates for RCC change fixture supports and internals stands supports by removing existing nuts, replacing them with blind nuts that were seal-welded to the baseplate, and applying a seal weld between the baseplate and the embedment plate. During refueling cavity flooding following these repairs, we found minor leakage in the regenerative heat exchanger room. We identified the guide tube supports as the likely leakage source due to the similarity of their design to that of the internals stand supports. Id. at 3, 8.

Prior to cavity flooding during the spring 2010 Unit 2 outage, we similarly completed a series of repairs to the Unit 2 reactor vessel internals stands supports and RCC assembly change fixture supports, in addition to the RCC assembly guide tube supports. Following these repairs, we observed leakage in sump B almost immediately after the cavity was flooded. Previously, this type of leakage would not appear for several days. The leak initially occurred at a rate of 0.8 gph which decreased gradually over a few days to 0.02 gph. This change in leak volume was also a departure from previously observed leaks, which tended to remain

steady while the cavity was flooded. By the time the refueling cavity was drained, leakage had diminished almost entirely in sump B. Because essentially no leakage was observed in the regenerative heat exchanger room ceiling, we inferred that lower cavity areas were not causing this leak. We believe that this leakage originated in the upper refueling cavity, from faulty seals at the sandplug covers, and was migrating into sump B either before or after reaching sump C. This newly identified potential leak path is significant because we previously had not associated leakage into sump B with sources in the upper cavity, as will be discussed in greater detail below. Exhibit 9 (NSP000009) (CE 1233806-2) at 1; Exhibit 10 (NSP000010) (EFR 1160372-04) at 1.

Q9. Why was NPSM unable for several years to establish the exact cause of the leakage?

A9. It is very difficult to establish with complete assurance what the causes of the leaks are. Leaking occurs only (if at all) for a few days every eighteen months, during the refueling process, and the areas where the leakage occurs are partially or totally inaccessible. None of the suspected leakage points are accessible at the time they are leaking. One cannot access the refueling cavity at all while the reactor is at power. The cavity is similarly inaccessible while flooded, making it impossible to observe precisely which points are responsible for leakage. It is accessible only during outages, in the few days before and after cavity flooding. The space in the upper cavity located directly underneath the sandplug covers is accessible only by removing the sandplugs, an operation that is scheduled to be performed only once every ten years. Sump C is not accessible while the cavity is flooded due to high radiation levels and can only be accessed for a period of approximately seven days after the upper cavity is drained. Thus, during refueling, we can only monitor leakage

indirectly based on the volume of water pumped out of sump C in response to the sump level alarm. Sump B remains accessible either directly or through the inspection opening for the majority of the outage.

As a result of our inability to access the leak source points during the refueling process, we must look for evidence of where leakage collects and identify probable sources based on inspecting and repairing possible leak points, followed by testing the effectiveness of those repairs by observing any leakage during the next flooding.

Due to the limited accessibility of relevant components, the refueling outage allows only limited opportunities for making repairs and performing inspections. There is only a few-day period at the beginning and end of each outage in which these actions may be taken. During the last two outages, PINGP personnel have allocated as much time as possible at the beginning of the outages in order to carry out repairs.

Q10. What is NSPM's current understanding of the cause of the leaks?

A10. Our understanding is that there are two leakage sources, corresponding to locations in the lower cavity and the upper cavity. The lower cavity leakage, which accounts for leakage entering sump B and the regenerative heat exchanger room, occurs where the internals stands and RCC change fixture anchor studs penetrate the associated embedment plates on the refueling cavity floor, and where anchor studs penetrate the RCC guide tube supports on the cavity wall. Exhibit 4 (NSP000004) at 3-4. The anchor studs are secured to the embedment plates by being set in through-holes in the plates and then seal-welded to the plates. These components have welds between the studs and embedment

plates that are designed to form a watertight seal. Although inspection of the condition of the seal welds is difficult due to their inaccessibility, we believe that they have developed pin-hole-sized leaks or cracks. Those cracks may be due to corrosion, fatigue, or construction defects, and if they exist they create a leak path along the threads of the studs, which then allows water to flow under the cavity liner into cracks in the concrete and down, emerging in the ceiling and walls of the regenerative heat exchanger room and eventually through the inner wall of the containment vessel. Once at the containment vessel, the water travels down and horizontally, potentially filling any voids between the containment vessel and concrete down to the low point of the bottom head of the containment vessel. As the water then rises, it starts to leak through various construction joints, cracks, and the grout in sump B. Id. at 3.

The upper cavity leakage, which we believe accounts for the leaked refueling water entering sump C and the leakage entering sump B during the Spring 2010 Unit 2 outage, results from faulty seals at the sandplug covers, allowing refueling water to reach sump C at the bottom of the reactor containment. This leakage from the upper cavity likely enters sump B indirectly through one of the construction joints. Id. at 6; Exhibit 9 (NSP000009) at 1-2.

Q11. What actions has NSPM taken to address the leakage issue since the RCE was completed?

A11. In the fall of 2009, NSPM undertook permanent repairs to the Unit 1 floor embedment plates for the RCC change fixture supports and the upper and lower internals stands supports by removing existing nuts, replacing them with blind nuts that are seal-welded to the baseplate, and applying a seal weld between the baseplate and the embedment plate. To ensure the quality of

the repair, we performed a visual inspection and a dye penetrant inspection. Exhibit 4 (NSP000004) at 8. During this outage, we conducted additional dye penetrant inspections of the embedment plate to liner welds and transfer tube welds to identify any potential additional sources of leakage. We also performed vacuum box testing of the weld seams on the lower cavity floor and about six feet up the cavity liner wall. Id. at 12. We plan to repair the RCC assembly guide tube supports during the next Unit 1 outage in order to eliminate the minor remaining leakage observed in the regenerative heat exchanger room.

During the spring 2010 Unit 2 outage, NSPM made permanent weld repairs to the reactor vessel internals stands and RCC change fixture embedment plates, as it did for Unit 1 in the fall of 2009. Repairs were also made to the unit's RCC assembly guide tube supports for all embedment plates, and a visual and dye penetrant examination of the embedment plate to liner welds was performed. We carried out the same fuel transfer tube weld examinations and vacuum box testing of refueling cavity liner plate seam welds that we performed for Unit 1 a year earlier. Exhibit 9 (NSP000009) at 1-2.

The spring 2010 Unit 2 repairs appeared to successfully correct the leakage in the lower cavity. We attribute the minor continued leakage in the upper cavity to leaks in the sandplug covers, consistent with the results of our post-refueling testing. Based on our discussions with other sites concerning their operating experience with sandplug covers, we have entered work requests to install gaskets for the covers at the next outage instead of continuing to caulk the covers because we believe the use of gaskets is a more effective long-term solution. Id. at 2.

Q12. How successful were these repairs in mitigating the leakage?

A12. The fall 2009 repair to the Unit 1 floor embedment plates eliminated 95 to 97.5 percent of the leakage historically experienced from the lower cavity. Following these repairs in Unit 1, only minor leakage occurred on the ceiling of the regenerative heat exchanger room, after the cavity had been flooded for over 14 days, in the amount of 0.05 gph, or roughly seven drops per minute. No leakage was observed in sump B. Exhibit 4 (NSP000004) at 3, 8. In response to this minor continued leakage in the regenerative heat exchanger room, we performed expanded inspections, including examination of the liner plate to floor embedment plate fillet welds and transfer tube welds. Although the inspection identified one porosity indication not believed to be a likely source of leakage, the weld will be repaired during the next Unit 1 refueling outage. Id. at 12.

Following the fall 2009 Unit 1 repairs, vacuum box testing of the seam welds of the liner plate in the lower cavity revealed no leakage. Examination of the transfer tube welds, including both dye penetrant and visual inspection, showed no indications of leakage. Inspection of the lower cavity presented no depressions or soft areas indicative of water having eroded the surface. There is no evidence of leakage having reached the containment vessel or the steel pressure vessel. Id.

The spring 2010 Unit 2 repairs similarly appear to have resolved the leakage from the lower cavity, eliminating over 97.5 percent of the total leakage historically experienced. During the spring 2010 Unit 2 outage, less than 0.01 gph of leakage, or roughly 1 drop per minute, was observed coming from the mezzanine adjacent to the regenerative heat exchanger room. Upon cavity flooding, leakage occurred in sump B at an initial rate of 0.8 gph,

which gradually decreased to essentially no leakage by the end of refueling once the cavity was drained below the upper cavity. Exhibit 9 (NSP000009) at 1.

Q13. To what do you attribute the continuing minor leakage in the Unit 1 regenerative heat exchanger room?

A13. We believe the source of this minor leakage in Unit 1 to be the embedment plates of the RCC change fixture guide tube supports located on the cavity wall. Exhibit 4 (NSP000004) at 3. These guide tube supports have a design similar to the supports for the internals stands and RCC change fixture. The following actions are planned to eliminate this remaining minor leakage: During the next Unit 1 refueling outage, we will make repairs to the RCC change fixture guide tube supports by removing existing nuts, replacing them with blind nuts that will be seal-welded to the baseplate, and applying a seal weld between the baseplate and the embedment plate. These repairs were successfully performed during the spring 2010 Unit 2 refueling outage, eliminating greater than 97.5 percent of the total leakage historically experienced. During the spring 2010 Unit 2 outage, less than 0.01 gph of leakage, or roughly 1 drop per minute, was observed coming from the mezzanine adjacent to the regenerative heat exchanger room. We expect that successful repairs to the guide tube supports will similarly resolve the remaining Unit 1 lower cavity leakage.

Q14. To what do you attribute the continuing minor leakage in sump B of Unit 2?

A14. We attribute this minor continued leakage to leaks in the sandplug covers in the upper cavity, consistent with the results of our post-refueling testing. We are confident that leakage observed in sump B of Unit 2 during the spring 2010 refueling outage was coming from the upper cavity only based on three factors: 1) the

successful completion of lower cavity permanent repairs; 2) the significant leakage shown by the sandplug covers when we conducted vacuum box tests after the cavity was drained; and 3) the variation of leak rate, suggesting that there was a mechanical joint which was changing, consistent with sandplug covers settling based on water pressure, in contrast to a crack or pinhole leak which would not change. At the end of the 2010 Unit 2 outage, our testing showed significant leakage in the sandplug covers, and we conducted extensive vacuum box testing of seam welds in both the upper cavity and up to 18-20 ft. high in the lower cavity. In order to prevent further sandplug cover leakage, we have entered a work order to install gaskets to seal the sandplug covers during the next outage in both units, which we believe will provide superior sealing protection relative to the current caulking procedure. Exhibit 9 (NSP000009) at 1-2; Exhibit 10 (NSP000010) at 1.

Q15. Has NSPM evaluated the potential effects of the leakage over the years on the integrity of the containment vessel?

A15. Yes. As we sought to eliminate the leakage, we also set out to investigate its potential safety significance. In 1998, we retained Automated Engineering Services ("AES") to evaluate the effects of borated water on the concrete, the reinforcing bars, and the containment vessel. AES's evaluation concluded that the effects of the leakage on the containment structure would be minimal and would have no safety significance. Exhibit 5 (NSP000005) at 4. This conclusion was confirmed by a 2006 AES follow-up evaluation (Exhibit 6 (NSP000006)) and by a new 2009 comprehensive evaluation by DEI (Exhibit 8 (NSP000008)). The lack of adverse effects on the concrete, rebar, and the reactor vessel was also confirmed at various times thereafter through visual observation.

Q16. What were the inspection findings on the condition of the concrete?

A16. The inspections conducted to date have found no evidence of degradation of the concrete surrounding the steel containment vessel. Exhibit 4 (NSP000004) at 4. We have evaluated potential degradation based on the leakage, assuming continuous wetting since the time of plant startup, even though we have no evidence that such leakage had occurred prior to 1987. Multiple independent engineering evaluations have concluded that exposure of the containment vessel and structures to refueling cavity water has not had an adverse impact on their compliance with design requirements, and that any potential degradation would be of low safety significance. Id. The NRC Staff has agreed with this evaluation in its October 2009 Safety Evaluation Report ("SER") (Exhibit 14 (NSP000014)), finding that "any loss of load carrying capacity of the concrete would be negligible since the concrete sections are four to five feet thick." Exhibit 14 (NSP000014) at 3-147.

The exposure of concrete to borated water is estimated to have affected the concrete under the cavity liner to a depth of no more than 0.31 inches. There has been no observed thinning or corrosion of the concrete. Although this is an insignificant amount of thinning in most areas, we will continue to evaluate it, especially in areas where concrete in contact with the liner is thin at the wall near the transfer tubes. Exhibit 4 (NSP000004) at 26-27. This thin area is insignificant for structural integrity purposes because it does not support any other components.

The 2009 DEI evaluation found no evidence of any significant degradation of the concrete due to borated water leakage. The estimated maximum credible degradation of 0.31 inch would not significantly affect the structural integrity of the refueling cavity

floor or wall, with the possible exception of some thinning in the area near the transfer tube. This area was evaluated as part of the Margin Assessment of Containment Vessel and Concrete Structures (EC 15651) (Exhibit 12 (NSP000012)), which determined that the wall thickness degradation postulated by the DEI evaluation would not challenge the functionality or integrity of this section of concrete. Exhibit 8 (NSP000008) at 2-4; Exhibit 12 (NSP000012) at 13.

Q17. What has been the result of the inspections of the containment vessel?

A17. In 1998, we removed grout in Unit 2, conducted a visual inspection of the containment vessel, and commissioned the AES engineering evaluation of the effects of borated water. In 2002, we removed grout in Unit 1 and conducted the same visual inspection of the containment vessel. From 2003 to 2004, we conducted ultrasound measurements of wall thickness in each unit in the area around the transfer tube. In 2006, AES reviewed its previous engineering evaluation to assess the exposure of the containment vessel and structures from 1998 to 2006. From 2008 to 2009, we again removed grout in both units and performed visual and ultrasound examinations of the containment vessel, including the area around the RHR pump suction lines. In all instances, we determined wall thickness measurements to be at or above ASTM specifications, and we observed no corrosion or pitting of the containment vessel. Exhibit 8 (NSP000008) at 4-3; Exhibit 4 (NSP000004) at 6-8, 24; Exhibit 14 (NSP000014) at 3-142.

The 2009 DEI evaluation conducted a comprehensive study of the measured and potential effects of borated water leakage on the concrete, reinforcing bars, and containment vessel of both units at PINGP. It concluded that there is no evidence of the steel

containment vessel having sustained any significant corrosion to date. Although some inaccessible areas of the containment vessel are likely to have been wetted for long periods, potentially intermittently since plant startup, evaluation of the likely environmental conditions in those areas indicates that effects of such wetting has been insignificant (<0.010 inch thinning). Based on a conservative estimate of 0.25 inch for the maximum plausible thinning (which the NRC Staff found was not representative of actual conditions, based on the absence of observed degradation at PINGP), there is no concern for the ability of the steel containment vessel to perform its safety-related functions. Exhibit 8 (NSP000008) at 2-1, 4-2; Exhibit 14 (NSP000014) at 3-145.

Q18. What do the inspections show with respect to the status of the rebar?

A18. The 2009 DEI evaluation determined that there were no detected signs of reinforcing bar corrosion, and that any areas in which the rebar cover depth is the minimum allowed would have experienced only insignificant corrosion of 0.016 inch. Even if the lower region of the containment had been wetted by borated water for much of the plant's life, no corrosion of this rebar is expected to have occurred. Exhibit 8 (NSP000008) at 2-1 to 2-3.

During the 2009 Unit 1 outage, rebar was uncovered during excavation of grout in sump B to access the containment vessel for inspection. Sump B was chosen for inspection because it is located at a lower elevation and consistently shows wetting when refueling cavity leakage occurs. The visible rebar did not exhibit any degradation or corrosion. Rebar thickness remains almost identical to its original condition when it was installed during plant construction. Exhibit 4 (NSP000004) at 8.

Q19. What are the anticipated short term effects of the refueling cavity leakage on the PINGP containment integrity?

A19. Following additional observed leakage following the DEI evaluation completed in March 2009, NSPM conducted a review of the potential impact of continued refueling cavity leakage through the 27th cycle of unit operations. Extrapolating the results from the DEI evaluation showed that the maximum additional wall loss or concrete and rebar corrosion attributable to an additional 1.5 years of postulated borated water exposure would have an insignificant impact on the safety functions of the containment vessel and other structures. Thus, the follow-up study, EC 15044 (Exhibit 11 (NSP000011)), concluded that the DEI evaluation's conclusions remain valid. Exhibit 11 (NSP000011) at 2-3. While this result is valid only for the next plant operating cycle, if leakage is observed during the next or a subsequent refueling outage, another evaluation of the potential impact of continued leakage will be performed. Based upon extrapolation of the DEI evaluation, any continued leakage that occurs in the future should be determined, after evaluation, not to have a significant impact on plant safety.

Per the RCE's recommended actions, NSPM completed an evaluation of the minimum wall requirements of potentially corroded areas of the containment vessel and allowable concrete degradation, including the areas around the transfer tube. The evaluation, EC 15651 (Exhibit 12 (NSP000012)), found that, accounting for stresses, the containment vessel could tolerate approximately a 0.5 inch loss of the nominal 1.5 inch wall thickness without a significant challenge to integrity or functionality. The containment internal concrete structures and reinforcing steel were determined to have design margins of 30%, according to data from the updated safety analysis report (USAR),

thus making DEI's postulated degradation of 3% of the concrete area around the transfer tube insignificant to the integrity or functionality of that section. Exhibit 12 (NSP000012) at 10-13.

Q20. What further steps will NSPM take in the future to ensure resolution of the refueling cavity leakage issue?

A20. During the 2011 Unit 1 outage, NSPM will repair the RCC change fixture guide tube supports and will repair a liner to floor embedment plate fillet weld porosity indication. Exhibit 7 (NSP000007) at 4, 12. We have also entered a work order to install gaskets to repair leaks in the sandplug covers in both units. Exhibit 9 (NSP000009) at 2.

During the two consecutive refueling outages following cavity leak repairs in each unit, NSPM will perform visual inspections of the areas where reactor cavity leakage had been observed previously to confirm resolution of the issue (Commitment 42). Exhibit 13 (NSP000013) (ACRS Letter) at 3; Exhibit 14 (NSP000014) at A-10. Prior to the last refueling outages in each unit, NSPM had initiated corrective actions to confirm resolution of the leakage issue, expecting the pre-refueling repairs to eliminate leakage and enable closing of the corrective actions. Because the prior repairs did not fully eliminate leakage, however, the corrective actions will remain open through the end of the next refueling outage in each unit to confirm resolution of the leakage issue. Exhibit 10 (NSP000010) at 1; Exhibit 15 (NSP000015) (EFR 1160372-03) at 1. The Structures Monitoring Program and ASME Code Section XI, Subsection IWE Program will continue to monitor for any remaining leakage and evaluate the condition and integrity of containment vessel structures.

Additionally, NSPM will be conducting further inspections and tests to ensure no vessel degradation has occurred. As discussed

earlier, all prior inspections of the containment vessel wall, conducted in both units periodically from 1998 to 2010, revealed no evidence of corrosion and in all instances showed wall thickness to be at or above ASTM specifications. The containment vessels in both units are subject to ongoing inservice inspections and evaluations to ensure that they have not degraded and maintain their required thickness, in accordance with ASME Code Section XI, Subsection IWE and 10 CFR § 50.55a. In the unlikely event that these inspections present indications not acceptable by evaluation under ASME Code Section XI, the PINGP Subsection IWE program requires that such indications be repaired or replaced in order to be acceptable for continued service. NSPM has committed (Commitment 41) that, during the next refueling outage for each unit, following removal of concrete from sump C below the reactor vessel, a contractor will be performing a visual and ultrasonic evaluation of the containment vessel to determine the thickness and validate the prior results showing no degradation. (Sump C has been identified as one of the areas most likely to experience any potential corrosion due to the possibility of trapped water collecting there.) NSPM will also perform a petrographic examination of any removed concrete, including an evaluation of any water found at the location of removed concrete. Exhibit 14 (NSP000014) at A-10.

Further, NSPM has committed (Commitment 44) to removing, during the next outage for each unit, a concrete sample that has been wetted by borated water and testing it for compression strength as well as a petrographic examination. Id.

NSPM will complete all leakage-related commitments (Commitments 41, 42, and 44) prior to the renewal period. It will continue to manage aging in the containment structure and the

vessel using the Structures Monitoring Program, as well as the ASME Code Section XI, Subsection IWE Program. Any items of concern will be entered into the corrective action program for evaluation and correction. Id.

In short, we are committed to eliminating the refueling cavity leakage for both units, and we will continue to work as necessary to achieve that goal and assess whether it has been realized.

IV. NSPM CONSIDERATION OF SAFETY CULTURE

Q21. Did NSPM consider the safety culture implications, if any, of its actions in response to the refueling cavity leakage issue?

A21. Yes. The RCE that evaluated the issue from the technical standpoint also provided an assessment of the safety culture implications of the issue that included, among other findings, the observations that there is no clear evidence of a systematic decision process or management involvement in what steps should be taken to mitigate leakage; that the corrective action process had not been effective in addressing the issue; and that the fact that leakage has been an issue for over twenty years suggests a past lack of organizational accountability to take the actions needed to permanently resolve the issue.

Exhibit 4 (NSP000004) at 31-32.

Q22. Are these RCE findings representative of the current status of the safety culture at PINGS?

A22. No. The issues identified in the RCE stem from past insufficiencies in accountability at the organizational level that have since been remedied by NSPM's shift to a strict process-driven approach to handling identified problems. In the past, resolution was "championed at the individual level" because it

was thought that the person handling an issue was best suited to assuming responsibility for its resolution. Under the current Corrective Action Program ("CAP"), the manager or supervisor responsible for a corrective action is accountable for it regardless of who handles it personally. Corrective action documents are now classified by issue significance, such that the more significant the issue is, the greater the levels of management accountability and responsibility that apply. The formal requirements of PINGP's current CAP ensure proper organizational decision-making and oversight, preventing a single individual from undermining its effectiveness.

If the refueling cavity leakage issue had first presented itself today, PINGP management would document the problem, recognize that its continuance is unacceptable, and set about understanding its cause(s) and applicable solution(s). First, since this issue is equipment-related, one of the licensed operators would determine whether it was an issue of technical specification operability and would document an initial conclusion if there was sufficient information to do so, or he might request an engineering evaluation. Next, the action request would be presented to the CAP screening team, who would determine the issue significance, decide what actions are warranted, and assign it to a manager whose level corresponds to the issue's significance. This manager would then take responsibility for successful resolution of the issue. The CAP screening team would also determine which level of causal evaluation applies depending on the issue's significance. Finally, following completion of the causal evaluation, PINGP's Performance Assessment Review Board, composed of members of the senior management team, would review and grade the evaluation, ensuring proper resolution before the corrective action was closed.

Q23. Were there positive safety culture findings in the RCE?

A23. Yes. The RCE noted positive safety culture assessments in two areas relating to PINGP's safety conscious work environment. First, the numerous CAP documents over the years "indicate an unimpeded recognition of the [refueling cavity] issue at all levels of the organization" and appear to have objectively assessed the issue, including an evaluation of the potential for and consequences of any theoretical degradation. Second, the RCE found that NSPM management has encouraged personnel involved with refueling cavity leakage to report the issue and take corrective action. Exhibit 4 (NSP000004) at 31.

V. ISSUES RAISED IN SAFETY CULTURE CONTENTION

Q24. PIIC's contention asserts that, "after twenty years of leakage, the applicant still cannot identify the exact source of the leak." PIIC Contention at 6. Is this statement correct?

A24. No. As discussed previously, we currently have a very good understanding of the sources of the leak. We believe the sources of the lower cavity leakage into sump B and the regenerative heat exchanger room to be the floor embedment plates for the reactor vessel internals stands and the RCC assembly change fixture, and the wall embedment plates for the RCC assembly guide tube supports. Exhibit 4 (NSP000004) at 5. We believe the source of the upper cavity leakage into sump C and sump B to be the sandplug covers or those of the Nuclear Instrumentation System ("NIS"), which is a system that protects the reactor core by monitoring the neutron flux and generating the appropriate protective signals and alarms for various operating and shutdown conditions. Exhibit 9 (NSP000009) at 1-2. Because of the inaccessibility of the potential sources of the leakage and its

intermittent nature, pinpointing "the exact source" of the leak is difficult and, from the safety standpoint, unnecessary.

Q25. PIIC's contention also asserts that NSPM's efforts to fix the leak "have not been successful." PIIC Contention at 5-6. Do you agree with this assertion?

A25. No. PINGP's recent repairs have eliminated 95-97 percent of previous leakage in Unit 1 and 97.5 percent of previous leakage in Unit 2. We believe that our recent permanent repairs to the floor embedment plates in both units fully eliminated that leakage source. We have identified the RCC assembly guide tube wall embedment plates as a likely source of the minor remaining leakage. We are scheduled to repair these plates for Unit 1 during the next Unit 1 outage in 2011 and will identify any additional inspections and repairs prior to that time. Based on our identification of the sandplug covers as the likely source of the remaining upper cavity leakage in Unit 2, we have entered work requests to install gaskets for those covers in both units during the next refueling outages. The current CAP actions will remain open through the end of the next refueling outage in each unit to verify successful elimination of the remaining minor leakage. For the next two consecutive refueling outages in each unit following repairs, we will continue to monitor the areas previously exhibiting leakage to confirm no recurrence of the leakage.

Q26. In its contention, PIIC states that "[t]he potential hazard of this leakage is that the borated water appears to be settling at the bottom of the containment liner, posing a danger to the integrity of the containment." PIIC Contention at 6. Does the leakage pose such a danger?

A26. No. In the first place, it is inaccurate to refer to "the containment liner." PINGP does not have a containment liner, but a containment vessel wall of 1.5 inch thickness, as described above.

After repeated inspections and evaluations over many years, we have found no evidence of degradation in the containment vessels, the rebar, or the concrete. On multiple occasions, we conducted ultrasonic and visual examinations of the containment vessel through the sump B wall. We removed grout, measured the wall thickness to be at or above ASTM specifications, and observed no corrosion of the rebar or containment vessel. We also performed an ultrasonic examination of the containment vessel from the annulus for Unit 2 in 2008 and Unit 1 in 2009. This testing similarly showed wall thickness at or above ASTM specifications and noted no corrosion of the rebar or containment vessel. To be conservative, we also commissioned multiple engineering companies to perform independent evaluations of the potential for degradation of the steel containment vessel and reinforced concrete. The data from these evaluations, completed in 1998, 2006, and 2009, showed that potential exposure of the containment vessel and structures to refueling cavity water has not had an adverse impact on their ability to meet design requirements, and that any potential degradation would be so limited as to have no safety significance.

During the fall 2009 Unit 1 outage, we again removed grout from sump B in order to conduct visual and ultrasonic inspection of the containment vessel. This inspection showed no evidence of any degradation of the vessel wall, nor of the rebar exposed during excavation. Following repairs to the embedment plates in each unit, we will remove concrete from sump C below the reactor vessel to expose the containment vessel, inspect the containment vessel, assess the exposed concrete, and conduct a petrographic examination of the removed concrete. Thus, the past inspections and evaluations and those to be performed in the future, together with the repair programs we have conducted and will continue to

perform, provide assurance that the theoretical danger to the integrity of the containment will not materialize.

Q27. PIIC's contention alleges that "applicant's deficient performance and dereliction of its obligations to promptly and effectively correct deficient conditions call into question the applicant's ability to effectively implement the aging management program during the period of extended operation." PIIC Contention at 5. Do you agree with this assertion?

A27. No. NSPM's performance has not been deficient nor has there been dereliction of its obligations as licensee. NSPM has been proactive in pursuing multiple avenues to resolve the leakage issue and conducting a detailed root cause evaluation when previous measures proved less than completely effective. We repaired the components identified as the source of the leakage, and we will continue to do so if new leakage sources are identified. Although the inconsistent success of early remedial actions have indicated the difficulty of pinpointing the exact source of leakage, we will not be satisfied with our resolution efforts and will not close the CAP actions associated with this issue until the leakage is fully eliminated. As I stated previously, we have committed to conduct further visual inspections in subsequent refueling outages to ensure that the leakage issue has been fully resolved, and to perform further testing of the integrity of the containment vessel in both units.

Q28. PIIC asserts that NSPM "did not acknowledge the importance of [the refueling cavity leakage] to aging management until the NRC audit in the fall of 2008." Is this assertion correct?

A28. No. In addition to multiple work orders and other measures to address the issue, NSPM ordered an independent safety evaluation from AES in 1998 to assess the potential degradation effects of

borated water leakage on the containment structure. Even though this evaluation, the conclusions of which were confirmed in 2006, found that any leakage effects would have no safety significance, we conducted numerous tests and implemented a series of repairs to stop the leakage. These measures were taken starting in 1999 under both the Structures Monitoring Program and the ASME Section XI, Subsection IWE Program. At this point, based on the successful repairs already performed, as well as extensive future site inspection and repair plans to fully monitor and eliminate the refueling cavity leakage issue, the NRC has indicated its satisfaction with NSPM's approach and commitment to aging management by closing the previously open item in the Safety Evaluation Report ("SER").

Q29. Does the refueling cavity leakage issue negatively reflect on NSPM's ability to conduct an aging management program during the license renewal period?

A29. No. In the SER, the NRC staff concluded that NSPM's remedial measures and commitments demonstrate its ability to effectively implement the aging management program: "The applicant's commitment to inspect the containment vessel in an area susceptible to corrosion, along with the fact that PINGP has no current signs of containment degradation, provides assurance that the IWE and Structures Monitoring Programs will effectively manage aging of the containment vessel during the period of extended operation." Exhibit 14 (NSP000014) at 3-23 . Following its own investigation and analysis, the Advisory Committee on Reactor Safeguards agreed with the NRC staff's assessment. Exhibit 13 (NSP000013) at 3. We agree with those independent assessments.

Q30. Does that conclude your testimony?

A30. Yes.