UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

November 30, 2009

EA-09-259

Mr. Charles G. Pardee
Senior Vice President, Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO), Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNIT 1, NRC FOLLOW-UP INSPECTION REPORT 05000456/2009007; PRELIMINARY YELLOW FINDING

Dear Mr. Pardee:

On November 3, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Braidwood Station. This report documents the actions taken to review an Unresolved Item (URI) from the 2009 baseline inspections at Braidwood Station (URI 05000456/2009003-04; 05000457/2009003-04). The results were discussed on November 3, 2009, with Mr. A. Shahkarami and members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

The enclosed report presents the results of this inspection including a finding that has preliminarily been determined to be Yellow, a finding with substantial safety significance that may require additional NRC inspections. As described in Section 1R22 of this report, the finding involves a June 24, 2009, failure of the B Train Containment Sump Suction Valve, 1SI8811B, to stroke full open during surveillance testing. The inspectors determined that measures were not established to ensure the selection and suitability of application of equipment essential to the safety-related function of the residual heat removal system. Specifically, the design of the 1SI8811B motor operated valve actuator and associated conduit were not suitable to the application, because the design allowed water to enter and collect inside the actuator. This resulted in the failure of Valve 1SI8811B to stroke full open during surveillance testing on June 24, 2009, due to corrosion of the torque switch.

The finding is also an apparent violation of NRC requirements and is being considered for escalated enforcement action in accordance with the Enforcement Policy, which can be found on the NRC's Website at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html.

The finding was not an immediate safety concern because, at the time of discovery, it did not involve a complete loss of safety function of the emergency core cooling system recirculation function, the 1B Residual Heat Removal train was out of service for maintenance, and repairs were performed to restore valve 1SI8811B to an operable state.

In accordance with NRC Inspection Manual Chapter (IMC) 0609, we intend to complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of the date of this letter. The Significance Determination Process (SDP) encourages an open dialogue between the NRC staff and the licensee. However, the dialogue should not impact the timeliness of the staff's final determination.

Before we make a final decision on this matter, we are providing you with an opportunity to either: (1) attend a Regulatory Conference where you can present to the NRC your perspective on the facts and assumptions the NRC used to arrive at the finding and assess its significance; or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter, and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of your receipt of this letter. If you decline to request a Regulatory Conference or submit a written response, you relinquish your right to appeal the final SDP determination, in that by not doing either; you fail to meet the appeal requirements stated in the Prerequisite and Limitation Sections of Attachment 2 of IMC 0609.

Please contact Mr. Richard Skokowski at (630) 829-9620 and in writing within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence.

Because the NRC has not made a final determination in this matter, no Notice of Violation is being issued for this inspection finding at this time. In addition, please be advised that the number and characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html.

Sincerely,

/RA/

Steven West, Director
Division of Reactor Projects

Docket No. 50-456
License No. NPF-72

Enclosure: Inspection Report 05000456/2009007
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No.:	50-456
Licensee No.:	NPF-72
Report No.:	05000456/2009007
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Unit 1
Location:	Braceville, IL
Dates:	June 24, 2009, through November 3, 2009
Inspectors:	B. Dickson, Senior Resident Inspector A. Garmoe, Resident Inspector L. Kozak, Senior Reactor Analyst N. Valos, Senior Reactor Analyst
Approved by:	Richard A. Skokowski, Chief Branch 3 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000456/2009007; 6/24/09 - 11/3/09; Braidwood Station Unit 1; Surveillance Testing.

This report covers the follow-up inspection activities conducted by resident and regional inspectors to close an open Unresolved Item (URI). One preliminary yellow finding was identified by the inspectors. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). Cross-cutting aspects were determined using IMC 0305, “Operating Reactor Assessment Program.” The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

Preliminary Yellow. The inspectors identified a finding of substantial safety significance and an associated apparent violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the failure to prevent water from entering the motor operated valve actuator for valve 1SI8811B that resulted in corrosion of the torque switch. This resulted in the valve failing to stroke full open on June 24, 2009. The licensee determined that water entered the valve actuator through a flexible conduit penetration and pooled in the actuator limit switch box. This caused corrosion of the torque switch and minor corrosion of the limit switch. As part of the corrective actions for this event, the licensee sealed the susceptible conduit. Also, to address extent of condition, the licensee subsequently performed successful valve strokes of the 1SI8811A and 2SI8811A/B valves as part of previously scheduled maintenance windows. Additionally, the licensee performed a walkdown of the other SI8811 valves on both Units. Open conduit terminations were identified on all three remaining valves. The 2SI8811B valve was found to have the same susceptible conduit/cable tray configuration while the 1SI8811A and 2SI8811A valves had horizontal conduit terminations that were less susceptible to water intrusion. As a result, the licensee sealed the 2SI8811B valve open conduit termination.

The inspectors determined that the finding was more than minor due to impacting the Equipment Performance attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems the respond to initiating events to prevent undesirable consequences. The finding associated with this apparent violation was assessed using a Phase 3 analysis in accordance with NRC Inspection Manual Chapter 0609 Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” and is preliminary determined to have substantial significant safety significance (Yellow). The inspectors determined that this issue is associated with the Corrective Action Program component of the Problem Identification and Resolution cross-cutting area. (P.1(a)) Specifically, licensee staff was aware for several years of water leakage from the overhead areas around the SI8811 valves. Several corrective action documents were generated previously but the licensee did not adequately evaluate the potential safety significance of the water leakage and did not correct the issue. (Section 1R22.1.b)

B. B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstone: Mitigating Systems

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the licensee's apparent cause evaluation (ACE) associated with the June 24, 2009, failure of the Unit 1 B Train Containment Sump Suction Isolation Valve, 1SI8811B, to stroke full open during a routine surveillance test. This issue was initially identified as an URI in NRC Inspection Report 05000456/2009003; 05000457/2009003 as URI 05000456/2009003-04; 05000457/2009003-04. Document reviewed are list in the Attachment of this report.

b. Findings

(Closed) Unresolved Item 05000456/2009003-04; 05000457/2009003-04: Failure of Containment Sump Suction Valve 1SI8811B to Stroke Open

Introduction: The inspectors identified a finding of substantial safety significance and an associated apparent violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, for the failure to prevent water from entering the motor operated valve actuator for Valve 1SI8811B that resulted in corrosion of the torque switch. This resulted in the valve failing to stroke full open on June 24, 2009. The licensee determined that water entered the valve actuator through a flexible conduit penetration and pooled in the actuator limit switch box. This caused corrosion of the torque switch and minor corrosion of the limit switch.

Description: On June 22, 2009, the licensee began a planned work window on the 1B Residual Heat Removal (RH) train. The planned work included a routine valve stroke surveillance of Valve 1SI8811B. This valve is a normally closed motor operated valve that provides a containment isolation function. The valve is required to open to provide a suction source from the Emergency Core Cooling System (ECCS) sump to the 1B RH train for the recirculation phase of emergency core cooling. The valve is stroked in accordance with Procedure 1BwOSR 5.5.8.SI-7B, "Safety Injection System Containment Sump 1SI8811B Valve Stroke Surveillance." This procedure satisfies Technical Specification Surveillance Requirement 3.6.3.5, which requires that Valve 1SI8811B be stroke-time tested every refueling outage but not to exceed two years, per Section 3.1.2 of the Braidwood Second Interval Inservice Testing Plan, Revision 0, dated January 1998.

The valve was last stroked successfully on September 20, 2007. During performance of the surveillance on June 24, 2009, control room operators observed dual position indication, i.e., the valve was in a mid-position. Field reports indicated the valve stopped moving in the 30-40 percent open range. Further investigation, using information from a May 2006 diagnostic test of Valve 1SI8811B, initially concluded that the valve stroked

37.6 percent open. During this evaluation, the licensee calculated valve flow as a percent of the valve's full stroke length. The inspectors determined that the evaluation incorrectly concluded that the percent valve open (0-100 percent open) was equivalent to the valve stroke length (0-23.6 inches). Review of the licensee's maintenance procedures showed that the open stroke ends somewhere between 90 and 98 percent of valve full stroke length (~22.42 inches). Additionally, a review of the vendor's drawing for the valve identified that when the valve travelled from the "Closed" to "Open" position the valve must first travel 0.75 inches to clear the seating area (uncover the port). After the corrections were made by the licensee, they determined that the valve was 34.3 percent open, which could provide adequate NPSH to run one RH pump.

As part of initial troubleshooting, the licensee opened the actuator limit switch compartment cover. The cover was difficult to remove due to corrosion along the cover flange. The licensee identified that the torque switch open side contacts were open and the open torque switch bypass limit switch contacts were also open. This combination of open contacts resulted in the valve motor de-energizing, which caused the valve to stop moving. Heavy corrosion was observed on the torque switch contact arm assemblies, spring, and main shaft, which rendered the torque switch not functional.

The actuator limit switch compartment was wet with standing water at the bottom. Rust markings were observed leading from the control power conduit penetration into the actuator limit switch compartment. The conduit extends from the actuator limit switch compartment to Cable Tray 1619F in the overhead, where it terminates open-ended at roughly a 45 degree vertical angle. This orientation was such that if water were spilled on top of the cable tray, runoff would be routed toward the opening of the conduit, and would then drain into the actuator limit switch compartment. According to the licensee’s environmental qualification binder, EQ-BB-027, Limitorque Motor Operated Valve Actuators Outside Containment, conduit penetrations into the actuator limit switch compartment are not required to be absolutely sealed. Specifically, Section F, “Qualification Documents,” Paragraph 3.2.3, states:

> *Limitorque actuators for nuclear plant applications are designed to permit them to survive normal and accident conditions without depending on absolute sealing. In fact, the ambient is not absolutely restricted from entering the actuator. The seals are of no importance for qualification and, therefore, require no consideration for the qualification.*

During the review of the vendor information regarding the actuator for 1SI8811B, a Limitorque series SMB actuator, the inspectors noted that in Section 4.2 of the Flowserve-Limitorque Installation and Maintenance guide for SMB and SB series actuators states, “The following check points should be performed to maintain safe operation of the SMB or SB actuator:” Immediately following that statement is a list of the specific checkpoints. The third checkpoint listed in Section 4.2 states, “Keep the switch compartment clean and dry.”

The licensee’s apparent cause evaluation (ACE) determined that water spilled from the overhead onto Cable Tray 1619F and drained through the conduit into the actuator limit switch compartment. This resulted in corrosion of the torque switch and the valve stroke failure on June 24, 2009. The licensee sampled the water in the actuator limit switch compartment and determined it was not Reactor Coolant System (RCS) water.

The licensee looked into possible sources of clean water above Cable Tray 1619F and whether any spills had occurred in the area. However, the source of the water ultimately could not be determined.

To address extent of condition, the licensee performed a walkdown of the other SI8811 valves on both Units. Open conduit terminations were identified on all three remaining valves. The 2SI8811B valve was found to have the same angled conduit/cable tray configuration while Valves 1SI8811A and 2SI8811A had horizontal conduit terminations that were less susceptible to water intrusion. As part of the corrective actions for this event, the licensee sealed the susceptible conduits for leading to Valves 1SI8811B and 2SI8811B. Successful valve strokes of the 1SI8811A and 2SI8811A/B valves as part of previously scheduled maintenance windows have been completed.

The finding was not an immediate safety concern because, at the time of discovery, it did not involve a complete loss of safety function of the emergency core cooling system recirculation function, the 1B RH train was already out of service for maintenance, and repairs were performed to restore the 1SI8811B valve to an operable state.

<u>Analysis</u>: The inspectors determined that the failure to design the containment sump suction valve, 1SI8811B, and its associated electrical conduit in a manner that protected the safety-related function to open and provide the 1B ECCS train with a suction source from the containment sump for ECCS recirculation was a performance deficiency. The inspectors screened the performance deficiency in accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, “Issue Screening.” The inspectors determined the issue was more than minor due to impacting the Equipment Performance attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

The finding was screened using the Significance Determination Process (SDP) in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 – Initial Screening and Characterization of Findings,” Table 4a for the Mitigating Systems Cornerstone. The inspectors answered the first question, “Is the finding a design or qualification deficiency confirmed not to result in loss of operability or functionality?”, “No,” because the valve failed to stroke fully open during a required Technical Specification valve stroke surveillance test. The failure to stroke fully open also resulted in a required interlock not being made up such that two other valves would not open. The licensee declared Valve 1SI8811B inoperable and appropriate Technical Specification Limiting Conditions For Operation requirements were entered.

Based on the first question being answered “No,” the inspectors then answered the second question, “Does the finding represent a loss of safety function?” This question was answered “Yes” because the RH system serves as the low-pressure portion of the ECCS during the injection and recirculation phases of a Loss Of Cooling Accident (LOCA). After the injection phase, the system provides long-term recirculation capability for core cooling. The function is accomplished by aligning the RH system to take water from the containment sump, cooling this fluid by circulating it through the RH heat exchangers, and supplying it to the core via the RCS cold leg penetrations. If the pressure in the RCS is greater than the discharge pressure of the RH pumps, water may be returned to the core via the centrifugal charging pump and the safety injection pumps.

The inspectors determined that the last time Valve 1SI8811B was stroked successfully was on September 20, 2007. Since it was not possible to determine exactly when the corrosion started or became significant enough to prevent operation of the valve, the inspectors used a time of "T/2" or one-half of the time between the last successful time the surveillance was performed and the time the valve failed its surveillance test. The inspectors reviewed records for the other train of RH and determined that it had been taken out of service for testing or maintenance during the time when the 1SI8811B was likely to have been inoperable. Therefore this event was a condition that could have prevented the fulfillment of a safety function. As this question was answered "Yes," the inspectors were directed to perform a Phase 2 evaluation.

For the Phase 2 SDP evaluation, the Senior Reactor Analyst (SRA) used the Risk-Informed Inspection Notebook, Revision 2.1a, for Braidwood Nuclear Power Station to evaluate the risk significance of the finding. The SRA assumed that one train of high and low pressure recirculation was unavailable due to the failure of Valve 1SI8811B to fully open. Since the inception of the condition was not known, a time of "T/2" (or one-half of the time between the last successful time the surveillance was performed and the time the valve failed its surveillance test) plus the repair time was used to calculate the exposure period. The valve operation was last successfully demonstrated during surveillance testing on September 20, 2007. The valve failed on June 24, 2009, and was repaired, and returned to service on June 26, 2009. The exposure period was determined to be 322 days.

Recovery of the valve by local manual operation was considered but not credited. The valve is a large valve that is encapsulated. The motor operator and handwheel are not encapsulated and are accessible and there is procedural direction to manually open the valve if it fails to open from the control room. However, the time required to manually open the valve may exceed the time available and it was not clear if the environment in the auxiliary building after a LOCA event would allow operator access.

The result of the Phase 2 SDP was a Yellow finding. The dominant sequence was a small LOCA followed by failure of ECCS recirculation.

To evaluate whether the Phase 2 SDP evaluation was conservative, a Phase 3 SDP evaluation was performed. The SRA used the Standardized Plant Analysis Risk (SPAR) Model, Revision 3.51, for Braidwood Nuclear Power Station for the analysis.

For the Phase 3 SDP evaluation, the following SPAR Model modifications were made:

- Valve 1SI8811B was modeled as failure to open, and

- Valve 1SI8804B was modeled as failure to open. Valve 1SI8804B is the RH heat exchanger to SI pump isolation valve, and is required to be open to allow RH Pump "B" to supply suction to the SI and centrifugal charging pumps for high pressure ECCS recirculation. This valve is interlocked with 1SI8811B and will not open if 1SI8811B is not full open.

The exposure time used was 322 days (same as Phase 2 SDP evaluation). Although no actual common cause failure occurred, the potential for a common cause failure of the "A" RH train of containment sump recirculation existed. This was accounted for in the Phase 3 evaluation. The following influential assumptions were used:

- Local, manual recovery of the valve was not credited (same as Phase 2).

- Operation of the RH pump with the 1SI8811B valve partially open was not credited. Procedure guidance does not support operation of the system in this configuration. In fact, there is procedural direction to go to the Emergency Operating Procedure 1BwCA-1.1, “Loss of Emergency Coolant Recirculation.”

The total estimated change in core damage frequency was 3.2 E-5/yr (Yellow). The dominant sequence cut-set was a small break LOCA followed by common cause failure of the 1SI8811A and B valves, which failed all containment sump recirculation capability and led to core damage. The second dominant sequence was a medium break LOCA followed by the same common cause failure of the 1SI8811A and B valves.

The contributions to the risk estimates from external events (e.g., fire, flooding, and seismic) were determined to be low as discussed below.

For fires, IMC 0609, Appendix A, Attachment 3, was used to screen external event contributions. Fire risk contribution did not screen because the 1SI8811A and 1SI8811B valves are included in the licensee’s Appendix R Fire Safe Shutdown Analysis. However, the safe shutdown function of these valves is to remain closed and the concern from a fire perspective was spurious operation which results in draining the refueling water storage tank to the containment sump. Sump recirculation is not a credited Appendix R safe shutdown function for either inventory control or decay heat removal. The Braidwood Individual Plant Examination for External Event (IPEEE) concluded that the dominant fire risk scenarios involved transient, loss of offsite power, and inter-system LOCA initiators. Since small and medium break LOCA events dominate the risk of this finding, the SRA concluded that the fire risk contribution was not likely to change the overall significance of this issue.

Flooding scenarios were screened using Table 3.1 from IMC 0609, Appendix A. No risk significant flooding scenarios were identified for Braidwood.

For seismic scenarios, the 1SI8811A and B valves are on the seismic safe shutdown equipment list provided in the licensee’s IPEEE evaluation. The seismic contribution was qualitatively evaluated using guidance from the Risk Assessment of Operational Events Handbook, Volume 2 for External Events. The licensee performed a seismic margins analysis against a review level earthquake of 0.3g for the IPEEE evaluation. Using the Risk Assessment of Operational Events handbook, the frequency of earthquakes of a magnitude 0.3g or greater for Braidwood was estimated to be 1.59E-5/yr. The conditional probability of a small or medium break LOCA was estimated using Figure 4-9 from the Risk Assessment of Operational Events handbook. For small break LOCAs, this conditional probability was on the order of E-2 and for medium break LOCAs, on the order of E-4. The earthquake frequency combined with the conditional probability of a LOCA event resulted in a seismically-induced LOCA frequency that was much less than the LOCA frequencies used in the internal event analysis. As a result, the SRA concluded that the seismic risk contribution was also not likely to change the overall significance of this issue.

The change in large early release frequency was evaluated to be negligible. The insights from IMC 0609 Appendix H, “Containment Integrity Significance Determination Process,” were used to screen the potential risk contribution from large early release frequency for this finding. Braidwood Station is a pressurized water reactor with a large dry containment. Sequences important to large early release frequency include steam generator tube rupture events and interfacing-system LOCA events. These events were not the dominant core damage sequences associated with this finding.

In summary, the conclusion of the Phase 3 SDP was an estimated change in core damage frequency of 3.2 E-5/yr which represented a finding with substantial importance to safety significance (Yellow). The dominant core damage sequences were a small break LOCA followed by common cause failure of the 1SI8811A and B valves, which failed all containment sump recirculation capability, and a medium break LOCA followed by the same common cause failure of the 1SI8811A(B) valves.

The inspectors determined that this issue is associated with the Corrective Action Program component of the Problem Identification and Resolution cross-cutting area. Specifically, licensee staff was aware for several years of water leakage from the overhead areas around the SI8811 valves. Several corrective action documents were generated but they did not adequately evaluate the potential safety significance of the water leakage and did not correct the issue. (P.1(a))

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems, and components.

Section 4.2 of the Flowserve-Limitorque Installation and Maintenance guide for SMB and SB series actuators states, "The following check points should be performed to maintain safe operation of the SMB or SB actuator:" Immediately following that statement is a list of the specific checkpoints. The third checkpoint listed in Section 4.2 states "Keep the switch compartment clean and dry."

Apparently, from initial design, measures were not established to ensure the selection and suitability of application of equipment essential to the safety related function of the RH system. Specifically, the design of the motor operated valve actuator, a Limitorque SMB series actuator, for Valve 1SI8811B and associated conduit, did not appear to be suitable to the application, because the design allowed water to enter and collect inside the actuator. This resulted in the failure of Valve 1SI8811B to stroke full open during surveillance testing on June 24, 2009, due to corrosion of the torque switch. The licensee documented this condition in its corrective action program as IR 934782 and took corrective actions to seal the conduit for the 1SI8811B and 2SI8811B valves.

The performance deficiency did not meet the criteria for an old design issue. NRC IMC 0305, "Operating Reactor Assessment Program," Section 04.11, defines an "old design issue" as an inspection finding involving a past design-related problem in the engineering calculations or analyses, the associated operating procedure, or installation of plant equipment that does not reflect a performance deficiency associated with existing licensee programs, policy, or procedures. As discussed in Section 12.01 of IMC 0305, some old design issues may not be considered in the assessment program. Section 12.01(a) provides guidance for the treatment of old design issues, and states that the NRC may refrain from considering safety significant inspection findings in the assessment program for a design-related finding in the engineering calculations or analysis, associated operating procedure, or installation of plant equipment that meets all of the following criteria. The inspectors' evaluation is provided below:

1. It was licensee-identified as a result of a voluntary initiative such as a design basis reconstitution. For the purposes of IMC 0305, self-revealing issues are not considered to be licensee-identified. Self-revealing issues are those deficiencies which reveal themselves to either the NRC or licensee through a change in process, capability or functionality of equipment, or operations or programs.

   No. The issue was self-revealing though a failure during a Technical Specification required surveillance, which is not a voluntary initiative.

2. It was or will be corrected, including immediate corrective action and long term comprehensive corrective action to prevent recurrence, within a reasonable time following identification (this action should involve expanding the initiative, as necessary, to identify other failures caused by similar root causes). For the purpose of this criterion, identification is defined as the time from when the significance of the finding is first discussed between the NRC and the licensee. Accordingly, issues being cited by the NRC for inadequate or untimely corrective action are not eligible for treatment as an old design issue.

   Yes. The licensee has implemented corrective actions that will prevent recurrence on the 1SI8811B valve. However, the licensee did not consider this failure as a significant condition adverse to quality and, therefore, did not classify their corrective actions as corrective actions to prevent recurrence. Additionally, the licensee did not expand its corrective actions to look at other equipment with the same or similar flexible conduit.

3. It was not likely to be previously identified by recent ongoing licensee efforts such as normal surveillance, quality assurance activities, or evaluation of industry information.

   No. In 2002, there was a similar failure of a valve to stroke due to water in the actuator/limit switch compartment. In that event, water entered the 1CS001B valve through sealtite flexible conduit and caused the valve to only partially stroke. At that time, a corrective action was initiated to have system engineers inspect components that had liquidtite conduit to ensure connections were adequate (the licensee used the terms "sealtite" and "liquidtite" flexible conduit interchangeably in the 2002 corrective action document). This represented a missed opportunity to identify the conduit design on 1SI8811B. Additionally, in 2007, a containment sump suction valve at another U. S. nuclear facility failed to stroke full open due to corrosion of the torque switch. Review of this INPO Operating Experience represented another missed opportunity to identify the potential path of water intrusion into the actuator/limit switch compartment.

4. The issue does not reflect a current performance deficiency associated with existing licensee programs, policy, or procedure.

   Yes. The inspectors did not identify that the issue was representative of current licensee performance.

Since this design-related finding did not satisfy all of the above criteria, it is not considered to be an old design issue and is being treated similar to any other inspection finding, in accordance with IMC 0305, Section 12.01(a). This guidance is consistent with Section VII.B.3 of the NRC Enforcement Policy. (AV 05000456/2009007–01)

URI 05000456/2009003-04; 05000457/2009003-04 is closed.

4. OTHER ACTIVITIES

4OA6 Management Meetings

.1 Exit Meeting Summary

On November 3, 2009, the inspectors presented the inspection results to Mr. A. Shahkarami, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

A. Shahkarami, Site Vice President
L. Coyle, Plant Manager
K. Aleshire, Emergency Preparedness Manager
L. Antos, Security Operations Manager
K. Appel, Corporate Emergency Preparedness Manager
G. Bal, Engineering Program Manager
S. Butler, Emergency Preparedness Manager
G. Dudek, Site Training Manager
R. Gadbois, Maintenance Manager
D. Gullott, Regulatory Assurance Manager
J. Knight, Nuclear Oversight Manager
T. McCool, Operations Manager
J. Moser, Radiation Protection Manager
J. Odeen, Project Management Manager
T. Schuster, Chemistry Manager
M. Smith, Engineering Manager

Nuclear Regulatory Commission

R. Skokowski, Chief, Reactor Projects Branch 3

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000456/2009007-01	AV	Failure of Containment Sump Suction Valve 1SI8811B to Stroke Open

Closed

05000456/2009003-04; 05000457/2009003-04	URI	Failure of Containment Sump Suction Valve 1SI8811B to Stroke Open

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R22 Surveillance Testing

- IR 300989; Rain Water Leaking Into U2 CWA; February 13, 2005
- IR 355188; Water Leakage From Unit 2 CWA Ceiling; July 20, 2005
- IR 518815; Water Leaking in 2B RHR Pump Room (Onto Pump Pedestal); August 10, 2006
- IR 523419; Possible Unmonitored Vent Path and Water Leakage U2 CWA; August 24, 2006
- IR 592949; Ground Water Leaking By 2SI8811B; February 18, 2007
- IR 597508; 2B RHR P.R. Material Condition, Spread Of Contamination Issues; February 28, 2007
- IR 651704; Water Leaking From U2 364 CWA Roof; July 19, 2007
- IR 659784; Aux Building Roof Leakage; August 9, 2007
- IR 709894; Rain Water Causes Slip Hazard and Potential Contamination; December 11, 2007
- IR 729692; Ground Water Intrusion Into U1 CWA 364; January 29, 2008
- IR 777749; In-Leakage (Ground Water) Continues Into 2B RHR Pump Room; June 8, 2007
- IR 808063; INPO Identified AFI Related to Groundwater In-Leakage (ER.3-1); August 17, 2008
- IR 934782; 1SI8811B Failed to Stroke Full Open During Surveillance; June 24, 2009
- IR 957685; Seal Open End of Large Conduit (C1A1454) for 1SI8811B; August 26, 2009
- IR 957692; Seal Open End of Conduits C2A1421/87 for 1SI8811B; August 26, 2009
- IR 986541; Water Dripping on the 1SI8811B Valve Stem; October 30, 2009
- IR 986738; Rain Water Leaking Into Aux Building; October 30, 2009
- IR 986803; NRC/IEMA Concern With Rain Intrusion to U1 CWA; October 30, 2009
- EC 358828 Unit 1 and EC 367264 Unit 2, BRW-06-0035-M; NPSHA for RHR & CS Pumps During Post-LOCA Recirculation; Revision 1
- EC 370748; BRW-06-001-M, SI/RHR/CS/CV System Hydraulic Analysis in Support of GSI-191; Revision 3
- EC 372731; SITH-1, Refueling Water Storage Tank (RWST) Level Setpoints; Revision 007A
- WO 0695341 01; 1SI8811B Motor Operated Valve Diagnostic Test; November 17, 2005
- WO 0804741 01; 1SI8811B Age-Related Degradation Inspection; November 3, 2005
- WO 0695341 01; 1SI8811B Motor Operated Valve Diagnostic Test; November 17, 2005
- WO 1245941 01; 1SI8811B Failed to Stroke Full Open During Surveillance; June 25, 2009
- 1BwCA-1.1; Loss of Emergency Coolant Recirculation Unit 1; Revision 202, WOG 2
- 1BwEP-0; Reactor Trip or Safety Injection Unit 1; Revision 202, WOG 2
- 1BwEP-1; Loss of Reactor or Secondary Coolant; Revision 203, WOG 2
- 1BwOSR 5.5.8.SI-7B; Safety Injection System Containment Sump 1SI8811B Valve Stroke Surveillance; June 26, 2009
- ANSI/IEEE Std 344-1975; IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations; Revision of IEEE Std. 344-1971
- EPRI Report Summary; Technical Repair Guidelines for Limitorque Model SMB-00 Valve Actuators
- NSWP-E-01; Electrical Cable Installation and Inspection; Revision 4, March 12, 1996

LIST OF ACRONYMS USED

ACE	Apparent Cause Evaluation
ADAMS	Agencywide Document Access Management System
AV	Apparent Violation
CFR	Code of Federal Regulation
ECCS	Emergency Core Cooling system
IMC	Inspection Manual Chapter
IPEEE	Individual Plant Examination for External Event
LOCA	Loss of Cooling Accident
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records
RCS	Reactor Coolant System
RH	Residual Heat Removal
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk
SRA	Senior Reactor Analyst
URI	Unresolved Item

In accordance with NRC Inspection Manual Chapter (IMC) 0609, we intend to complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of the date of this letter. The Significance Determination Process (SDP) encourages an open dialogue between the NRC staff and the licensee. However, the dialogue should not impact the timeliness of the staff's final determination.

Before we make a final decision on this matter, we are providing you with an opportunity to either: (1) attend a Regulatory Conference where you can present to the NRC your perspective on the facts and assumptions the NRC used to arrive at the finding and assess its significance; or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter, and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of your receipt of this letter. If you decline to request a Regulatory Conference or submit a written response, you relinquish your right to appeal the final SDP determination, in that by not doing either; you fail to meet the appeal requirements stated in the Prerequisite and Limitation Sections of Attachment 2 of IMC 0609.

Please contact Mr. Richard Skokowski at (630) 829-9620 and in writing within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence.

Because the NRC has not made a final determination in this matter, no Notice of Violation is being issued for this inspection finding at this time. In addition, please be advised that the number and characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html.

Sincerely,
/RA/
Steven West, Director
Division of Reactor Projects

Docket No. 50-456
License No. NPF-72

Enclosure: Inspection Report 05000456/2009007
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

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NAME	RSkokowski:dtp/cms		SOrth		SWest			
DATE	11/20/09		11/24/09		11/30/09			

OFFICIAL RECORD COPY

Letter to C. Pardee from S. West dated November 30, 2009.

SUBJECT: BRAIDWOOD STATION, UNIT 1, NRC FOLLOW-UP INSPECTION REPORT 05000456/2009007; PRELIMINARY YELLOW FINDING

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