UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BLVD., SUITE 100
KING OF PRUSSIA, PA 19406-2713

February 11, 2015

EA-14-178

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

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION – NRC INSPECTION REPORT NO. 05000219/2014009 AND PRELIMINARY YELLOW FINDING/OLD DESIGN ISSUE

Dear Mr. Hanson:

On December 16, 2014, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Oyster Creek Nuclear Generating Station. The enclosed report documents the inspection results, which were discussed on January 29, 2015, with Mr. G. Stathes, Site Vice President, and other members of your staff. This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and conditions of your license. The focus of this inspection was on failures of two electromatic relief valves (EMRVs) identified by your staff during as-found bench testing.

The enclosed inspection report discusses a finding that the NRC has preliminarily determined to be Yellow, a finding of substantial safety significance, and to meet the definition of an old design issue, a past design-related problem that is not indicative of current licensee performance. The criteria for determining whether a finding is an old design issue are contained in Inspection Manual Chapter (IMC) 0305, "Operating Reactor Assessment Program," and the basis for this determination is described in Attachment 1 to the enclosed report. In accordance with IMC 0305, the performance issue will not aggregate in the Action Matrix with other performance indicators and inspection findings if it is finalized as an old design issue. Additionally, depending on the final significance determination, the NRC will perform a supplemental inspection to review your root cause evaluation and corrective action plan for this issue.

As described in Section 4OA2 of the enclosed report, the finding is associated with an apparent violation of Title 10 of the *Code of Federal Regulations* (10 CFR) 50, Appendix B, Criterion III, "Design Control," because Exelon did not establish adequate measures for selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the EMRVs. Specifically, the original design of the EMRV actuators, and the associated maintenance refurbishment processes, were inadequate because they did not account for the impact of vibration associated with plant operation, which caused an increased gap between the posts and guides such that the springs could wedge between them. This inadequate design resulted in two EMRVs being inoperable for a period greater than the Technical Specification allowed outage time.

The NRC determined that this finding does not represent an immediate safety concern since Exelon replaced all of the actuators with redesigned actuators during the refueling outage in October 2014. The NRC assessed this finding based on the best available information using the applicable significance determination process. The basis for the NRC's preliminary significance determination is described in Attachment 3 to the enclosed report. Because the finding is also an apparent violation of NRC requirements, it is being considered for escalated enforcement action in accordance with the NRC Enforcement Policy, which appears on the NRC's Web site at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html.

The NRC will inform you, in writing, when the final significance has been determined in accordance with IMC 0609, "Significance Determination Process." We intend to complete and issue our final safety significance determination within 90 days from the date of this letter. The NRC's significance determination process is designed to encourage an open dialog between your staff and the NRC; however, the dialogue should not affect the timeliness of our final determination.

We believe that we have sufficient information to make a final significance determination. However, before we make a final decision, we are providing you an opportunity to provide your perspective on the facts and assumptions that the NRC used to arrive at the finding and assess its significance. Accordingly, you may notify us of your decision within 10 days to: (1) request a regulatory conference to meet with the NRC and provide your views in person; (2) submit your position on the finding in writing; or, (3) accept the finding as characterized in the enclosed inspection report.

If you choose to request a regulatory conference, the meeting should be held in the NRC Region I office within 30 days of the date of this letter, and will be open for public observation. The NRC will issue a public meeting notice and a press release to announce the date and time of the conference. We encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If you choose to provide a written response, it should be sent to the NRC within 30 days of the date of this letter. You should clearly mark the response as a "Response to Preliminary Yellow Finding/Old Design Issue in Inspection Report No. 05000219/2014009; EA-14-178," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, Region I, and a copy to the NRC Senior Resident Inspector at the Oyster Creek Nuclear Generating Station.

You may also elect to accept the finding as characterized in this letter and the inspection report, in which case the NRC will proceed with its regulatory decision. However, if you choose not to request a regulatory conference or to submit a written response, you will not be allowed to appeal the NRC's final significance determination.

Please contact Silas Kennedy at (610)337-5046 within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. Because the NRC has not made a final determination in this matter, a Notice of Violation is not being issued for this inspection finding at this time. In addition, please be advised that the number and characterization of the apparent violation may change based on further NRC review. The final resolution of this matter will be conveyed in separate correspondence.

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 component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Ho K. Nieh, Director
Division of Reactor Projects

Docket No.: 50-219
License No.: DPR-16

Enclosure: Inspection Report 05000219/2014009
Attachment 1: Evaluation of IMC 0305 Criteria for Old Design Issues
Attachment 2: Supplementary Information
Attachment 3: Detailed Risk Significance Evaluation

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.:	50-219
License Nos.:	DPR-16
Report Nos.:	05000219/2014009
Licensee:	Exelon Nuclear
Facility:	Oyster Creek Nuclear Generating Station
Location:	Forked River, New Jersey
Dates:	June 20, 2014 – December 16, 2014
Inspectors:	A. Patel, Resident Inspector S. Pindale, Senior Reactor Inspector C. Cahill, Senior Reactor Analyst N. Floyd, Reactor Inspector B. Bollinger, Project Engineer
Approved by:	Silas R. Kennedy, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY

IR 05000219/2014009; June 20, 2014 – December 16, 2014; Oyster Creek Nuclear Generating Station; Problem Identification and Resolution Annual Sample.

This NRC team inspection was performed by four regional inspectors and one resident inspector. The inspectors identified one finding of preliminary Yellow significance during this inspection and classified this finding as an apparent violation. The significance of most findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Aspects Within Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated July 9, 2013. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 5.

Cornerstone: Mitigating Systems

- Preliminary Yellow. The NRC identified a preliminary Yellow finding and associated apparent violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," and Technical Specification 3.4.B, "Automatic Depressurization System," because the station did not establish adequate measures for selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the electromatic relief valves (EMRVs). The violation was also preliminarily determined to meet the IMC 0305, Section 11.05, criteria for treatment as an old design issue. Specifically, on June 20, 2014, during refurbishment of EMRVs that were removed from the plant during the 2012 refueling outage, Exelon personnel identified deficiencies with the 'B' and 'D' EMRVs. As part of the planned EMRV actuator testing and refurbishment activities, Exelon personnel conducted bench testing on June 26, 2014. Both valves did not stroke satisfactorily and resulted in two inoperable EMRVs for greater than the Technical Specification allowed outage time of 24 hours. Exelon's immediate corrective actions included placing this issue into the corrective action program as issue report 1679428 and redesigning the EMRV actuators to ensure the spring is on the outside of the guide bushing, therefore removing the possibility of the spring entering the guide bushing area and subsequently jamming the actuator causing valve failure. All of the actuators were replaced with redesigned actuators during the refueling outage in October 2014. In addition, Exelon issued a 10 CFR Part 21 report to inform the industry of the deficient EMRV actuator design.

  This finding is more than minor because it adversely affected the design control quality attribute of the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the design deficiency of the EMRVs and the inadequate maintenance process led to the inability of the 'B' and 'D' EMRVs to perform their safety function. The inspectors screened this issue for safety significance in accordance with IMC 0609, Appendix A, Exhibit 2, and determined a detailed risk evaluation was required because the EMRVs were potentially failed or unreliable for greater than the Technical Specification allowed outage time. As described in Attachment 3 to this report, a detailed risk evaluation concluded that the increase in core damage frequency (CDF) related to failure of the 'B' and 'D' EMRVs is in the mid E-5 range; therefore, this finding was preliminarily determined to have a substantial safety significance (Yellow). Due to the nature of the failures, no recovery credit was assigned. The dominant sequences included

loss of main feedwater with failures of the isolation condensers, and failure to depressurize. This finding does not represent an immediate safety concern because Exelon replaced all of the actuators with the redesigned actuators during the refueling outage in October 2014. Further, the NRC is considering treatment of this finding as an old design issue because the condition existed since the original installation of the EMRVs, and is not indicative of current licensee performance. Additional details are discussed in Attachment 1. The inspectors determined that this finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency was not reflective of current licensee performance. Specifically, the inspectors determined that the performance deficiency existed since original installation of the EMRVs. Although an opportunity to identify this issue following original installation occurred in 2006 when Quad Cities changed the EMRV actuator design due to similar issues, the inspectors could not conclude that the issue would have likely been identified during that period since a Part 21 Report was not issued to inform the industry and NRC of the design change and industry operating experience focused on plants that completed or were scheduled to complete an extended power uprate. [Section 4OA2.1.c]

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution (71152 – 1 sample)

All documents reviewed during this inspection are listed in Attachment 2 to this report.

.1 Annual Sample: Root Cause – Failure of EMRVs ‘B’ and ‘D’

a. Inspection Scope

The inspectors conducted an in-depth review of Exelon’s root cause analysis and corrective actions associated with issue report 1679428 associated with two inoperable EMRVs. Specifically, on June 20, 2014, during refurbishment of EMRVs that were removed from the plant during the 2012 refueling outage, Exelon personnel identified deficiencies with the ‘B’ and ‘D’ EMRVs. As part of the planned EMRV actuator testing and refurbishment activities, Exelon personnel conducted bench testing on June 26, 2014. Both valves did not stroke satisfactorily and were subsequently determined to be inoperable from July 27, 2012, to October 22, 2012.

The inspector assessed the following attributes: operability of the EMRVs, identification of the root and contributing causes, extent of condition reviews and previous occurrences. The inspector also assessed the timeliness of corrective actions and whether they will preclude repetition of the event. The inspector performed reviews of the documents noted in Attachment 2 to this report and interviewed engineering and maintenance personnel to assess the effectiveness of the planned, scheduled, and completed corrective actions to resolve the identified deficiency.

b. Findings

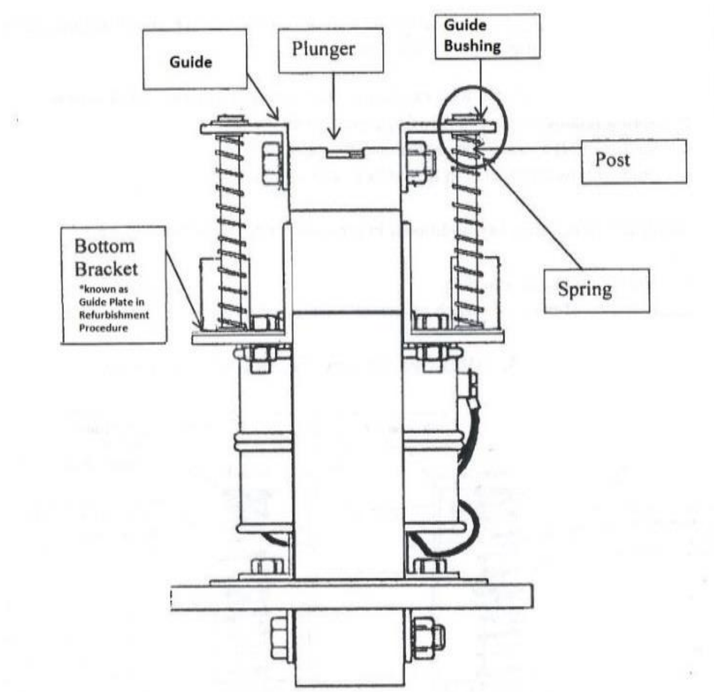
Introduction: A preliminarily Yellow finding and associated violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” and Technical Specifications 3.4.B, “Automatic Depressurization System,” was identified by Exelon for the failure to establish adequate measures for selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the EMRVs. The violation was also preliminarily determined to meet IMC 0305, Section 11.05a criteria for treatment as an old design issue.

Description: The Oyster Creek EMRVs are six-inch electrically actuated pressure relief valves manufactured by Dresser Industries. There are five EMRVs on the main steam lines between the reactor pressure vessel and the main steam line isolation valves within the drywell. The EMRVs consist of a main valve assembly, pilot valve assembly, and a solenoid actuator. When energized, the solenoid actuates a plunger, which pushes down the pilot valve operating lever, thereby opening the pilot valve. When the pilot valve opens, pressure under the main valve disc is vented, resulting in an unbalanced steam pressure across the main disc and moving the main disc downward from its seat opening the main valve.

The function of the EMRVs is to depressurize the reactor during a small break loss-of-coolant accident to permit the low pressure core spray system to inject water into the reactor core. Oyster Creek Technical Specifications section 3.4.B, "Automatic Depressurization System," states, in part, five EMRVs shall be operable and if more than one are inoperable then reactor pressure shall be reduced to 110 psig or less, within 24 hours.

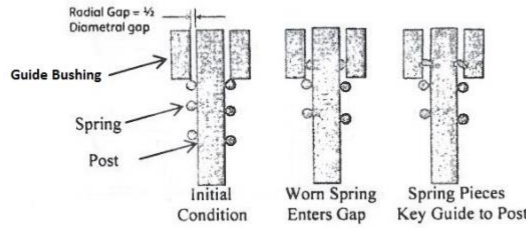
The EMRV solenoid actuator, as seen in Figure 1 below, includes a "floating design" plunger that rests on two guides on two springs that ride along two posts when the valve is operated. When the valve is energized, the solenoid pushes the plunger downward compressing the spring to open the valve. When the solenoid is de-energized the spring decompresses to move the plunger upward to close the valve.

Figure 1. Original Actuator



On June 20, 2014, during refurbishment of EMRVs that were removed from the plant during the 2012 refueling outage, a maintenance technician identified deficiencies with the 'B' and 'D' EMRVs. Specifically, the technician documented that a segment of the spring was lodged between the guide and the post in the EMRV actuator, restricting movement of the spring. As part of the planned EMRV actuator testing and refurbishment activities, the station then conducted bench testing of these actuators on June 26, 2014, and both valves did not stroke satisfactorily. This resulted in two inoperable EMRVs for greater than the Technical Specification allowed outage time of 24 hours for two inoperable EMRVs.

Figure 2. Progression of Wear to Binding



Exelon determined that the failure of the 'B' and 'D' EMRV actuators was due to plunger binding caused by excessive component wear, likely resulting from misalignment between the posts and the guides. This misalignment resulted in an increased gap between the posts and the guides, allowing the spring to enter the gap and wear at a fast rate (see Figure 2 above).

Exelon also concluded the design of the EMRV actuators was inadequate because when they were placed in an environment where the actuator was subject to vibration associated with plant operation, the mechanical tolerance between posts and guides created a condition where the springs could wedge between the guides and the posts jamming the actuator plunger assembly. In addition, Exelon determined that the maintenance procedure for refurbishment of the EMRV actuator did not provide the necessary acceptance criteria for alignment of the posts to guides to ensure that the spring, posts, and guides did not interact in a way that caused excessive wear that allowed the plunger assembly jamming mechanism to exist. Exelon's review of plant operating history concluded that as a result of the design deficiency and inadequate maintenance procedure, the two EMRVs were likely to have been inoperable from July 27, 2012, to October 22, 2012. This conclusion was based on successful actuation of the 'D' EMRV on July 27, 2012. Since the 'D' EMRV exhibited more wear than the 'B' EMRV, it is reasonable to conclude that the 'B' and 'D' EMRVs were operable during the cycle until July 27, 2012.

The inspectors conducted an independent review of this issue and determined that the original design of these valves required that the internal component clearances for the solenoid should be maintained within tight tolerances to limit excessive fretting and wear of the plunger, spring, and guides which can prevent the operation of these components, ultimately affecting the operation of the valves. During valve refurbishment activities completed in accordance with maintenance procedure, 2400-SME-3918.03, step 6.2.25, Oyster Creek maintenance technicians maintain tolerance for these components by bending the bottom bracket as necessary to ensure that the post remains aligned with the guide. The acceptance criterion for this activity, as defined by the procedure step was, "adjust both left and right guide plates (i.e., bottom brackets) and/or posts as required by bending equally until plunger slides smoothly on guides." No specific clearances were specified in the procedure. The inspectors determined that this critical activity was controlled as "skill of the craft." In addition, the inspectors noted that vendor guidance for refurbishment of the EMRV actuator does not provide the necessary acceptance criteria for alignment of the posts to guides to ensure that the springs, posts, and guides do not interact in a way that causes preferential wear of the post allowing the jamming mechanism to exist.

The inspectors reviewed applicable operating experience and determined that there was an opportunity to identify this issue in 2006 when Quad Cities changed the EMRV actuator design due to similar issues. However, the inspectors could not conclude that the issue would have likely been identified during that period since a Part 21 Report was not issued to inform the industry and NRC of the design change and industry operating experience focused on plants that completed or will complete an extended power uprate. As a result, the inspectors concluded that this issue is not reflective of current Exelon performance.

Based on a review of the failure analysis report, the root cause evaluation, and the licensee event report, the inspectors independently verified Exelon's conclusion regarding the inadequate design of the EMRV actuators. In addition, the inspectors concluded that given the original design of the valve, the maintenance refurbishment processes were not adequate to maintain the required internal tolerances to limit excessive fretting and wear of the internal components. Exelon's immediate corrective actions included entering this issue into their corrective action program, and shutting down the unit on July 7, 2014 to test and inspect the then-installed EMRVs. Following successful testing, Exelon installed rebuilt solenoids on all five EMRVs and performed an operability evaluation to demonstrate that these rebuilt actuators would perform their required function until redesigned actuators could be installed during the next refueling outage. Inspectors reviewed the operability evaluation and determined it was acceptable since the degradation of the actuator was progressive over the operating cycle. Exelon redesigned the EMRV actuator to ensure the spring would remain on the outside of the guide bushing, therefore removing the possibility of the spring to enter the guide bushing area and subsequently jamming the actuator causing valve failure. All of the actuators were replaced with the redesigned model during the refueling outage in October 2014. In addition, Exelon issued a Part 21 Report to inform the industry and the NRC of the deficient EMRV actuator design.

Further, the NRC preliminarily determined this issue meets the IMC 0305, Section 11.05a, criteria for treatment as an old design issue. Specifically, the issue was licensee-identified during as-found testing which is not required by NRC regulations, the issue was immediately corrected by the licensee, the issue was not likely to be previously identified during normal operations, routine testing, or maintenance, and the issue is not reflective of current licensee performance. If finalized as such, this finding will not be used as an input in the assessment process or NRC Action Matrix. Further details are discussed in Attachment 1.

Analysis: The inspectors determined that this issue is a performance deficiency because Exelon did not establish adequate measures per 10 CFR 50, Appendix B, Criterion III, "Design Control," for selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the EMRVs. The inspectors reviewed IMC 0612, Appendix B, "Issue Screening," and determined that the finding is more than minor because it adversely affected the design control quality attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the design deficiency of the EMRVs and the inadequate maintenance process led to the inability of the 'B' and 'D' EMRVs to perform their safety function.

The inspectors screened this issue for safety significance in accordance with IMC 0609, Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” and determined a detailed risk evaluation was required because the EMRVs were potentially failed or unreliable for greater than the Technical Specification allowed outage time. As described in Attachment 3 to this report, a detailed risk evaluation concluded that the increase in CDF related to failure of the ‘B’ and ‘D’ EMRVs is in the mid E-5 range; therefore, this finding was preliminarily determined to have a substantial safety significance (Yellow). Due to the nature of the failures, no recovery credit was assigned. The dominant sequences included loss of main feedwater with failures of the isolation condensers, and failure to depressurize. This finding does not present an immediate safety concern because Exelon replaced all of the actuators with redesigned actuators during the refueling outage in October 2014.

The inspectors determined that this finding did not have a cross-cutting aspect because the most significant contributor of the performance deficiency was not reflective of current licensee performance. Specifically, the inspectors determined that the performance deficiency existed since original installation of the EMRVs and that an opportunity to identify this issue following original installation was in 2006 when Quad Cities changed the EMRV actuator design due to similar issues.

Enforcement: 10 CFR 50, Appendix B, Criterion III, “Design Control,” states in part, “Measures shall be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.” Technical Specification 3.4.B states, in part, five electromatic relief valves shall be operable and if more than one are inoperable, then reactor pressure shall be reduced to 110 psig or less, within 24 hours.

Contrary to the above, since original installation of the EMRVs in 1969, until the valves were redesigned and reinstalled during the 2014 refueling outage, Exelon did not establish adequate measures for the suitability of applications of materials and processes (maintenance) for the EMRV solenoid-operated actuators. Specifically, the original design of the EMRV actuators was inadequate because when they were placed in an environment where the actuator was subject to vibration associated with plant operation, the mechanical tolerance between posts and guides created a condition where the springs could wedge between the guides and the posts, jamming the actuator plunger assembly. In addition, given the original design of the valve, the maintenance refurbishing processes were not adequate to maintain the required internal tolerances to prevent excessive fretting and wear of the internal components. Between July 27, 2012 and October 22, 2012, this inadequate design resulted in two EMRVs being inoperable for greater than the Technical Specification allowed outage time of 24 hours. Exelon’s immediate corrective actions for this issue included placing this issue into the corrective action program as issue report 1679428 and redesigning the EMRV actuators to ensure the spring is on the outside of the guide bushing, therefore removing the possibility of the spring to enter the guide bushing area and subsequently jamming the actuator causing valve failure. All of the actuators were replaced with the redesigned actuators during the refueling outage in October 2014. In addition, Exelon issued a 10 CFR Part 21 report to inform the industry of the deficient EMRV actuator design. This issue is being characterized as an apparent violation in accordance with the NRC’s Enforcement Policy, and its final significance will be dispositioned in separate future correspondence. **(AV 05000219/2014009-01, Inadequate Application of Materials, Parts, Equipment, and Processes Associated with the Electromatic Relief Valves)**

c. Observations

The inspectors noted that while there may have been two potential opportunities for identifying the deficient EMRV design and associated maintenance process, earlier identification was not reasonably likely given the available information and the historical good performance of the valves. The inspectors reviewed completed work orders from 1996 to 2012 and noted that a majority of the work orders (years 1996, 2000, 2004, 2006, 2008, and 2010) documented that EMRV internal components were replaced due to wear. However, no issue reports were written documenting these deficiencies. Inspectors identified a second missed opportunity during review of industry operating experience. The inspectors determined that in August 2006, Exelon initiated corporate issue report 515977 to document failures of the main steam EMRVs at Quad Cities. The issue report stated, in part, that work and event histories associated with the EMRVs on each unit at Quad Cities over many years indicated that vibration of the main steam lines, coupled with imprecise rebuild procedures and installation tolerances, affected the performance and operating life of the valves. The valve failures at Quad Cities were in large part due to increased vibration from an extended power uprate, whereas Oyster Creek did not have an extended power uprate, and the Oyster Creek EMRVs had operated for over 40 years without failure. This issue report also stated that the EMRV actuator and pilot valve rebuild procedures at Quad Cities were revised to improve rebuild guidance, and that the EMRV solenoid actuators were redesigned and replaced on both units. However, the Exelon Fleet Operating Experience Coordinator did not assign a site formal review to Oyster Creek for this issue.

Notwithstanding, a Part 21 Report was not issued to inform the industry and the NRC of the EMRV actuator design deficiency, and the industry operating experience focused on plants that completed or will complete an extended power uprate. Also, although wear has been previously identified during refurbishment activities, there is no record of EMRV actuators failing as-found stroke testing prior to the June 20, 2014 failures. Because the inspectors did not directly attribute these issues to the inoperability of the EMRVs, the inspectors determined that failure to document component wear in the corrective action program and failure to assign a formal operating experience review were of minor significance and not subject to enforcement action in accordance with the NRC's Enforcement Policy. Exelon documented this issue in issue report 2413924.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153 – 1 sample)

(Closed) Licensee Event Report (LER) 05000219/2014-002-00: Technical Specification Prohibited Condition Caused by Two Electromatic Relief Valves Inoperable for Greater than Allowed Outage Time

On June 26, 2014, during as-found testing of the EMRV actuators removed from the plant that were in the plant from October 2010 to October 2012, two ('B' and 'D') of the five EMRV actuators failed to operate. August 25, 2014, Exelon completed a root cause evaluation and determined the most probable time of failure of the two EMRVs was between July 27, 2012 and October 22, 2012, approximately 87 days. July 27, 2012 was the last known date that the 'D' EMRV functioned and October 22, 2012 was the beginning of the refueling outage which replaced these EMRV actuators with refurbished actuators. Based on these dates, Exelon concluded that two of the five EMRVs were inoperable for longer than the Technical Specification Allowed outage time of 24 hours.

The enforcement aspects of this issue are discussed in Section 4OA2. The inspectors did not identify any new issues during the review of the licensee event report. This licensee event report is closed.

4OA6 Meetings, Including Exit

On December 16, 2014, the inspectors presented the inspection results to Mr. G. Stathes, Site Vice President, and other members of the Oyster Creek Nuclear Generating Station staff. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

On January 29, 2014, the inspectors presented changes to the characterization of the inspection finding to Mr. G. Stathes, Site Vice President, and other members of the Oyster Creek Nuclear Generating Station staff. Changes in the characterization of the finding resulted from additional information provided after the original exit meeting which affected the risk analysis. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

ATTACHMENT 1: Evaluation of IMC 0305 Criteria for Treatment of Old Design Issues
ATTACHMENT 2: Supplementary Information
ATTACHMENT 3: Detailed Risk Significance Evaluation

<u>Evaluation of IMC 0305 Criteria for Treatment of Old Design Issues</u>

The NRC staff evaluated this issue to determine if it met the definition of an "Old Design Issue," per IMC 0305, Section 11.05a. The inspectors preliminarily determined that this issue met the definition of an old design issue, as discussed below.

a. <u>The finding was licensee-identified as a result of a voluntary initiative, such as a design basis reconstitution</u>.

   On June 20, 2014, the 'B' and 'D' EMRV actuators failed to operate during as-found testing. Further inspection of these actuators found unexpected wear of the posts, springs, and guides. On June 26, 2014, the 'B' and 'D' EMRV actuators failed bench testing that Exelon conducted as part of refurbishment activities. Failure analysis was performed and could not identify a definitive cause of the failure. However, analysis determined the damage observed on the components of all actuators removed from the plant in 2012 was consistent with fretting due to in-service vibration.

   The inspectors reviewed technical specifications, design basis documents, vendor manuals, and other information and determined that the only required testing of the EMRVs is as-left testing of the refurbished EMRV actuators once they are installed in the plant. Additionally, as-found stroke testing (i.e. bench testing) of the EMRV actuators at the end of the operating cycle is not required. Because there are no regulations or industry standards that require as-found testing of the EMRV actuators, the inspectors concluded that this issue was licensee-identified as a result of a voluntary initiative.

   The NRC staff determined that credit for this criterion was appropriate.

b. <u>The finding was or will be corrected, including immediate corrective actions and long term comprehensive corrective actions to prevent recurrence within a reasonable time following identification</u>.

   Exelon's corrective actions for this issue included redesigning the EMRV actuators to ensure the spring is on the outside of the guide bushing, therefore removing the possibility of the spring to enter the guide bushing area and subsequently jamming the actuator, causing valve failure. The EMRVs currently installed in the plant all have this revised design. Exelon also revised procedures for EMRV solenoid operator removal, refurbishment, installation to strengthen preventative maintenance quality (component wear and acceptance criteria, component adjustment tolerances, and transportation guidance).

   The NRC staff determined that credit for this criterion was appropriate.

c. <u>The finding was not likely to be previously identified by recent ongoing licensee efforts, such as normal surveillance, quality assurance activities, or evaluation of industry information</u>.

   The inspectors determined that the EMRV design deficiency was not likely to be previously identified by recent ongoing licensee efforts. The only required surveillance testing per Technical Specifications is as-left testing of the refurbished EMRV actuators once they are installed in the plant. Although wear has been routinely identified during refurbishment activities, there is no record of EMRV actuators failing as-found stroke testing in over 40 years of Oyster Creek's operating history prior to the June 20, 2014 failures. Exelon's procedures only required further review of EMRV actuator performance if the actuator failed as-found testing. The inspectors noted that ongoing licensee efforts (as-found testing, refurbishment, and as-left testing) had been consistent for several years and the design deficiency was only identified when a failure during bench-testing occurred.

Further, there is no established quality assurance activity that would have identified this issue, and the inspectors did not identify any recent operating experience items that addressed similar problems with EMRVs that would have prompted Exelon to review their design. The inspectors did identify operating experience from 2006 related to EMRV failures at Quad Cities that resulted in an EMRV actuator design change. However, a Part 21 Report was not issued for the design change, and the operating experience issued by Quad Cities focused on plants that had completed or planned to complete an extended power uprate, which could lead to excessive vibration. Because a Part 21 Report was not issued for the 2006 event, and Oyster Creek had not completed an extended power uprate, the inspectors determined that a review of the operating experience prior to an actual failure would not have likely identified the design deficiency.

The NRC staff determined that credit for this criterion was appropriate.

d. <u>The finding does not reflect a current performance deficiency associated with existing licensee programs, policy, or procedure</u>.

The inspectors determined that the performance deficiency existed since original installation of the EMRVs and that the best opportunity to identify this issue was in 2006, when Quad Cities changed the EMRV actuator design due to similar issues. As described above, because the operating experience was presented to the industry as being applicable to plants completing power uprates, the inspectors determined that this issue does not represent a current performance deficiency associated with existing licensee programs, policy, or procedure.

The NRC staff determined that credit for this criterion is appropriate.

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Exelon Personnel
G. Stathes, Site Vice-President
J. Dostal, Plant Manager
D. Chernesky, Maintenance Director
J. Chrisley, Regulatory Assurance Specialist
R. Fitts, Senior Regulatory Specialist, Corrective Action Program Manager
M. Ford, Director, Operations
G. Malone, Director, Engineering
M. McKenna, Manager, Regulatory Assurance
K. Paez, Regulatory Assurance Specialist
H. Ray, Senior Manager, Design Engineering
S. Schwartz, System Manager

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened

05000219/2014009-01	AV	Inadequate Application of Materials, Parts, Equipment, and Processes Associated with the Electromatic Relief Valves

Closed

05000219/2014-02-00	LER	Technical Specification Prohibited Condition Caused by Two Electromatic Relief Valves Inoperable for Greater than Allowed Outage Time

LIST OF DOCUMENTS REVIEWED

Section 4OA2: Problem Identification and Resolution

Condition Reports *(* indicates that condition report was generated as a result of this inspection)*

1100873	0520889	0567038	0714331	0808422	0808416
0842613	0967389	0843338	0842614	1162972	1194456
1235185	1673665	1678550	1679879	1679428	1686134
1391942	2413924*				

Drawings
214564, 6 in 1525VX Electromatic Relief Valve, Revision 3
3NC117, Consolidated 1525VX Electromatic Relief Valve, Revision 8

Miscellaneous
Exelon Risk Management Memorandum C467140003.160-11921, from L. Lee to A. Bready, “Oyster Creek EMRV PRA Success Criteria to Support SDP Risk Evaluation,”

Revision 2, November 26, 2014

Exelon Risk Management Memorandum C467140003.160-12060, from L. Lee to A. Bready, "Review of Top Cutset in NRC SPAR Model for OC EMRV SDP Risk Evaluation," November 21, 2014

Exelon Risk Management Memorandum, from D. Bidwell and G. Parry to G. Krueger, "Evaluation of Oyster Creek 2014 EMRV As-Found Failures for Common Cause," November 14, 2014

Exelon Risk Management Memorandum H0467140003-3149, from D. Bidwell to L. Lee, "Development of Oyster Creek EMRV Common Cause Factors," October 27, 2014

Failed as Found Testing of B and D Electromatic Relief Valve Actuators, IR 01679428 EMRV Root Cause Report, dated August 15, 2014

OC-SDP-001, Significance Determination for the EMRV B and D Failures During As-Found Testing on 6/20/2014 Due to Vibration Inducted Failure of the EMRV Actuators, Revision 5

Procedures

1000-ADM-7216.01, Corrective Action Process, Revision 3
1000-ADM-7216.01, Corrective Action Process, Revision 6
LS-OC-125, Corrective Action Program Procedures, Revision 2
LS-AA-125-1006, CAP Process Expectation Manual, Revision 0
LS-AA-120, Issue Identification and Screening Process, Revision 1
602.3.014, EMRV Pressure Sensor/Pilot Valve Control Relay Test and Calibration, Revision 7
2400-SME-3918.03, EMRV Solenoid Operation Removal, Refurbishment, and Installation, Revision 3 – to Revision 20

Work Orders

R2174290	R2173443	C2032482	R2129857	R2129831	R2129860
R2129858	R2129861	R2092646	R2097558	R2092648	R2092647
R2097559	R2058256	R2058257	R2120639	R2059919	R2120641
R0808396	R0808398	R0808395	R0808399	R0808391	R0808390
R0502818	JO 546724	JO 543511	JO 543513	JO 522470	JO 522471
JO 522472	JO 522473	JO 522474	JO 50614	JO 506138	JO 506139
JO 506142	JO 502818				

LIST OF ACRONYMS

ADAMS	Agency-wide Documents Access and Management System
ATWS	Anticipated Transient Without a Scram
AV	Apparent Violation
CCF	Common Cause Failure
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
ΔCDF	delta Core Damage Frequency
EDG	Emergency Diesel Generator
EMRV	Electromatic Relief Valve
FHA	Fire Hazards Analysis
IMC	Inspection Manual Chapter
IPEEE	Individual Plant Examination External Events
LER	Licensee Event Report
LERF	Large Early Release Frequency
MGL	Multiple Greek Letter
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
RASP	Risk Assessment Standardization Project
SAPHIRE	Systems Analysis Programs for Hands-On Evaluation
SRA	Senior Risk Analyst
SPAR	Standardized Plant Analysis Risk

Detailed Risk Significance Evaluation

1. Initial Screening

   The inspectors reviewed IMC 0612, Appendix B, "Issue Screening," and determined that the issue was more than minor because it adversely affected the design control quality attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors screened for safety significance using IMC 0609, Appendix A, "Mitigating Systems Screening Questions," and determined a detailed risk evaluation was required, because the EMRVs were potentially failed or unreliable for greater than the Technical Specification allowed outage time.

2. Detailed Risk Evaluation

   The Senior Reactor Analyst (SRA) used the Systems Analysis Programs for Hands-On Evaluation (SAPHIRE), Revision 8.1.0, and the Standardized Plant Analysis Risk (SPAR) Model for Oyster Creek, Model Version 8.22, to conduct the internal events detailed risk evaluation and the licensee's Individual Plant Evaluation External Events (IPEEE) and fire hazards analysis (FHA) to assess the external events risk contribution for this issue. A review of plant operating history concluded that the two EMRVs were potentially inoperable from July 27 to October 22, 2012. This conclusion was based on successful actuation of the 'D' EMRV on July 27, 2012. Since the 'D' EMRV exhibited more wear than the 'B' EMRV, it is reasonable to conclude that the 'B' and 'D' EMRVs were operable during the cycle until July 27, 2012. In accordance with Risk Assessment Standardization Project (RASP), Volume 1, Rev. 2, section 2.4, an exposure period of t/2 was assigned resulting in an analyzed exposure period of 44 days. Due to the nature of the failures, no operator recovery credit was assigned.

3. Internal Events Contribution

   The SRA made the following changes to the Oyster Creek SPAR model:

   - The SPAR Model default setting for ADS valve testing assumes a staggered testing regime. Since the Oyster Creek EMRVs are all tested as a group (essentially at the same time), the ADS valves common cause failure basic event was changed to non-staggered testing.
   - In accordance with RASP Volume 1, Rev. 2, sections 3.2 and 5.1, ADV-SRV-CC-108B and ADV-SRV-CC-108D were set to TRUE.
   - Based on a review of thermo-hydraulic analysis completed by the licensee and a comparison to other similar designs, the success criteria for depressurization was modified to allow for successful depressurization with 2 out of 5 EMRVs for non-ATWS sequences.
   - The initiating event frequency for loss of feedwater control was modified from 1.69E-1 to 7.54E-2, to more realistically represent the event. This revised event frequency is considered more representative of the population of similar boiling water reactors.
   - Basic Event ISO-ICWATERHMR, for failure of the isolation condenser due to a water hammer event, was modified from a base failure value of 0.5 to 0.1 to more realistically represent the event.

- Errors were identified in the 2012 Common Cause Failure (CCF) Parameter Estimations for BWR Safety Relief Valve of a size grouping of 5. As a result, the values were recalculated. The results are as follows:

	mle	a	b	mean	median	5 pct	95 pct
1	8.38E-01	1.10E+02	6.14E+00	9.47E-01	9.50E-01	9.09E-01	9.76E-01
2	1.22E-01	4.08E+00	1.12E+02	3.52E-02	3.26E-02	1.24E-02	6.71E-02
3	2.74E-02	1.42E+00	1.14E+02	1.22E-02	9.57E-03	1.32E-03	3.23E-02
4	1.04E-02	5.41E-01	1.15E+02	4.68E-03	2.29E-03	2.77E-05	1.74E-02
5	1.97E-03	1.04E-01	1.16E+02	8.97E-04	6.69E-06	1.57E-15	5.21E-03

These were applied to basic events ZA-ADS-SRV-CC-05A01 through ZA-ADS-SRV-CC-05A05, respectively.

- Based on the significant contribution from common cause failure, the NRC Office of Research independently reviewed the adequacy of the CCF calculation generated from SAPHIRE. As a result of this review, it was determined that the most accurate failure mode for the ADS-SRV-CF-VALVS basic event is to change from the default C (compound event) to R (common cause failure) method. This change better represents the most up-to-date common cause modeling technique, referenced in the RASP guidance. Additionally, the independent failure was updated to the 2010 data set.

Based upon these changes to the model, the delta core damage frequency (ΔCDF) is 1.2E-5/year. The increase in CDF was heavily influenced by the increase in the common cause failure probability of the EMRVs.

The dominant sequences included loss of main feedwater with failures of the isolation condensers, and failure to depressurize.

4. <u>External Events Contribution</u>

To evaluate the external events contribution for this issue, the SRA reviewed the Oyster Creek IPEEE, the FHA, and the licensee fire probabilistic risk assessment (PRA) insights. From these the SRA determined that the importance of the EMRVs in mitigation of fire events is significantly higher than seismic and other external event hazards (i.e., tornadoes, high winds, external flooding, transportation, and plane crashes). Consequently, the external event risk contributions to events other than fire were determined to be only a minor contributor.

Oyster Creek credits two methods to achieve safe hot shutdown from a fire. The majority of the areas credit the isolation condenser path. The remainder of the scenarios credits the EMRV path. According to Oyster Creek's fire protection document, Specification SP-1302-06-013, Attachment E, the following areas are specifically credited for the ERMVs; RB-FZ-1A, RB-FZ-1B, RBFZ-1C, DG-FA-17 and MT-FA-12. In the event of a fire, if the normal preferred systems are available, Oyster Creek procedures allow the use of these systems. As such the importance of the EMRV's can be reflected in other areas, such as the turbine building, where random failure of the credited system can occur.

Fires areas RB-FZ-1A, RB-FZ-1B, and RBFZ-1C have low combustible loading. In addition, for postulated fires in these areas, offsite power is available to the 4Kv buses via the startup transformer. Additionally, IPEEE Table 4.6-1 shows that the CDF contribution from fires is approximately two orders of magnitude lower than for fires in MT-FA-12. As a result these areas were screened as minor contributors.

Fire area DG-FA-17, is the No. 2 emergency diesel generator (EDG) room. Cables and equipment in this room are associated with train B power systems. For postulated fires in these areas, offsite power is available to the 4Kv buses via the startup transformer. The base case condition was modeled as a transient with ISO A and B failed (probability =1), EDG 2 failed (probability =1), 4140 Vac Bus 1D failed (probability =1) and the initiating event frequency (IE-TRANS) set to 1.5E-2 (EDG fire frequency from IPEEE Table 4.1-11). The conditional case included the failures of B and D EMRV (TRUE). The ΔCDF was 3E-8.

Fire area MT-FA-12, which includes the main transformer, is a large contributor to the overall fire risk. The IPEEE initiating event frequency for fires in this area is 2.14E-2. Transformer fires tend to be large and have the potential impact of disabling offsite power. In addition, fires in this area can challenge the availability of the isolation condensers. For this issue postulated fires in this area were determined to significantly contribute to the risk.

Based on insights from the IPEEE and the licensee fire PRA, fires in the turbine building and the transformer area have the highest risk associated with this issue. Specifically:

<u>Scenario TR-FA-3B_E1</u>

The dominant cutset for a fire in this area results in a failure of the feedwater system, failure to make up to the isolation condenser and a failure to depressurize. The estimated change in CDF due to fire in this area is approximately 3E-5.

<u>Scenario MT-FA-12_H</u>

The dominant cutset for a fire in this area results in a failure of the feedwater system, failure of the isolation condenser and a failure to depressurize. The estimated change in CDF due to fire in this area is approximately 5E-6.

5. <u>Overall Core Damage Risk Significance</u>

The total change in CDF associated with the failure of the ‘B’ and ‘D’ EMRVs for the t/2 for the period between July 27, 2012, and October 22, 2012 is the sum of internal and external conditional core damage frequencies. Therefore, the estimated change in CDF is 1.2E-5 + 5E-6 + 3E-5 = 4.8E-5/year, which is of preliminary Yellow or substantial safety significance.

6. <u>Large Early Release Frequency (LERF)</u>

For issues involving an increase in CDF > 1E-7, IMC 0609 requires an evaluation of LERF using the guidance of NUREG-1765, “Basis Document for Large Early Release Frequency Significance Determination Process.” The failure of the EMRVs would be considers a Type A finding. NUREG-1765, Table 2, assigns a LERF factor of 1.0 for high-pressure sequences with a dry drywell, and 0.6 for high-pressure sequences with a flooded drywell. The former value is bounding, but not necessarily conservative, in that liner melt-through is expected to occur shortly after vessel failure if the drywell is dry. The latter value is affected by the type and size of reactor coolant system rupture, operator actions following the onset of core damage, and phenomenological issues related to direct containment heating and fuel-coolant interactions.

The licensee's model explicitly estimates LERF, and considers relevant high-pressure vessel breach phenomena (namely, fuel-coolant interaction, liner-melt-through, and direct containment heating). The multiplier for converting CDF to LERF according to the licensee is ~ 2E-2.

As noted by the licensee, recent evaluations (e.g., State of the Art Reactor Consequence Analysis at Peach Bottom) have indicated that the likelihood of severe accident-induced main steam line creep rupture or a stuck-open relief valve prior to vessel breach is potentially higher than typically estimated in PRAs. This same case was made in a 2003 report prepared for NRC Office of Research by Energy Research, Inc.[1] These failure modes would lead to a more benign containment response at the time of vessel breach, in terms of direct containment heating and fuel-coolant interaction-induced containment failure. Consequently, the SRA reasoned that the licensee's 2E-2 LERF multiplier was a more reasonable and appropriate value, resulting in a LERF value of 9.6E-7/year. This LERF value represents a low to moderate safety significance, or White issue. In accordance with IMC 0609, the higher of the two risk metric values is used to assign the preliminary safety significance of this issue. Therefore, this issue is assigned a preliminary Yellow safety significance based upon the calculated increase in CDF.

7. <u>Sensitivity</u>

The following factors and assumptions were found to contribute most to the sensitivity of the analysis:

- Common Cause Failure – As discussed previously and in the following licensee risk insights paragraph, the conditional EMRV common cause failure probability significantly influences the final risk estimate, resulting in a potential order of magnitude difference.
- Exposure Time – The approach used assumed that both 'B' and 'D' EMRV had the same exposure based on the 'D' EMRV successfully operating for 1 cycle 88 days prior to the outage. Given the nature of the failure, a t/2 exposure was assigned per the RASP guidance. A change in exposure period would have a proportional change on the outcome.
- Isolation Condenser Water Hammer – Both the licensee's and SPAR models lacked a detailed basis for the baseline failure probability. Additional evaluation by the SRA assumed a lower value of 0.1. Application of a 0.5 failure would approximately double the overall outcome.
- Isolation Condenser Cycling - As shown during the loss of offsite power event of July 12, 2009 (Oyster Creek Inspection Report 05000219/2009009 and ASP Analysis for LER 219/09-005, ML101600186), the isolation condensers can undergo significant cycling in order to control cooldown rates and procedurally to prevent flooding of the isolation condenser. This could result in a higher cumulative demand failure likelihood of valves ISO-MOV-CC-V-1434 and 1435, which impact the success of the isolation condensers. Current modeling techniques are not developed to model this potential increase for significance determination process degraded condition assessments.

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[1] Esmaili et al., "The Probability of High-Pressure Melt Ejection-Induced Direct Containment Heating Failure in Boiling Water Reactors with Mark I Design," ERI/NRC-03-204, November 2003.

8. Oyster Creeks PRA Staff Risk Insights and Considerations

Discussions with the Oyster Creek PRA staff identified risk insights and assumptions that significantly influence the conditional core damage frequency estimates. The licensee provided evaluation OC-SDP-001, Revision 5, which estimates the change in CDF for internal and external events to be 8.9E-6 per year. Oyster Creek believes that the method for calculating CCF over predicts the contribution from common cause failure. The licensee calculated a CCF of the EMRV equal to 0.3. This was done with the multiple Greek Letter (MGL) method using the maximum likelihood estimation values. The NRC SPAR model uses the Alpha Factor method. Using the mean Alpha Factor values for the same conditions, the NRC model calculates a CCF equal to 0.35. Both the Alpha Factor and MGL are acceptable methods for calculating CCF. However, the licensee chose to use a value of 0.1 in their model because "increasing the CCF probability to 0.3 for this basic event would have a significant impact on the risk." Since most of the dominant sequences include this basic event, applying the appropriate value of 0.3 would essentially triple their results from about 9E-6 to about 2.7E-5, or White to Yellow. Using the available data Exelon calculated an EMRV CCF of 0.3, but the NRC determined that Exelon did not provide an adequate technical basis for changing the CCF to 0.1. As discussed above, the NRC used 0.1 as a more realistic isolation condenser failure probability.

Additionally, failure modes of the isolation condensers due to water hammer are not well understood and may be overestimated. The licensee base model assumed a failure probability of the isolation condenser due to water hammer on loss of level control to be 0.5. Due to the lack of technical information for this event, the licensee conducted additional analysis on the isolation condenser and assumed a water hammer failure probability of 0.01. Although a failure probability of 0.5 is likely high, operating experience does show that significant damage to the isolation condenser could occur due to water hammer. For example, Dresden 2, experienced an isolation condenser water hammer event in 2002 (Inspection Report 50-249/02-03) which that resulted in piping support and heat exchanger pass plate damage. Additionally, NUREG 0927, Evaluation of Water Hammer Events in Nuclear Power Plants," published in 1984, identify three other isolation condenser water hammer events caused by high level. Given this event, the licensee's assumption of a water hammer failure probability of 0.01 may underestimate the probability of an isolation condenser failure.

Oyster Creek performed additional internal and external model modifications that resulted in a lesser impact to the overall results. Additionally, they apply a "Plant Availability Factor" of 93.8%, which serves to reduce the overall CDF by about 6%. This is not consistent with the NRC's guidance for performing an SDP risk assessment. The dominant cutset was a loss of feedwater, failure to depressurize the reactor due CCF of the EMRVs and failure to control isolation condenser return valves. This logic is generally consistent with the independent assessment done by the NRC. With this data and an evaluation of external events, the licensee concluded that the issue was White. As noted above, application of the calculated CCF failure probability of 0.3 would significantly impact the results presented by the licensee, and result in a Yellow finding.

9. Overall Risk Significance

Based on the tools and methods available and consideration of Exelon's risk insights, the NRC concluded that the failure of two EMRVs during the past operating cycle represents a preliminary Yellow (substantial safety significance) issue, based on the calculated ΔCDF.