March 13, 2008

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

SUBJECT: BYRON STATION
NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000454/2008006(DRS); 05000455/2008006(DRS)

Dear Mr. Pardee:

On January 31, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed a combined baseline inspection of the Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications at the Byron Station. The enclosed report documents the results of the inspection, which were discussed with Mr. B. Adams and others of your staff at the completion of the inspection on January 31, 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, three 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

NRC Public Document Room or from the Publicly Available Records (PARS) component ofNRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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. 05000454/2008006(DRS); and 05000455/2008006(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

NRC Public Document Room or from the Publicly Available Records (PARS) component ofNRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

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. 05000454/2008006(DRS); and 05000455/2008006(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

DOCUMENT NAME:G:\DRS\Work in Progress\Byron 2008 006 DRS RAL.doc
□ Publicly Available □ Non-Publicly Available □ Sensitive □ Non-Sensitive
To receive a copy of this document, indicate in the concurrence box "C" = Copy without attach/encl "E" = Copy with attach/encl "N" = No copy

OFFICE	RIII		RIII		RIII					
NAME	RLangstaff		RSkokowski		DHills					
DATE	03/07/08		03/11/08		03/13/08					

OFFICIAL RECORD COPY

Letter to Mr. Charles Pardee from Mr. David E. Hills dated March , 2008.

SUBJECT: BYRON STATION
NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000454/2008006(DRS); 05000455/2008006(DRS)

DISTRIBUTION:
RAG1
TEB
MMT
RidsNrrDirsIrib
MAS
KGO
JKH3
RML2
SRI Byron
CAA1
LSL
CDP1
DRPIII
DRSIII
PLB1
TXN
ROPreports@nrc.gov

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-454; 50-455
License Nos:	NPF-37; NPF-66
Report Nos:	05000454/2008006(DRS); 05000455/2008006(DRS)
Licensee:	Exelon Generation Company, LLC
Facility:	Byron Station, Units 1 and 2
Location:	Byron, IL
Dates:	January 14 through January 31, 2008
Inspectors:	R. A. Langstaff (Lead) A. K. Dahbur N. J. Feliz Adorno J. M. Jacobson
Approved by:	D. E. Hills, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000454/2008006(DRS);05000455/2008006(DRS); 01/14/2008 through 01/31/2008; Byron Station, Units 1 and 2; Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications.

The inspection covered a two week announced baseline inspection on evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by four regional based engineering inspectors. Two Green and one Severity Level IV 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. <u>Inspector-Identified and Self-Revealed Findings</u>

Cornerstone: Initiating Events

- Green. The inspectors identified an NCV, having very low safety significance, of license condition 2.C(6) in that the licensee failed to implement and maintain in effect all provisions of the approved fire protection program. Specifically, the inspectors identified that unauthorized transient combustibles were left adjacent to a cable riser in the auxiliary building contrary to implementing fire protection procedures. This issue was entered into the licensee’s corrective action program and the transient combustibles were removed.

  This finding was more than minor because a credible fire scenario involving the transient combustibles could affect equipment important to safety. This finding was of very low safety significance because the transient combustibles represented a low degradation against the combustible controls program. This finding had a cross-cutting aspect in the area of Human Performance because the licensee failed to appropriately plan work activities by incorporating job site conditions (H.3(a)) in that a work bench located next to cable risers resulted in transient combustibles being staged in a location which presented a credible fire scenario. (Section 1R17.2.b.2)

Cornerstone: Mitigating Systems

- Severity Level IV. The inspectors identified a Severity Level IV NCV, having very low safety significance, of 10 CFR 50.59, “Changes, Tests, and Experiments,” from the licensee’s failure to provide a documented basis for determining that changes in how operator response times for postulated steam generator tube ruptures were credited in accident analyses did not require prior NRC approval. This issue was entered into the licensee’s corrective action program.

  Because the issue 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 the changes would not have ultimately required NRC prior approval. The finding was determined to be of very low safety significance because there was evidence that operator actions could be performed in time to prevent overfill of a steam generator. This finding had a cross-cutting aspect in the area of Human Performance because the licensee failed to ensure that training was adequate to assure nuclear safety (H.2(b)) in that training for 10 CFR 50.59 safety evaluations and screenings was deficient. (Section 1R17.1.b.1)

- Green. The inspectors identified an NCV having very low safety significance of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to take prompt corrective actions for a condition adverse to quality. Specifically, when it was identified in 2003 that the magnetic trip setting for breakers associated with three essential service water motor operated valves (MOVs) was below calculated required values for motor reversal conditions, the licensee neither performed an evaluation which justified operability nor did the licensee provide procedural guidance to operators which addressed the setting for the breakers. The setting for the breakers was modified in 2006 to address MOV functionality. This issue was entered into the licensee's corrective action program.

  This finding was more than minor because the failure to assure that the feeder breakers would not spuriously trip due to motor reversal effect could have affected the reliability of the service water MOVs to respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance because the safety function for the affected MOVs would not have been totally lost in the event that the MOVs were subjected to potential motor reversal conditions. The inspectors determined that there was no cross-cutting aspect to this finding. (Section 1R17.2.b.1)

B. <u>Licensee-Identified Findings</u>

None

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17)

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

The inspectors reviewed seven evaluations performed pursuant to 10 CFR 50.59 to determine if the evaluations were adequate and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 25 screenings where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. In regard to the changes reviewed where no 10 CFR 50.59 evaluation was performed, the inspectors verified that the changes did not meet the threshold to require a 10 CFR 50.59 evaluation. The evaluations and screenings were chosen based on risk significance, safety significance, and complexity. The list of documents reviewed by the inspectors is included as an attachment to this report.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000.

b. Findings

b.1 Failure to Perform 10 CFR 50.59 Evaluations for Changes in Assumed Operator Times

Introduction: The inspectors identified two instances where the licensee failed to provide a documented basis when determining that changes in operator response times for steam generator tube ruptures (SGTRs) that were credited in accident analyses did not require prior NRC approval. The issue was of very low safety significance and was dispositioned as a Severity Level IV Non-Cited Violation (NCV) of 10 CFR 50.59, "Changes, Tests, and Experiments." In addition, the inspectors identified that the issue was associated with the Resource component of the Human Performance cross-cutting area because the licensee failed to ensure that training was adequate to assure nuclear safety in that training for 10 CFR 50.59 safety evaluations and screenings was deficient.

Description: In April 2006, the licensee performed a screening (screening 6E-06-0041) to revise the Updated Final Safety Analysis Report (UFSAR) to reflect a change to operator response times in response to steam generator tube rupture margin-to-overfill analyses. The UFSAR credited assumed operator response times of 11 minutes for isolating auxiliary feedwater (AFW) flow to a ruptured steam generator and 31 minutes for completing remaining mitigating actions for the margin-to-overfill case in SGTR analyses. The 11 and 31 minute times originated from the nominal times assumed in the analyses for replacement steam generators (installed on Unit 1) which was reviewed by the NRC and discussed in a safety evaluation report dated January 28, 1998. The subsequent UFSAR change for which the screening was performed allowed a two second credit for performing remaining mitigating actions for every second AFW was isolated early (hereafter referred to as the “2 for 1 credit”).

The licensee had documented a technical justification for the “2 for 1 credit” in memorandum NFM:PSA 99-050, “Operator Response Time Requirements for B/B SGTR Analysis,” dated July 28, 1999. The licensee memorandum stated that if the observed time to isolate the ruptured steam generator was less than 11 minutes, for every second of early isolation, the criterion for completing the remaining mitigation actions could be increased by two seconds. The inspectors noted that the calculations which supported the “2 for 1 credit” included RETRAN computer analyses which evaluated the margin to overfill based on operator actions being timed in simulator scenarios. The computer analyses demonstrated that mitigating actions could take as long as 39 minutes provided that auxiliary feedwater was isolated sufficiently early. However, memorandum NFM:PSA 99-050 did not establish an upper limit for an acceptable time associated with mitigating actions.

Screening 6E-06-0041 failed to identify the change as requiring evaluation under 10 CFR 50.59. The inspectors identified the following concerns associated with the screening:

- Use of the “2 for 1 credit” would allow operators to take longer than the 31 minutes specified in the UFSAR prior to the change. Allowing operators to take more than the 31 minutes assumed in the UFSAR was a change in the non-conservative permitted to be slower direction to procedures as described in the UFSAR, i.e., operators were allowed to perform mitigating actions using more time than what was assumed for the UFSAR accident analyses.

- The January 28, 1998, Safety Evaluation Report for SGTR analyses acknowledged that, in some simulator runs, operating crews had exceeded the times assumed in the steam generator tube rupture analyses up to 31 minutes 40 seconds for overall response time. The NRC had considered the differences between the assumed and demonstrated times to be acceptable because the differences were small, in addition to the sensitivity analyses performed by the licensee which showed that margin to overfill would be maintained. The screening did not address the limitation of differences in operating response times being small.

- The screening took credit for a design basis SGTR drill guide as already having the “2 for 1 credit.” The inspectors noted that no 10 CFR 50.59 screening had been performed for the incorporation of the “2 for 1 credit” into the drill guide. In addition, the inspectors noted that the drill guide did not form part of the licensing basis for Byron Station.

Title 10 CFR 50.59(a)(5) defines procedures as described in the UFSAR as those procedures that contain information described in the UFSAR such as how structures, systems, and components are operated and controlled (including assumed operator actions and response times). The inspectors noted that Section 4.2.1.2, “Screening of

Changes to Procedures as Described in the UFSAR," of NEI 96-07, Revision 1, provided the following guidance: "Changes are 'screened in' (i.e., require a 10 CFR 50.59 evaluation) if they adversely affect how SSC [system, structure, or component] design functions are performed or controlled (including changes to UFSAR-described procedures, assumed operator actions and response times)." The inspectors noted that the UFSAR change allowed operators to perform mitigating actions using more time than what was assumed in the accident analyses described in the UFSAR. As such, the inspectors considered the change to be adverse and concluded that a 10 CFR 50.59 safety evaluation was required.

The inspectors identified a second example of a failure to perform a 10 CFR 50.59 evaluation when required. Specifically, in October 2007, the licensee completed screening 6E-07-0117 which addressed compensatory measures associated with identified SGTR analysis non-conservatisms. The compensatory measures were discussed in Operability Evaluation 07-007. The analysis potential non-conservatisms involved steam generator power operated relief valve flow rates and decay heat calculations. The non-conservatisms resulted in less margin to overfill than what was originally believed. As a compensatory measure, the licensee took analysis credit for operators isolating auxiliary feedwater less than the 11 minute time assumed in the UFSAR. Specifically, the licensee credited operators isolating auxiliary feedwater in 8 minutes 49 seconds, which was the longest time for isolating auxiliary feedwater recorded during evaluations of operator response times conducted in 2006 and 2007. Taking credit in this manner was a change to procedures described in the UFSAR. Because the analysis assumptions were revised such that operator actions would have to be performed faster than what was assumed in the UFSAR, the inspectors considered the change to be adverse and concluded that a 10 CFR 50.59 safety evaluation was required.

The inspectors noted that the licensee's administrative procedure for performing 10 CFR 50.59 safety evaluations and screenings contained appropriate guidance in this area from NEI 96-07. Based on discussions with licensee individuals who had prepared 10 CFR 50.59 screenings, licensee staff did not recognize that changing how operator actions were credited for accident analyses could require a 10 CFR 59.59 safety evaluation. As such, the inspectors concluded training provided to those who prepared and reviewed 10 CFR 50.59 screenings was deficient.

Analysis: The inspectors determined that this issue was a performance deficiency warranting a significance evaluation because the licensee failed to provide a basis as to why non-conservative changes to actual and assumed operator response times did not require prior NRC approval. The finding was determined to be more than minor because the inspectors could not reasonably determine that the changes would not have ultimately required NRC prior approval.

Because violations of 10 CFR 50.59 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. In this case, the inspectors completed a significance determination of the underlying technical issue using NRC's inspection manual chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor

Inspection Findings for At-Power Situations," dated January 10, 2008. This finding affected the mitigating systems reactor safety cornerstone and was determined to be of very low-safety significance using Phase 1 screening because the licensee had provided evidence that the operator actions could be performed in time to prevent overfill of a steam generator, albeit with less margin. In accordance with the Enforcement Policy, the violation was therefore classified as a Severity Level IV violation. In addition, the finding affected the Resources component of the cross-cutting area of Human Performance because the licensee failed to ensure that training was adequate to assure nuclear safety (H.2(b)). Specifically, the training provided to preparers and reviewers of 10 CFR 50.59 screenings did not ensure that licensee staff understood that changes to how operator actions were credited in analyses could require a 10 CFR 50.59 safety evaluation.

Enforcement: Title 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments as described in the UFSAR. These records must include a written evaluation which provides a basis for the determination that the change, tests, or experiments does not require a license amendment. Contrary to the above, as of January 31, 2008, the licensee made changes pursuant to 10 CFR 50.59(c) to procedures as described in the UFSAR and had not performed a written evaluation which provided the bases for determining that the changes did not require a license amendment. Specifically, the licensee changed procedures as described in the UFSAR by revising analysis assumptions which would allow operators to perform mitigating actions in response to a steam generator tube rupture in a time greater than the 31 minutes discussed in the UFSAR (screening 6E-06-0041). In addition, to support an operability evaluation, the licensee credited operator response to a stream generator tube rupture as less time than that assumed in the UFSAR (screening 6E-07-0117). Once this issue was identified, the licensee initiated corrective actions by placing the issue into their corrective action program (Issue Report (IR) 00729270). Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000454/2008006-01(DRS); 05000455/2008006-01(DRS))

Review of Permanent Plant Modifications

The inspectors reviewed 14 permanent plant modifications that had been installed in the plant during the two years prior to the inspection. The modifications were chosen based upon risk significance, safety significance, and complexity. The inspectors reviewed the modifications to verify that the completed design changes were in accordance with the specified design requirements, and the licensing bases, and to confirm that the changes did not adversely affect any systems safety function. Design and post-modification testing aspects were reviewed to ensure the functionality of the modification, its associated system, and any support systems. The inspectors also used applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors is included as an attachment to this report.

b. Findings

b.1 Inadequate Corrective Action for Motor Operated Valve Breaker Magnetic Trip Settings:

Introduction: The inspectors identified that the licensee failed to take prompt interim corrective actions associated with the magnetic trip setting for breakers associated with three essential service water motor operated valves (MOVs). Specifically, when it was identified in 2003 that the setting for the breakers was below calculated required values for motor reversal conditions, the licensee neither performed an evaluation which justified operability nor did the licensee provide procedural guidance to operators which addressed the setting for the breakers. This issue was considered to be of low safety significance (Green) and was dispositioned as an NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action."

Description: Licensee Event Report 1999-012 documented an event at Clinton Power Station. The Licensee Event Report provided an industry event example of a spurious MOV motor reversal breaker trip due to insufficient breaker trip setting margin from forward momentum of a motor rotor being countered by reverse power (known as motor reversal condition). A motor reversal condition arises when an MOV motor rotor is in rotation to move a valve stem in one direction, but suddenly the motor stator electrical configuration changes to move the valve stem in the other direction. Under these conditions, a line current in excess of locked rotor current will be experienced.

As a result of the event at Clinton Power Station, the licensee at Byron prepared a modification design change package (EC 0334667), in 2003, which included a calculation for the new desired breaker magnetic trip setting for three essential service water MOVs (0SX007, 1SX007 and 2SX007) that were found susceptible to motor reversal conditions. The calculation indicated that the existing magnetic trip setting for the three feeder breakers for the service water MOVs was below the minimum magnetic trip current value. The existing breakers magnetic setting was 5 (corresponding to 24 Amps) and the calculated minimum magnetic trip current value was 27.6 Amps. The calculation also indicated that a new magnetic trip setting value of 9 (corresponded to magnetic trip current of 33 Amps) would offer an improvement in the reliability of the valves to perform their design safety functions in the event of a motor reversal conditions. However, it was not until 2006, that the licensee implemented the change to increase the breakers magnetic trip setting for all three MOVs from the previous value of 5 to the new desired magnetic trip setting value of 9.

The inspectors questioned whether an operability evaluation was performed or interim corrective actions were established when the licensee determined that the previous setting for the breakers was found to be below the calculated values. The licensee stated that they had provided procedural guidance to operators in 2006 which addressed the reversing direction motor current concern by providing instructions for the operators to pause at least 10 seconds after open or closed indication was received before giving the MOV another start. The instructions also indicated that magnetic overload breaker trips can occur during routine valve operation if the operator did not pause long enough between operating the valve in one direction and manipulating the control switch to operate the valve in the opposite direction. However, the licensee was unable to identify any similar guidance made available to operators or an evaluation justifying operability in 2003, when the concern was identified.

Analysis: The inspectors determined that the failure, in 2003, to justify operability or perform interim corrective actions, such as providing procedural guidance to operators, to provide assurance that the breakers for the three essential service water MOVs would not spuriously trip in the event of motor reversal conditions was a performance deficiency warranting a significance evaluation. The inspectors determined that the performance deficiency was more than minor in accordance with IMC 0612, Appendix B, "Issue Disposition Screening," dated September 20, 2007, because the finding was associated with the design control attribute of the Mitigating System cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of three essential service water MOVs to respond to initiating events to prevent undesirable consequences. Specifically, the feeder breakers for the service water MOVs, 0SX007, 1SX007, and 2SX007 could spuriously trip in the event of motor reversal conditions and challenged the ability of these MOVs to perform their required safety functions during accident conditions. The safety function of MOVs 0SX007, 1SX007 and 2SX007 was to regulate or isolate service water system flow through component cooling water system heat exchangers 0CC01A, 1CC01A, and 2CC01A, respectively.

In 2006, the licensee implemented a design change and increased the magnetic trip setting for the feeder breakers associated with the three essential service water MOVs to prevent spurious trips due to motor reversal conditions. However, from the time the deficiency was identified in 2003 until 2006, the licensee did not have any interim corrective actions such as an evaluation justifying operability or procedural guidance for operators which addressed the issue.

The inspectors evaluated the finding using the Phase 1 screening discussed in IMC 0609, Appendix A. The inspectors answered "No" to all the screening questions because in the event of the supplied power loss (breakers spuriously tripped), due solely to motor reversal effects, the essential service water MOVs would remain in their position (throttle) and their safety function could be degraded but not lost. Therefore, the finding screened as having very low safety significance (Green). The inspectors did not identify a cross-cutting aspect to this finding because the performance deficiency occurred more than two years prior to the inspection and, as such, was not necessarily representative of current licensee performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" requires, in part, that conditions adverse to quality are promptly identified and corrected. Contrary to this requirement, from 2003 to 2006, the licensee did not implement corrective actions for a condition adverse to quality associated with three essential service water MOVs. Specifically, prior to the licensee implementation of design change EC 334667 in 2006, the licensee did not have corrective actions, such as an evaluation justifying operability or procedural guidance for operators which addressed the issue, to provide assurance that the feeder breakers for MOVs 0SX007, 1SX007 and 2SX007 would not spuriously trip due to motor reversal conditions. However, because this violation was of very low safety significance and because the issue was entered into the licensee's corrective action program (IR 00730300), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 0500454/2008006-02 (DRS); 0500455/2008006-02 (DRS)) b.2 Unauthorized Transient Combustibles:

Introduction: The inspectors identified unauthorized transient combustibles in the auxiliary building. The issue was considered to be of very low safety significance (Green) and was dispositioned as an NCV of license condition 2.C(6) for implementation of the fire protection program. In addition, the finding was associated with the Work Control component of the Human Performance cross-cutting area because the licensee failed to appropriately plan work activities by incorporating job site conditions in that a work bench located next to cable risers resulted in transient combustibles being staged in a location which presented a credible fire scenario.

Description: On January 15, 2008, the inspectors identified Class A transient combustibles on a metal work bench adjacent to a cable riser in the auxiliary building. Specifically, a plastic bin filled with cotton glove liners and a three-ring binder were left on the work bench. The materials were less than two feet from the cable risers. The licensee initiated corrective action by removing the transient combustible materials, relocating the work bench, and placing the issue into their corrective action program (IR's 00722678 and 00729683). Section 4.4.2, Paragraph 6 of Procedure OP-AA-201-009, "Control of Transient combustible Material," Revision 6, directed licensee personnel to avoid staging exposed Class A combustible material immediately adjacent to (i.e., within approximately three feet) cable risers.

Analysis: The inspectors determined that staging transient combustible materials near cable risers was a performance deficiency, warranting a significance evaluation. Specifically, the procedure for implementing transient combustible controls required that Class A transient combustibles not be staged unattended near cable risers. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because the failure to adequately control combustible materials was associated with an increase in the likelihood of an initiating event, i.e., fire. In addition, the issue met the threshold of being more than minor, as discussed in example 4.k of IMC 0612, Appendix E, "Examples of Minor Issues," dated September 20, 2007, because a credible fire scenario involving the transient combustibles could affect equipment important to safety. Specifically, the transient combustibles were located within the zone of influence for a 70 kW fire for the thermoset cables located within the vertical cable tray risers (IMC 0609, Appendix F, "Fire Protection Significance Determination Process," dated February 28, 2005, Table 2.3.2). Consequently, the transient combustibles presented a credible fire scenario involving equipment important to safety (such as the safety related cables within the vertical cable risers). The cable risers had cables for safety-related systems which included auxiliary feedwater, component cooling water, containment spray, safety-injection, and service water. In addition, the finding affected the Work Control component of the cross-cutting area of Human Performance because the licensee failed to appropriately plan work activities by incorporating job site conditions (H.3(a)). In this instance, the work bench was used for work activities by radiation protection personnel which resulted in unattended staging of transient combustible materials on the tables. The placement of the work bench near the cable risers failed to consider job site conditions in that the placement of unattended transient combustible materials on the tables presented a credible fire scenario involving equipment important to safety and was contrary to site procedures.

The inspectors reviewed IMC 0609, Appendix A, and determined that since the finding affected administrative controls for fire protection, a significance determination evaluation under IMC 0609, Appendix F, was required. The inspectors completed a significance determination of this issue using IMC 0609, Appendix F, Attachment 2, “Degradation Rating Guidance Specific to Various Fire Protection Program Elements,” dated February 28, 2005. The inspectors determined that the staging of Class A combustibles was a low degradation finding against the combustible controls program because the identified materials would not cause a fire from existing sources of heat or electrical energy. Question 1 of IMC 0609, Appendix F, Task 1.3.1, “Qualitative Screening for All Finding Categories,” showed that the finding was of very low safety significance (Green) due to the low degradation rating.

Enforcement: License condition 2.C(6) required the licensee to implement and maintain in effect all provisions of the approved fire protection program as described in the licensee’s Fire Protection Report, and as approved in the Safety Evaluation Report dated February 1987 through Supplement No. 8. Section 9.5.1, “Fire Protection Systems,” of the UFSAR, stated that the design bases, system descriptions, safety evaluation, inspection and testing requirements, personnel qualification, and training were described in the Byron/Braidwood Fire Protection Report. Section 3.2, Paragraph c., of the Byron/Braidwood Fire Protection Report stated that the station complied with the NRC guideline that administrative controls should be used to maintain the performance of the fire protection system and personnel. The controls established procedures to govern the handling of and limit transient fire loads such as combustible and flammable liquids, wood and plastic products, or other combustible materials in buildings containing safety related systems or equipment during all phases of operating, and especially during maintenance, modification, or refueling operations. Procedure OP-AA-201-009 provided the administrative controls to satisfy the commitment outlined in Section 3.2, Paragraph C., of Byron/Braidwood Fire Protection Report. Procedure OP-AA-201-009 directed licensee personnel to avoid staging exposed Class A combustible material immediately adjacent to (i.e., within approximately three feet) cable risers. Contrary to the above, on January 15, 2008, the inspectors identified transient combustibles on a metal work bench adjacent to a cable riser in the auxiliary building. Specifically, a plastic bin with filled with cotton glove liners and three-ring binder were left on the work bench. Once identified, the licensee removed the transient combustible materials and entered the issue into their corrective action program (Issue Reports 00722678 and 00729683). Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000454/2008006-03(DRS); 05000455/2008006-03(DRS))

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

During this inspection, the inspectors reviewed four corrective action documents that identified or were related to 10 CFR 50.59 evaluations and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent plant modifications and evaluations for changes, tests, or experiments issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA5 OTHER ACTIVITIES

.1 Pressurized Water Reactor Containment Sump Blockage (Temporary Instruction 2515/166)

a. Inspection Scope

The purpose of this Temporary Instruction (TI) was to support Nuclear Regulatory Commission review of licensee's activities in response to NRC Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors (PWRs)." This TI required NRC inspectors to verify actions implemented in response to NRC GL were complete and where applicable were programmatically controlled.

The inspectors performed a review of the licensee's activities related to GL 2004-02 in accordance with TI 2515/166. The inspectors also reviewed the documentation supporting the engineering, procedure changes, and testing associated with the containment sump strainer modification on both units.

b. Inspection Documentation

The questions posed by TI 2515/166 and associated status are outlined below:

(1) Question: Did the licensee implement the plant modifications and procedure changes committed to in their GL 2004-02 responses? List the commitments and the actions taken to meet each commitment. List when each action to meet each commitment was completed. State whether additional inspections are required to ensure all commitments have been met by the plant.

- Commitment: If a strainer modification is required, Byron Station will complete a preliminary debris head loss analysis by September 1, 2005. The final debris head loss analysis will be completed as part of the strainer modification process in accordance with the NRC schedule for Generic Safety Issue (GSI)-191 resolution.

  Status: Byron had completed a preliminary debris head loss analysis (BYR05-042, Rev 0) by September 1, 2005 and a final debris head loss analysis (3 SA-096.018) was completed prior to December 31, 2007.

- Commitment: Byron Station will complete the containment walkdown surveillance for potential debris sources during the next scheduled outage for Unit 1 and Unit 2.

  Status: Containment walkdown surveillances for potential debris sources were completed for Unit 1 in March 2005 and in September 2005 for Unit 2.

- Commitment: The recirculation functions for the emergency core cooling systems and containment spray systems for Byron Station will be in compliance with the requirements listed in the Applicable Regulatory Requirements section of GL 04-02 by December 31, 2007.

  Status: Installation of the new strainers for Byron was previously reviewed and documented in NRC Inspection Reports 05000454/2006004 and 05000455/2007003. The recirculation functions for the emergency core cooling systems and containment spray systems are in compliance with the regulatory requirements except for downstream effects on Unit 1. The Exelon Generation Company (EGC) requested and received approval (letter from R. F. Kuntz, NRC, to C. M. Crane, EGC, dated July 21, 2006), for an extension until Spring 2008 to complete the installation and testing of emergency core cooling system throttle valves and containment spray system cyclone separators for Unit 1.

- Commitment: Overall completion of the downstream effects analysis, including the fuel impact under the current Westinghouse WCAP guidance, will occur by January 31, 2006.

  Status: The original downstream effects analyses, including the fuel impact under the current Westinghouse WCAP guidance, were completed by January 31, 2006. The analyses are BYR05-043 (GSI-191 Downstream Effects Flow Clearances), BYR05-061 (GSI-191 Evaluation of Long Term Downstream Effects), and BYR06-17 (GSI-191 Downstream Effects - Vessel Blockage and Fuel Evaluation). BYR05-061 was revised and CN-SEE-1-07-38 (GSI 191: LOCADM Analysis) in December 2007 to incorporate more recent guidance on downstream effects in WCAPs 16406 and 16793.

- Commitment: The chemical effects analysis will be complete by January 31, 2006.

  Status: The original chemical effects analysis, BYR05-059 (GSI-191 Chemical Effects Evaluation), was complete on January 27, 2006. This analysis was augmented by BYR06-030, “Post - LOCA Chemical Effects Analysis,” in September 2006 to incorporate Westinghouse WCAP guidance.

- Commitment: Byron will validate that adequate margin exists to bound the impact of chemical effects once the vendor's test results to quantify chemical debris effect on head loss have been published.

  Status: The head loss testing, performed by CCI, included chemical effects. This is documented in calculation 3 SA-096.018 (Head Loss Calculation). The acceptability of the resulting head loss is documented in BYR06-058 (NPSHA for RHR and CS Pumps During Post - LOCA Recirculation).

(2) Question: Has the licensee updated its licensing bases to reflect the corrective actions taken in response to GL 2004-02? Licensing bases may not be updated until the licensee fully addresses GL 2004-02 (by December 31, 2007, unless an extension has been granted).

Status: The licensing bases have been updated except for the portion relative to the installation of the downstream effects modification on Unit 1. The EGC requested and received approval (letter from R.F.Kuntz, NRC, to C. M. Crane, EGC, dated July 21, 2006), for an extension until Spring 2008 to complete the installation and testing of emergency core cooling system throttle valves and containment spray system cyclone separators for Unit 1.

(3) Question: If the licensee or plant has obtained an extension past the completion date of this TI, document what actions have been completed, what actions are outstanding, and close the TI for the plant that has the extension. Items not finished by the TI completion date can be inspected in the future using the generic refueling outage inspection procedure.

Status: The EGC requested and received approval (letter from R. F. Kuntz, NRC, to C. M. Crane, EGC, dated July 21, 2006), for an extension until Spring 2008 to complete the installation and testing of emergency core cooling system throttle valves and containment spray system cyclone separators for Unit 1.

Actions that are complete, including all associated design analysis are:

- Installation of new sump screens, both units;
- Installation of new emergency core cooling system throttle valves, Unit 2; and
- Replacement of fiberglass insulation on steam generators (in the zone of influence) with reflective metallic insulation (Unit 1).

Temporary Instruction 2515/166 will remain open for both Unit 1 and Unit 2 pending review of:

- Installation of new emergency core cooling system throttle valves, Unit 1 (scheduled for Spring 2008); and
- Installation of containment spray system cyclone separator, Unit 1 (scheduled for Spring 2008).

The status of final licensee commitments, documented by letter dated December 31, 2007, will be discussed in a future NRC inspection report.

4OA6 Meeting(s)

.1 Exit Meeting

The inspectors presented the inspection results to Mr. B. Adams and others of the licensee’s staff on January 31, 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. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
B. Adams, Plant Manager
E. Blondin, Supervisor, Design Engineering
G. Contrady, Nuclear Oversight
A. Giancatarino, Peformance Improvement
S. Greenlee, Director, Engineering
W. Grundmann, Supervisor, Regulatory Assurance
V. Nashansky, Supervisor, Design Engineering

Nuclear Regulatory Commission
B. Bartlett, Senior Resident Inspector

State of Illinois
C. Thompson, Illinois Emergency Management Agency

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000254/2008006-01(DRS) 05000255/2008006-01(DRS)	Failure to Perform 10 CFR 50.59 Evaluations for Changes in Assumed Operator Times
05000254/2008006-02(DRS) 05000255/2008006-02(DRS)	Inadequate Corrective Actions for Motor Operated Valve Breaker Magnetic Trip Settings
05000254/2008006-03(DRS) 05000255/2008006-03(DRS)	Unauthorized Transient Combustibles

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.

<u>10 CFR 50.59 Safety Evaluations</u>

6G-06-0001; TRM Change 06-019, TRM Section 3.9.a “Decay Time;” Revision 0 (Ron)

6G-06-0003; Removal of Recirculation Sump Level Indication - Byron Unit 1; Revision 0

6G-07-0001; Removal of Recirculation Sump Level Indication - Byron Unit 2; Revision 0

6G-07-0002; Interim Procedure Revision to Install a Jumper in 2PA10J to Support Surveillance Testing of 2PA09J; Revision 0

6G-07-0003; EC 364080, Heavy Load Drop Issues in Turbine Building – SX Piping; Revision 2

6G-07-0004; Modification to the Safety Injection throttle Valves 2SI8810 A-D, 2SI8816 A-D, and 2SI8822A-D; Revision 0

6G-07-0007; EC 364470, 1SX136 and 1SX011 Valve Replacement; Revision 0

<u>10 CFR 50.59 Screenings</u>

6D-06-0036; Add Steps to Allow Refill of SFP Through PW in FHB and Aux Building; Revision 0

6D-06-0063; BOP AF-7/BOP AF-7T1/BOP AF-12, Diesel Driven AF Pump B Startup on Recirc/Diesel Driven Auxiliary Feedwater Pump Operating Log/Filling the AF Diesel Jacket Water System; Revision 0

6D-06-0107; Power Decension – Revise Procedure 1/2 BGP 100-4; Revision 0

6D-06-0108; UFSAR Change for Figure 6.3-3, “Residual Heat Removal Pump Performance Curve”; Revision 0

6D-07-0038; BOP RH-6, Operation of the RH system in shutdown cooling; Revision 0

6D-07-0107; 1B Diesel Generator Voltage Regulator Test Special Procedure; Revision 1

6D-07-0120; TRM Revision to Section 1.0, “Use and Application to Allow Departure from TRM Requirements; Revision 0

6D-07-0124; Revision to TRM Section 1.5, “TLCO and TSR Implementation”; Revision 0

6D-07-0127; BOP CC-10, Alignment of the U-0 CC Pump and U-0 CC HX to a Unit; Revision 0

6D-07-0128; BOP CC-10, Alignment of the U-0 CC Pump and U-0 CC HX to a Unit; Revision 0

6E-06-0032; Primary Coolant Sources Outside Containment Program; Revision 2

6E-06-0041; DRP 11-072 – UFSAR Change to Clarify SGTR Operator Response Time Requirement; Revision 0

6E-06-0045; Revision to UFSAR Section 10.4.8.1, “Design Bases” for the Steam Generator Blowdown System; Revision1

6E-06-0052; Revise UFSAR Table 15.0-7 and Associated Figures and Text to Identify Systems and Equipment “Credited”; Revision 0

6E-06-0108; Replacement of the Containment Recirculation Sump Screens; Revision 0

6E-06-0120; Revision to UFSAR Table 3.12-2, Note 5, to Change the Maximum Abnormal Temperature in the MEERs; Revision 0

6E-07-0032; Revise the Applicable Byron Section of Table 8.3-5 to Reflect Loading in Accordance with the UHS DBA; Revision 0

6E-07-0033; Revise T.S. Bases Section B 3.66, Containment Spray and Cooling Systems; Revision 0

6E-07-0041; EC 355895, Installation of Blank Off Plate in VC Ductwork Upstream of Damper 0VC05Y; Revision 0

6E-07-0056; DRP 12-018, DRP for Describing Additional Analyses for Containment Spray Pumps; Revision 2

6E-07-0102; Document As-Built Pressurizer Relief Tank Temperature High Alarm Setpoint of 115o F; Revision 0

6E-07-0104; Revise Pressurizer PORV Accumulator Air Pressure Low Alarm Setpoint; Revision 0

6E-07-0114; Revise Feedwater Flow Transmitter 1FT-0531 and 2FT-0540 Calibration to Include Static Pressure Correction for the Resemount 1153 Series B dp Transmitter; Revision 0

6E-07-0117; Byron Station Operability Evaluation 07-007, Evaluation of Compensatory Measures Associated with Degraded Condition; Revision 1 (Ron)

6E-07-0125; Remove Elbows from Essential Service Water Make - up Lines at the SX Cooling Tower; Revision 0

<u>Corrective Actions (Issue Reports) Initiated as a Result of Inspection</u>

00722678; NRC Identified Transient Combustible Issue; dated January 18, 2008

00729270; NRC Identified 10CDR50.59 Screening Issue; dated January 30, 2008

00729610; 50.59 Evaluation Not Performed for BGP 100-4

00729641; NRC Concern of Potential Tech Spec Violation; dated January 31, 2008

00729683; NRC Identified HU Work Control Cross-Cutting Issue; dated January 31, 2008

00730300; NRC Audit Identified Potential Ineffective Corrective Action; dated February 1, 2008

<u>Corrective Actions</u>

00461308; Design Basis SGTR Operator Acceptance Criteria Challenge; dated March 2, 2006

00635159; RY PROV Accumulator Instrument Discrepancies; dated May 30, 2007

00644501; Potential Non-Conservatism In Steam Relief Capacity Modeling; dated June 26, 2007

00692498; TRM Revision 54 Deficient; dated October 31, 2007

Engineering Analyses (Engineering Changes)

12.1.11.15-BYR07-054; Evaluate the structural adequacy of Non-Standard Detail for Comparison Angle/End-Cap, located at the end of the branch line housing balancing damper No. 0VC305Y; Revision 0

CCI Report 680/41222; Chemical Test Filter Performance Report; Revision 3

EC 367065; OP Eval 07-007, Main Steam PORV Steam Relief Capacity; Revision 1

EC 367423; Evaluation of Decay Heat Impact on the Byron & Braidwood SGTR Analysis; Revision 0

EC 367774; Steam Generator Tube Rupture Operator Response Time Sensitivity Evaluation; Revision 0

NFM-PSA 90-050; Operator Response Time Requirements for B/B SGTR Analysis; dated July 28, 1999

PSA-B-97-24; B/B Observed Operator Response Time Evaluation; Revision 0

Permanent Plant Modifications

EC 334667; Change Load Breaker Magnetic Trip Setting for Valve 0SX007; Revision 2

EC 346092; 2B AF Diesel Enhancements for Monitoring Equipment and Move Governor Oil Reservoir Down to Prevent Siphoning; Revision 1

EC 346092; 2B AF Diesel Enhancements for Monitoring Equipment and Move Governor Oil Reservoir Down to Prevent Siphoning; Revision 1

EC 348110; 2B AF Pump Bearing Lube Oil Line Reroute; Revision 1

EC 349674; Implement Logic Change to Valve 1SX147A; Revision 1

EC 350266; 2B Diesel Driven Auxiliary Feedwater Pump Sensing Line Rerouting; Revision 3

EC 355895; Install Blank Off Plate in VC System 0VC305Y; Revision 0

EC 359447; Replacement of the Controller 1MS018JB for the Main Steam Power Operated Relief Valve (PORV) 1MS018B; Revision 1

EC 360138; Revise 0A SX Makeup Pump Low Lube Oil Pressure Pump Trip Time; Revision 0

EC 362163; Criteria for ½ BEP-0 to Check if RCS is Intact Due to Removal of Recirculation Sump Level Switches; dated July 30, 2007

EC 362644; Installation of High Point Vent Assemblies on Line 1SI01B-24"; Revision 1

EC 364470; Perform Linestop and Draining of 1SX03B-42” to Support Replacement of Valves 1SX011 and 1SX136; Revision 2

EC 364979; Evaluate SI Throttle Valve Test Results from Wyle Labs to Document Acceptability of New Trim Design; Revision 0

EC 367972; Remove Elbows on Discharge of SX Make - Up Pump Supply to SXCT Basins; Revision 0

Procedures

1B Diesel Generator Voltage Regulator Special Procedure; Revision 0

1BCA 1.3; Sump Blockage Control Room Guideline; Revision 2

1BEP-0; Rx Trip or Safety Injection; Revision 108

1BOA PRI-1; Excessive Primary Plant Leakage; Revision 104

1BOA PRI-7; Essential Service Water Malfunction; Revision 103

2BCA 1.3; Sump Blockage Control Room Guideline; Revision 2

2BEP-0; Rx Trip or Safety Injection; Revision 107

2BOA PRI-1; Excessive Primary Plant Leakage; Revision 105

2BOA PRI-7; Essential Service Water Malfunction; Revision 104

BOP AP-73A1; Control and Indication Functions Lost During Bus 141 Outage; Revision 4

BOP AP-74A1; Control and Indication Functions Lost During Bus142 Outage; Revision 3

BOP CC-10; Alignment of the U-0 CC Pump and U-0 CC HX to a Unit; Revision. 18, Interim No. 07-0-052

BOP RH-6; Operation of the RH system in Shutdown Cooling; Revision 34

OP-AA-103-105; Limitorque Motor-Operated Valves Operations, Revision 0

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
CFR	Code of Federal Regulations
EGC	Exelon Generating Company
GSI	Generic Safety Issue
IMC	Inspection Manual Chapter
IR	Issue Report
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
SDP	Significance Determination Process
SGTR	Steam Generator Tube Rupture
TI	Temporary Instruction
UFSAR	Updated Final Safety Analysis Report