November 12, 2008

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

SUBJECT: BYRON STATION – IDENTIFICATION AND RESOLUTION OF PROBLEMS (ANNUAL SAMPLE) NRC BASELINE INSPECTION REPORT 05000454/2008009(DRS); 05000455/2008009(DRS)

Dear Mr. Pardee:

On October 10, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed a baseline annual inspection sample of Identification and Resolution of Problems at the Byron Station. The enclosed report documents the results of the inspection, which were discussed with Mr. Hoots, and others of your staff at the completion of the inspection on October 10, 2008.

The inspectors 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. Based on the results of the inspection, two NRC identified findings of very low safety significance, which involved violations of NRC requirements were identified. However, because these violations were of very low safety significance and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of an NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission – Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Byron Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the

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Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos. 50-454; 50-455
License Nos. NPF-37; NPF-66

Enclosure: Inspection Report No. 05000454/2008009(DRS); 05000455/2008009(DRS)
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Byron Station
Plant Manager - Byron Station
Regulatory Assurance Manager - Byron Station
Chief Operating Officer and Senior Vice President
Senior Vice President - Midwest Operations
Senior Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Director - Licensing and Regulatory Affairs
Manager Licensing - Braidwood, Byron, and LaSalle
Associate General Counsel
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
J. Klinger, State Liaison Officer,
Illinois Emergency Management Agency
P. Schmidt, State Liaison Officer, State of Wisconsin
Chairman, Illinois Commerce Commission
B. Quigley, Byron Station

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Plant Manager - Byron Station
Regulatory Assurance Manager - Byron Station
Chief Operating Officer and Senior Vice President
Senior Vice President - Midwest Operations
Senior Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Director - Licensing and Regulatory Affairs
Manager Licensing - Braidwood, Byron, and LaSalle
Associate General Counsel
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
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Illinois Emergency Management Agency
P. Schmidt, State Liaison Officer, State of Wisconsin
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B. Quigley, Byron Station
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Letter to Mr. Charles G. Pardee from Mr. David E. Hills dated November 12, 2008.

SUBJECT: BYRON STATION - IDENTIFICATION AND RESOLUTION OF PROBLEMS (ANNUAL SAMPLE) NRC BASELINE INSPECTION REPORT 05000454/2008009(DRS); 05000455/2008009(DRS)

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-454; 50-455
License Nos:	NPF-37; NPF-66
Report Nos:	05000454/2008009(DRS); 05000455/2008009(DRS)
Licensee:	Exelon Generation Company, LLC
Facility:	Byron Station, Units 1 and 2
Location:	Byron, IL
Dates:	June 20, 2007, October 4, 2007, and October 6 through October 10, 2008
Inspectors:	M. Holmberg, Reactor Inspector (Lead) G. O'Dwyer, Reactor Inspector M. Mitchell, Health Physicist J. Neurauter, Reactor Inspector
Approved by:	D. E. Hills, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000454/2008009(DRS); 05000455/2008009(DRS); 10/06/2008 through 10/10/2008; Byron Station, Units 1 and 2; Identification and Resolution of Problems.

The inspection covered a one-week announced baseline inspection on identification and resolution of problems. The inspection was conducted by four region based inspectors. Two Non-Cited Violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red), using Inspection Manual Chapter 0609, "Significance Determination Process (SDP)." Findings for which the SDP does not apply, may be Green, or be assigned a severity level after NRC management review. 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. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

Severity Level IV. The inspectors identified a Severity Level IV Non-Cited Violation (NCV), having very low safety significance of 10 CFR 50.71, "Maintenance of Records, Making of Reports," for the licensee's failure to adequately update the Byron Station Updated Final Safety Analysis Report. Specifically, the description of: (1) the boron recycle system did not identify if the system was designed or capable of handling discharges from the safety injection and residual heat removal relief valves; (2) the residual heat removal system did not identify deviations from the system design standard with respect to the suction pipe relief valve single failure analysis and collection of relief valve discharges outside containment. The licensee entered this issue into the corrective action system.

Because this finding affected the NRC's ability to perform its regulatory function, this issue was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors could not reasonably determine that a change to correct the Final Safety Analysis Report to reflect actual design would not have ultimately required NRC prior approval. The finding was determined to be of very low safety significance because the design deviations associated with the residual heat removal system and boron recycle system did not impact system operability. The inspectors determined that the finding did not have a cross-cutting aspect. (Section 4OA2.b.1)

Cornerstone: Barrier Integrity

Green. The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance, associated with the licensee's failure to analyze and establish an adequate quench volume within the boron recycle system holdup tanks and failure to analyze the water hammer loads on boron recycle system holdup tank inlet piping induced by relief valve discharges. Insufficient holdup tank quench volume could result in an overpressure failure of the holdup tank and the water hammer induced piping loads could damage the boron recycle system holdup tank inlet piping system. The licensee corrective actions included maintaining a minimum 40 percent boron recycle holdup tank level as a quench volume for system relief valves

and initiated an action to perform an analysis to investigate the magnitude of the potential water hammer loads on the inlet piping.

The finding was more than minor because, the finding affects the Barrier Integrity Cornerstone objective for maintaining the Radiological Barrier Function of the Containment. The finding was associated with the design control and procedure quality attributes of the Barrier Integrity Cornerstone. The inspectors determined that the failure to establish an adequate boron recycle system holdup tank quench volume and analyze the magnitude of water hammer loads on boron recycle system holdup tank inlet piping degraded the Radiological Barrier Function of the Containment. Because this issue did not represent an actual open pathway from containment, the finding screened as having very low safety significance (Green). The inspectors determined that the finding did not have a cross-cutting aspect. (Section 4OA2.b.2)

B. **Licensee-Identified Violations**

None

REPORT DETAILS

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

.1 Annual Sample Review

a. Inspection Scope

On April 21, 2001, and May 1, 2007, the Byron licensee had experienced lifting of the residual heat removal (RHR) system relief valves due to back-leakage through isolation valves with the system isolated. The inspectors noted that Byron Station had a similar configuration as the Braidwood Station, which had experienced the unplanned inter-system discharge of approximately 68,000 gallons of reactor coolant (reference Information Notice 90-05) caused by the inadvertent opening of the RHR system suction relief valve.

On June 20, 2007 and October 4, 2007, the NRC identified a number of concerns associated with the design and operation of the Byron Station boron recycle system holdup tanks (HUTs). From October 6, 2008 through October 10, 2008, the inspectors completed one inspection sample regarding problem identification and resolution related to review of the following Assignment Reports (ARs) associated with the RHR system and HUT.

- AR 622574, Concerns regarding relief valves and inputs into the HUT;
- AR 649710, Potential Vulnerability with RHR suction relief valve to HUT;
- AR 679954, Recycle Holdup Tank Level Administrative Controls; and
- AR 680626, NRC Potential Green Finding and Associated Non-Cited Violation (NCV) - HUT Level.

The inspectors performed a walkdown of portions of the boron recycle system and reviewed the licensee's actions for the issues identified above, to verify whether: (1) the problems were accurately identified; (2) the causes were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) previous occurrences were considered; and (5) corrective actions proposed/implemented were appropriately focused to address the problems and were commensurate with the safety significance of the issues.

b. Findings

1. Failure to Update the Boron Recycle and RHR system Descriptions in the Updated Final Safety Analysis Report

Introduction: The inspectors identified a Severity Level IV NCV of 10 CFR 50.71, “Maintenance of Records, Making of Reports,” for the licensee’s failure to adequately update the Byron Station Updated Final Safety Analysis Report (UFSAR). Specifically, the licensee had not updated the UFSAR: (1) description of the boron recycle system

(UFSAR Section 9.3.4) to identify that it was designed or capable of handling discharges from the safety injection (SI) system and RHR relief valves; (2) description of the RHR system (UFSAR Section 5.4.7.2.7) to identify deviations from the RHR system design standard with respect to the suction pipe relief valve single failure analysis and collection of relief valve discharges outside containment.

Description: On October 6, 2008, the inspectors identified a concern for failure to update the UFSAR Sections associated with the boron recycle and RHR system design and operation.

The UFSAR Section 9.3.4.2.1, identified the design bases for the boron recycle system including the collection requirements from various systems. For example, Section 9.3.4.2.1.1.b, identified that the chemical and volume control system (CVCS) charging pump suction pressure relief valve was a source to be collected by this system. However, this Section did not identify that this system was designed or capable of handling discharges from the SI and RHR relief valves, which were directed into the HUTs. Further, the licensee had not initiated actions to update this UFSAR Section to reflect that a quench volume was needed to mitigate the effects of hot discharges from the RHR suction relief valves as discussed in Report Section 4OA2.1.b.2 below.

For the RHR System, the Byron Station was committed to meet the NRC's Branch Technical Position (BTP) RSB 5-1 "Design Requirements of the Residual Heat Removal System." Specifically, the UFSAR Section 5.4.7.2.7 stated that "Byron/Braidwood is subject to the technical requirements of RSB 5-1 as they apply to Class 2 Plants." It also stated, "The functional requirements of RSB 5-1 impose the following assumptions on the system(s) used to go to cold shutdown: (1) a loss of offsite power, the most limiting single failure; and (2) that only safety grade systems are available." The specific single failures considered for RHR system were identified in UFSAR Table 5.4-18 "Single Failure Evaluation of Systems Required to Reach Cold Shutdown per BTP RSB 5-1." However, this table did not identify the impact of a single failure of an RHR suction relief valve (e.g., stuck open or stuck closed). The Technical Specification Bases Section 3.4.12 stated "RHR suction relief valves are considered active components. Thus, the failure of one valve is assumed to represent the worst case single active failure." Because the UFSAR lacked a description of the "worst case" single active failure associated with a failed RHR suction relief valve, it deviated from the RHR design standard BTP RSB 5-1.

Section C.2, of BTP RSB 5-1 required that "Fluid discharged through the RHR system pressure relief valves must be collected and contained such that a stuck open relief valve will not: (c) Result in a non-isolatable situation in which the water provided to the RCS to maintain the core in a safe condition is discharged outside of containment." In Section 5.4.3.3 "RHR Pressure Relief Requirements" of NUREG-0876 "Safety Evaluation Report Related to the Operation of Byron Station Unit 1 and 2," the NRC had documented that "Discharge from the RHR suction line relief valves is directed to the pressurizer relief tank." However, the discharge from the RHR system relief valves was routed to the HUTs, which are located in the Auxiliary Building, outside containment. Therefore, the licensee's RHR relief valve discharge piping configuration, which resulted in collection of fluid from a stuck open relief valve outside containment, deviated from BTP RSB 5-1, and NUREG-0876, and this deviation was not identified in the Byron UFSAR.

Analysis: The inspectors determined that the licensee's failure to update the UFSAR Sections associated with the boron recycle and RHR system design and operation was a performance deficiency. The finding was determined to be more than minor because the inspectors could not reasonably determine that a change to correct the UFSAR to reflect actual design would not have ultimately required NRC prior approval.

Because violations of 10 CFR 50.71(e) are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the significance determination process (SDP). However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. The inspectors completed a significance determination of the underlying technical issue using Inspection Manual Chapter (IMC) 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." Under the Mitigating Systems Cornerstone Column of Table 4a, the inspectors answered "Yes," that this was a design or qualification deficiency, confirmed not to result in loss of operability or functionality, because the design deviations associated with the residual heat removal system and boron recycle system did not impact system operability. Therefore, the finding screened as having very low safety significance, and in accordance with the Enforcement Policy, the violation was classified as a Severity Level IV violation. Because the performance errors, which led to this finding occurred prior to the approval of the plant operating license, this issue did not reflect current plant performance and therefore, no cross-cutting aspect was identified.

Enforcement: Title 10 CFR 50.71(e) requires, in part, that licensees periodically update the UFSAR originally submitted as part of the application for the operating license to assure that the information included in the UFSAR contains the latest information developed. Title 10 CFR 50.71(e) requires that the submittal contain all changes made in the facility and all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirements since the submission of the original UFSAR or, as appropriate, the last updated UFSAR.

Contrary to the above, as of October 10, 2008, the licensee failed to update the UFSAR: (1) description of the boron recycle system (UFSAR Section 9.3.4) to identify that it was designed or capable of handling discharges from the SI and RHR relief valves, which are directed into this system; and (2) description of the RHR system (UFSAR Section 5.4.7.2.7) to identify deviations from the RHR system design standard with respect to the suction pipe relief valve single failure analysis and collection of relief valve discharges outside containment. Because this was a Severity Level IV violation that was not willful and was entered into the licensee's corrective action program (AR 00828770), this violation was treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000454/2008009-01; NCV 05000455/2008009-01).

2. Failure to Analyze Inlet Piping Loads and Establish an Adequate HUT Quench Volume

Introduction: The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance (Green), associated with the licensee's failure to analyze and establish an adequate quench volume within the HUT, and failure to analyze the water hammer loads on HUT inlet piping induced by relief valve discharges. Insufficient HUT quench volume could result in an overpressure failure of the HUT, and the water hammer induced piping loads could damage the HUT inlet piping system.

Description: On June 20, 2007, the NRC identified a concern for available quench volume in the HUT, and lack of an analysis for water hammer loads on HUT inlet piping to accommodate discharges from the RHR suction relief valves. The licensee's past practice of maintaining the HUT water level below the inlet piping entrance when the RHR, SI, or CVCS relief valves were lined up to discharge to the HUT, provided no quench volume for the steam, and hot water discharges from these relief valves to avoid pressure buildup within the HUT.

The RHR suction and discharge relief valves were originally planned to discharge to the pressurizer relief tank (PRT) through a sparger pipe below the normal tank water level to ensure adequate quenching of steam for pressure suppression. This design configuration ensured that the PRT did not operate above the tank design pressure of 100 pounds per square inch gauge (psig). Further, the PRT had a cooling system to reduce PRT water temperatures after a discharge to ensure the design temperature of 200 degrees Fahrenheit (ºF) is not exceeded. During plant construction, the piping was rerouted so that the RHR system suction relief valves discharged to the HUT instead of the PRT. In contrast with the PRT design, the HUT design and operating parameters did not ensure that an RHR relief valve discharge (up to 360 psig and 350 ºF) would not exceed the design pressure and temperature of the HUT (15 psig and 200 ºF). Specifically, the relief valve discharge piping enters the HUT with no sparger at about 35 percent tank level; and the minimum tank level allowed by procedures was 5 percent. Further, the HUTs did not have a means to cool the tank after a relief discharge. With this configuration, the inspectors were concerned that the licensee had not established an adequate cold water volume for quenching hot relief valve discharges into the HUT, to ensure that the tank design pressure and temperature was not exceeded.

To address this concern, the licensee established a 40 percent level in the HUT for a quench volume, and incorporated this requirement to maintain this quench volume in Procedure BOP RH-6, “Operation of the RHR system in Shutdown Cooling.” The RHR suction relief valves provided the highest capacity/volume of hot water discharge to the HUTs and therefore, the licensee believed these valves were the limiting component for evaluation of this concern. To assess the HUT response (with and without a quench volume), the licensee had a vendor complete calculation, CN-CRA-07-50 "Byron/Braidwood RHUT [Recycle System Holdup Tank] Response to Opening of the RHR Relief Valve." In the first case, with an initial tank level of 5-feet below the inlet nozzle, the calculation results demonstrated that the HUT would have been overpressurized and potentially failed after 50 seconds of steam discharge. The HUT failure would release the contents of the HUT to the HUT room in the Auxiliary Building.

In the second case, with an initial HUT level was 5-feet over the inlet nozzle, the water volume was sufficient to ensure that the HUT air space pressure would not exceed the HUT design pressure even if the RHR suction relief valve was open for 30 minutes. The inspectors noted that this calculation did not attempt to determine the minimum acceptable tank level to ensure adequate pressure suppression occurred (e.g., quench volume).

On February 14, 2008, a licensee-contracted vendor completed calculation, CN-CRA-08-9 "Byron/Braidwood RHUT Response to Opening of the RHR Relief Valve," to provide a quantitative analysis for the licensee's previous engineering judgment that a 40 percent tank level provided sufficient quenching of hot discharges to prevent exceeding HUT design pressure. In this calculation, the vendor concluded that a 40 percent tank level would be sufficient to quench an RHR relief valve discharge such that the tank design pressure would not be exceeded. The licensee completed an owner's acceptance review of this vendor calculation on September 28, 2008. The licensee had entered the concerns for inadequate quench volume into the corrective action program (AR 00680626 and AR 00622574) and initiated additional actions to complete a more detailed design basis calculation for the minimum HUT quench volume.

The inspectors performed a walkdown of the Unit 1 HUT inlet pipe on October 7, 2008, and confirmed that the licensee had installed equipment status tags on inlet pipe isolation valves (0AB8557A(B)). These tags identified the need to maintain at least one HUT aligned for receiving discharges from relief valves to maintain an overpressure protection path and that a 40 percent or greater level was required for the inservice HUT. The inspectors noted that the discharge pipe routing from the RHR suction side relief valves to the HUTs contained a number of loop seals created as the elevation of the piping changes. These loop seals would allow for fluid to collect in the low points. If a relief valve lifted, the dynamic loads would be created as the noncondensable gases in the high points compress and the fluid columns accelerate. Although, the noncondensable gases would help cushion the fluid impact on downstream fluid columns, the discharge piping was not analyzed for these transient dynamic "water hammer" loads. Therefore, the magnitude of these transient loads was not known.

The licensee did not believe that an RHR relief valve discharge would likely create a violent transient (e.g., steam bubble collapse) in the discharge piping, and if any transient "water hammer" loads did occur they would be relatively low. The licensee identified that the existing piping stress analysis from the RHR suction relief valves to the HUT considered the reaction loads associated with actuation of the relief valves. In addition, the licensee stated that the potential for condensation-induced water hammer was low due to: (1) the approximate 30-feet elevation change in the piping between the tank and the RHR relief valves, which would limit the driving pressure available to move a slug backwards in the line; and (2) an extended blowdown into the tank would heat the water in the vicinity of the nozzle thus reducing the condensation rate and limiting the reduction in void pressure, thereby, limiting the available driving pressure. Furthermore, the licensee stated that Braidwood Unit 1 (similar configuration to Byron) had experienced a lift of the RHR suction relief valve and had not experienced damage to pipe and pipe supports. Based on this operating experience, the licensee postulated that resulting transient loads on piping and pipe supports would have a similar magnitude at Byron, and concluded that any damage to piping and pipe supports would be minimal.

The licensee reported that the HUT inlet pipe supports were analyzed to include seismic loads which provided margin to accommodate dynamic water hammer loads in the direction of the support restraint. However, the postulated dynamic water hammer loads could create piping forces in directions that were not restrained by the existing pipe supports. The licensee entered the concern for relief valve induced water hammer loads on the HUT inlet pipe into the corrective action program (AR 00680626 and AR 00622574). The licensee initiated an action to perform an analysis to investigate the magnitude of the potential water hammer loads on the inlet piping. During the walkdown of the Unit 1 HUT inlet pipe from the HUT to the RHR suction relief valves, the inspectors did not identify evidence of damaged piping or movement at the piping supports.

Analysis: The inspectors determined that the failure to evaluate and establish an adequate HUT quench volume and analyze the magnitude of water hammer loads on HUT inlet piping to accommodate discharges from the RHR suction relief was a performance deficiency warranting a significance evaluation. The inspectors determined the finding was more than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” because the finding affects the Barrier Integrity Cornerstone objective for maintaining the Radiological Barrier Function of the Containment. The finding was associated with the design control and procedure quality attributes of the Barrier Integrity Cornerstone. Specifically, the licensee's past practice of maintaining the HUT water level below the inlet piping entrance when the RHR relief valves were lined up to discharge to the HUT provided no quench volume for the steam and hot water discharges from these relief valves to avoid pressure buildup within the HUT.

The inspectors evaluated the finding using IMC 0609, Attachment 0609.04, “Phase 1 – Initial Screening and Characterizations of Findings.” The inspectors determined in Tables 2 and 4a of the attachment that the failure to establish a minimum level for adequate quenching in the HUTs degraded the Radiological Barrier Function of the Containment. Because this issue did not represent an actual open pathway from containment, the finding screened as having very low safety significance (Green). Because the performance errors, which led to this finding occurred prior to the approval of the plant operating license, this issue did not reflect current plant performance and therefore, no cross-cutting aspect was identified.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that design control measures shall provide for verifying the adequacy of design and that the design basis is correctly translated into procedures and instructions.

Contrary to the above, from plant construction to September 28, 2008, the licensee failed to verify the adequacy of the HUT design. Specifically, the licensee failed: (1) to evaluate and maintain the required water volume necessary to quench the RHR system relief valve discharges into the HUT and incorporate appropriate minimum HUT level requirements into the HUT level control procedures; and (2) to evaluate the effect of dynamic "water hammer" loads on inlet from relief valve discharges to the HUT. However, because this issue was of very low safety significance, and was entered into the licensee’s corrective action program (AR 00680626 and AR 00622574), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000454/2008009-02; NCV 05000455/2008009-02).

3. <u>Radiological Release Analysis Did Not Include Normal HUT Configurations</u>

On June 20, 2007, the NRC identified a concern that the UFSAR analysis for rupture of a HUT failed to recognize that the gas spaces of the HUTs were normally cross-connected and that a gas decay tank normally had open communication with at least one HUT.

Section 15.7.2 of the UFSAR included the results of an analysis for the worst case radioactive atmospheric release from the HUT and assumed that the postulated event was initiated by cracks in the HUT and operator error. The analysis assumed that only one HUT would fail, and a failure of both HUTs is beyond what was analyzed. The inspectors noted that the existing UFSAR analysis for the rupture of a recycle holdup tank failed to recognize that the gas spaces of the HUTs were normally cross-connected and that a gas decay tank normally had open communication with at least one HUT. The licensee stated that based on conservative assumptions in the analysis, the actual plant configuration (the gas decay tank providing cover gas to two HUTs) was bounded. Specifically, the calculated dose for a postulated recycle HUT failure was based on the following:

- The assumed inventory of noble gases in the HUT was based on transferring the total inventory of primary coolant from one unit at maximum purification letdown flow;
- No removal of noble gas was assumed in the purification letdown flow; and
- When the tank failure occurred, a portion of the iodine in the water and all of the noble gas activity was assumed to become airborne and released to the environment.

The inspectors reviewed calculation CN-CRA-00-47, "Braidwood/Byron Doses from Recycle Holdup Tanks and Spent Resin Tank Failures," and noted that calculation CN - CRA-00-47 assumed the HUT was initially filled to 80 percent capacity, the water contents of the tank would be released in 5-minutes, and the HUT was isolated from the other HUT and Gas Decay tank. These assumptions were consistent with those in UFSAR Section 15.7.2, which stated that the postulated events that could cause the worst case radionuclide inventory were cracks in the HUT and operator error. The inspectors identified that the calculation of record, CN-CRA-00-47, did not account for the actual plant conditions or the failure mechanism described in Section 4OA2.1.b.2 of this report. Specifically, the initial level of 80 percent was not consistent with current operation because HUT level could be as high as 91 percent or as low as 5 percent. A crack in the HUT and subsequent 5-minute release of water content was not consistent with the expected rupture of the HUT with HUT water level below the relief valve discharge line.

Although the calculation contained conservative assumptions with respect to gaseous releases, it did not specifically address how these assumptions bound the actual plant configuration and/or operation of the waste gas systems. The inspectors reviewed the calculation and determined these discrepancies (i.e., quicker release rate, lower initial volume, and cross-connected gas systems) would have an impact on the calculated release rates; however, the margin contained in the calculation was sufficient.

The inspectors concluded that the failure to account for actual plant configurations in an accident analysis was a violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” The inspectors assessed this violation in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” and determined that the finding was minor because the available margin and other conservative assumptions were sufficient to compensate for identified discrepancies. The licensee entered this issue into the corrective action system (AR 00649710 and AR 00680630). The licensee planned to revise the HUT rupture analysis to correct inputs and assumptions and update the UFSAR to reflect the revised analysis. Therefore, in accordance with IMC 0612, this violation of minor significance not subject to enforcement action.

4OA6 Meeting(s)

.1 Exit Meeting

The inspectors presented the inspection results to Mr. Hoots, and others of the licensee’s staff on October 10, 2008. Licensee personnel acknowledged the inspection results presented. Licensee personnel were asked to identify any documents, materials, or information provided during the inspection that were considered proprietary. Proprietary materials reviewed during the inspection were returned to the licensee.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
D. Hoots, Site Vice President
B. Adams, Plant Manager
S. Greenlee, Director, Engineering
W. Grundmann, Supervisor, Regulatory Assurance

Nuclear Regulatory Commission
B. Bartlett, Senior Resident Inspector

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000454/2008009-01; 05000455/2008009-01	NCV	Failure to Update the Boron Recycle and RHR System Descriptions in the UFSAR
05000454/2008009-02; 05000455/2008009-02	NCV	Failure to Analyze Inlet Piping Loads and Establish an Adequate HUT Quench Volume

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but rather, that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document in this list does not imply NRC acceptance of the document.

4OA2 Identification and Resolution of Problems (71152)

CN-CRA-00-47; Braidwood/Byron Doses from Recycle Holdup Tanks and Spent Resin Tank Failures; dated June 7, 2000

BOP RH-6; Operation of the RH System in Shutdown Cooling; Revision 36

AR 00622574; Concerns regarding relief valves and inputs into the HUT; dated April 27, 2007

AR 00624329; Hot Leg Check Valve Leak-By Indication; dated May 2, 2007

AR 00649710; Potential Vulnerability with RHR suction relief valve to HUT; dated July 13, 2007

AR 00679954; Recycle Holdup Tank Level Administrative Controls; dated October 4, 2007

AR 00680626; NRC Potential Green Finding and Associated NCV – HUT Level; dated October 4, 2007

AR 00828394; HUT Design Temperature and Connected Piping Pressure; dated October 8, 2008.

AR 00828451; NRC Identified Concerns About 1/2RH8707A/B Test Failures; dated October 8, 2008

AR 00533719; As Found Relief Valve Test Failed for 1RH8708A; dated September 20, 2006

AR 00178792; 1RH8708A Relief Valve Removed From System Failed Bench Test; October 1, 2003

AR 00744911; 1RH8708B; dated March 5, 2008

AR 00177954; 1RH8708B Relief Valve Removed From System Failed Bench Test; September 28, 2003

AR 00753650; Loss of One Available ECCS Relief Header Flow Path; dated March 22, 2008

AR 00828770; NRC Identified Incomplete UFSAR Information; dated October 9, 2008.

CN-CRA-08-9; Byron/Braidwood RHUT Response to Opening of the RHR Relief Valve; dated September 28, 2008.

CN-CRA-07-50; Byron/Braidwood RHUT Response to Opening of the RHR Relief Valve; dated September 25, 2008

Drawing M-65; Diagram of Boric Acid Processing; Sht 2C; Revision AO

Drawing M-65; Diagram of Boric Acid Processing; Sht 2A; Revision AR

Drawing M-65; Diagram of Boric Acid Processing; Sht 2B; Revision AU

Drawing M-62; Diagram of Residual Heat Removal; Revision BB

Drawing M-61; Diagram of Safety Injection, Sht 1A; Revision AG

Drawing M-136; Diagram of Safety Injection, Sht 1; Revision AV

Drawing M-137; Diagram of Residual Heat Removal; Revision BD

Drawing M-136-3; Boric Acid Processing Auxiliary Building Unit 2; Revision 3G

Drawing M-550; Aux Bldg. El. 346’ & 364’ Boric Acid Processing, Sht 3, Revision P

Drawing M-550; Aux Bldg. El. 346’ & 364’ Boric Acid Processing, Sht 4; Revision K

Drawing Project 4361; Boric Acid Processing Byron Station Unit 1; dated August 18, 1984

Chicago Bridge and Iron Drawing 62573; General Plan 27’ X 24’-9” High Elliptical Roof Tank; dated February 14, 1978

Calculation SD/SA-CVA-117; RHR Suction Relief Valve Flowrate; dated September, 28, 1978

Specification G-678837; Auxiliary Relief Valves ASME Boiler and Pressure Code Section III, Class 2 and 3; dated October 5, 1971

Byron Procedure 1BGP 100-5; Plant Shutdown and Cooldown; Revision 52

LIST OF ACRONYMS USED

AR	Assignment Report
BTP	Branch Technical Position
CFR	Code of Federal Regulations
CVCS	Chemical and Volume Control System
HUT	Boron Recycle System Holdup Tank
IMC	Inspection Manual Chapter
ºF	Degrees Fahrenheit
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
PRT	Pressurizer Relief Tank
psig	Pounds per Square Inch Gauge
RHR	Residual Heat Removal
RHUT	Recycle System Holdup Tank
SDP	Significance Determination Process
SI	Safety Injection
UFSAR	Updated Final Safety Analysis Report