UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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LISLE, IL 60532-4352

June 16, 2016

Mr. David Hamilton
Site Vice President
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P. O. Box 97, 10 Center Road, A–PY–A290
Perry, OH 44081–0097

SUBJECT: PERRY NUCLEAR POWER PLANT - EVALUATIONS OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000440/2016007

Dear Mr. Hamilton:

On May 27, 2016*,* the U.S. Nuclear Regulatory Commission (NRC) completed an Evaluations of Changes, Tests, and Experiments, and Permanent Plant Modifications inspection at your Perry Nuclear Power Plant. The enclosed inspection report documents the inspection results, which were discussed on May 27, 2016, with you and other members of your staff.

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

Three NRC-identified finding of very-low safety significance (Green) were identified during this inspection. The findings were determined to involve a violation of NRC requirements. However, because of their very-low safety significance, and because the issues was entered into your Corrective Action Program, the NRC is treating the issues as Non-Cited Violations in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the subject or severity to any of these Non-Cited-Violations, 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 copies to the Regional Administrator, Region III; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Perry Nuclear Power Plant.

In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Perry Nuclear Power Plant

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, “Public Inspections, Exemptions, Requests for Withholding,” 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’s Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA Mark Jeffers Acting for/

Robert C. Daley, Chief
Engineering Branch 3
Division of Reactor Safety

Docket No. 50–440
License No. NPF–58

Enclosure:
IR 05000440/2016007

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	05000440/2016007
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Perry Nuclear Power Plant
Location:	Perry, OH
Dates:	May 9 – 27, 2016
Inspectors:	A. Shaikh, Senior Reactor Inspector (Lead) I. Khan, Reactor Inspector J. Hafeez, Reactor Inspector M. Domke, Reactor Inspector
Approved by:	R. Daley, Chief Engineering Branch 3 Division of Reactor Safety

SUMMARY

Inspection Report 05000440/2016007; 05/9/2016 - 05/27/2016; Perry Nuclear Power Plant; Evaluations of Changes, Tests, and Experiments and Permanent Plant Modifications.

This report covers a 2-week announced baseline inspection on evaluations of changes, tests, and experiments, and permanent plant modifications. The inspection was conducted by Region III based engineering inspectors. Three findings of very-low safety significance were identified by the inspectors. Each violation was considered a Non-Cited Violation (NCV) of U.S. Nuclear Regulatory Commission (NRC) regulations. The significance of most findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process (SDP)". Cross-cutting aspects were determined using IMC 0310, "Aspects within the Cross-Cutting Areas." Findings and/or violations for which the SDP does not apply may be Green, or be assigned a severity level after NRC management review. 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, dated February 2014.

<u>NRC-Identified and Self-Revealed Findings</u>

Cornerstone: Mitigating Systems

<u>Severity Level IV</u>: The inspectors identified a Severity Level IV, NCV of Title 10 of the *Code of Federal Regulations* (CFR), Part 50.59, "Changes, Tests, and Experiments," having very-low safety significance (Green) for failure to document the basis for performing a plant modification where a manual operator action was replaced with an automatic action. Specifically, the licensee did not evaluate whether adding a safety-related function to a nonsafety-related component was within the licensing basis of the facility.

The inspectors determined that the failure to perform a 10 CFR 50.59 evaluation for Plant Modification 11-0794 was contrary to 10 CFR 50.59(d)(1) and was a performance deficiency. The performance deficiency was determined to be more-than-minor and a finding, because the finding impacted mitigating systems cornerstone attribute of Design Control and adversely affected the Cornerstone Objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, this plan modification added a Safety-Related function to a Nonsafety-Related component and, therefore, impacted the availability, reliability, and capability of the Safety-Related Battery Room ventilation system and the Safety-Related Motor Control Center, Switchgear, and Miscellaneous Electrical Equipment Area ventilation system. In addition, the associated violation 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 inspectors determined that finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process." Using Attachment 0609.04, "Initial Characterization of Findings," Table 2 the inspectors determined that the finding affected the Mitigating Systems cornerstone. As a result, the inspectors evaluated the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 2, for the Mitigating Systems cornerstone. The inspectors answered "No" to question A.4 in Exhibit 2 – Mitigating System Screening Questions. Specifically, the inspectors determined the finding did not represent an actual loss of the Battery Room ventilation system or Motor Control Center, Switchgear, and Miscellaneous Electrical Equipment Area ventilation system because the automatic action had not been implemented at the time of the finding. Therefore, the inspectors determined the significance of this finding to be of very-low safety significance (Green). In accordance with Section 6.1.d of the NRC Enforcement Policy this violation is categorized as Severity Level IV because the resulting changes were evaluated by the SDP as having very-low safety significance (i.e., green finding). The inspectors determined the finding was associated with the cross-cutting aspect of Procedure Adherence in the area of Human Performance, because the licensee failed to follow the screening criteria in Attachment 2 of Procedure NOBP-LP-4003A, FENOC 10 CFR 50.59 User Guidelines. [H.8]

<u>Severity Level IV</u>: The inspectors identified a Severity Level IV, NCV of 10 CFR 50.59, “Changes, Tests, and Experiments,” having very-low safety significance (Green) for the licensee’s failure to conclude that site flooding modifications and associated analysis included a standard that resulted in a departure from the method of evaluation as described in the Updated Final Safety Analysis Report. Specifically, the licensee used a new method for evaluation of design basis flooding at Perry Nuclear Power Plant that is different from the method described in the Updated Final Safety Analysis Report and not approved by the NRC.

The inspectors determined that the licensee’s use of an unapproved methodology for site flooding modifications and associated analysis that constituted a departure from a method of evaluation was contrary to 10 CFR 50.59(c)(2)(8) and was a performance deficiency. Specifically, the licensee used a new method for evaluation of design basis flooding at Perry Nuclear Power Plant that is different from the method described in the Updated Final Safety Analysis Report and not approved by the NRC. The performance deficiency was determined to be more-than-minor, and a finding, because it affected the cornerstone attribute of protection against external factors and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. In addition, the associated violation was determined to be more-than-minor because the inspectors determined that there was a reasonable likelihood that the changes would have required prior NRC approval. The inspectors determined that finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process”. Using Attachment 0609.04, “Initial Characterization of Findings,” Table 2 the inspectors determined that the finding affected the Mitigating Systems cornerstone. As a result, the inspectors evaluated the finding using Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” Exhibit 2, for the Mitigating Systems cornerstone. The inspectors answered “Yes” to question A.1 in Exhibit 2 – Mitigating Systems Screening Questions. Specifically, the inspectors determined the finding did not result in systems, structures, and components not being able to maintain their operability or functionality. Therefore, the inspectors determined the significance of this finding to be of very-low safety significance (Green). In accordance with Section 6.1.d of the NRC Enforcement Policy this violation is categorized as Severity Level IV because the resulting changes were evaluated by the SDP as having very-low safety significance (i.e., green finding). The inspectors determined that this finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Problem Identification, for the licensee’s failure to identify issues completely, accurately, and in a timely manner. Specifically, the licensee’s 50.59 review committee failed to accurately identify the methodology change concern in Evaluation 14-01234 during a review documented in CR2015-14025. [P.1]

Cornerstone: Initiating Events

Green: A finding of very-low safety significance (Green) and associated NCV of 10 CFR 50.55a(g)(4) was identified by the inspectors for the licensee's failure to maintain the American Society of Mechanical Engineers (ASME) Code Class 1 component in accordance with ASME Code Section XI requirements. Specifically, the licensee failed to measure and document the method of measuring the cavity created after removal of indications on the reactor water clean-up line prior to return to service.

The inspectors determined that the licensee's failure to maintain the ASME Code Class 1 component in accordance with ASME Code Section XI requirements was a performance deficiency. This performance deficiency was found to be more-than-minor, and a finding, because the performance deficiency, if left uncorrected could become a more significant safety concern. Specifically, absent NRC identification, the licensee would not have questioned the potential challenge to component functionality since the cavity measurements were not performed. This condition could potentially lead to the failure of the reactor water clean-up bottom head drain, which in turn, could lead to a potential loss of reactor coolant. The inspectors reviewed the finding using Attachment 0609.04, "Initial Characterization of Findings," Table 3 – SDP Appendix Router. The inspectors answered 'No' to the question in Section A of Table 3 and therefore the finding was evaluated using the SDP in accordance with IMC 0609, "The Significance Determination Process (SDP) for At-Power Operations," Appendix A, Exhibit 1, "Initiating Events Screening Questions". The inspectors answered "No" to the questions in Exhibit 1 and determined this finding to have a very-low safety significance (Green). The inspectors determined that this finding has a cross-cutting aspect in the area of Human Performance, Design Margin, for the licensee's failure to maintain equipment within design margins. Specifically, the licensee staff failed to ensure that metal removal performed on an ASME Code Class 1 component did not result in a condition where the minimum design wall thickness of the component was compromised, and therefore, failed to ensure design margin was maintained. [H.6]

Licensee-Identified Violations

None

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

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

.1 Evaluation of Changes, Tests, and Experiments

a. Inspection Scope

The inspectors reviewed 5 safety evaluations performed pursuant to Title 10 of the *Code of Federal Regulations* (CFR), Part 50.59 to determine if the evaluation was adequate and that prior U.S. Nuclear Regulatory Commission (NRC) approval was obtained as appropriate. The inspectors also reviewed 19 screenings and/or applicability determinations where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. The inspectors reviewed these documents to determine if:

- the changes, tests, and experiments performed were evaluated in accordance with 10 CFR 50.59, and that sufficient documentation existed to confirm that a license amendment was not required;
- the safety issue requiring the change, tests or experiment was resolved;
- the licensee conclusions for evaluations of changes, tests, and experiments were correct and consistent with 10 CFR 50.59; and
- the design and licensing basis documentation was updated to reflect the change.

The inspectors used, in part, Nuclear Energy Institute 96-07, “Guidelines for 10 CFR 50.59 Implementation,” Revision 1, to determine acceptability of the completed evaluations, and screenings. The Nuclear Energy Institute 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. The inspectors also consulted Part 9900 of the NRC Inspection Manual, “10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments.”

This inspection sample constituted 5 evaluations and 19 samples of screenings and/or applicability determinations as defined in Inspection Procedure 71111.17-04. The inspectors could not review the minimum sample size of 6 evaluations because the licensee only performed 5 evaluations during the triennial sample period.

b. Findings

Failure to Document 50.59 Evaluation for Replacement of a Manual Action with an Automatic Action

Introduction: The inspectors identified a Severity Level IV, Non-Cited Violation (NCV) of 10 CFR 50.59, "Changes, Tests, and Experiments," for failure to document the basis for performing a plant modification where a manual operator action was replaced with an automatic action. Specifically, the licensee did not evaluate whether adding a safety-related function to a nonsafety-related component was within the licensing basis of the facility.

Description: The inspectors reviewed Plant Modification 11-0794 where obsolete Nonsafety-Related pneumatic flow controllers in the Safety-Related Battery Room ventilation system and the Safety-Related Motor Control Center, Switchgear, and Miscellaneous Electrical Equipment Area ventilation system were replaced with new digital controllers. As part of this modification an automatic action, to automatically shift to RECIRC MODE during Loss of Offsite Power, Loss of Coolant Accident, or Loss of Nonsafety-Related power events, was added to the replacement digital flow controller. The function to switch to RECIRC MODE during Loss of Offsite Power, Loss of Coolant Accident, or Loss of Nonsafety-Related power events had been performed manually by the operators; therefore, this change converts a manual operator action to an automatic action. The 10 CFR 50.59 Screening performed to support this modification identified that this change was not adverse and further concluded that no 10 CFR 50.59 evaluation or License Amendment Request was required to implement this modification. The inspectors reviewed Procedure NOBP-LP-4003A, FENOC 10 CFR 50.59 User Guidelines, and determined that Attachment 2 of this procedure required the licensee to consider replacing a manual action by automatic action when a design function relies on accomplishing the action to be an adverse change, and, therefore, required the licensee to perform an evaluation. The inspectors also identified that the modification added a Safety Related function, switching to RCIRC MODE during Loss of Offsite Power, Loss of Coolant Accident, or Loss of Nonsafety-Related power events, to a component that was classified as Nonsafety-Related without re-classing the component as a Safety-Related Component. The inspectors determined that adding a Safety-Related function to a Nonsafety-Related component was not within the design basis of the facility; therefore the inspectors could not reasonably determine that this change would not have never required prior NRC review and approval.

Analysis: The inspectors determined that the failure to perform a 10 CFR 50.59 evaluation for Plant Modification 11-0794 was contrary to 10 CFR 50.59(d)(1) and was a performance deficiency. Specifically, Plant Modification 11-0794 replaced a manual action with an automatic action which is considered an adverse change and required, at a minimum, a 10 CFR 50.59 evaluation to be performed. The performance deficiency was determined to be more-than-minor and a finding, because the finding impacted mitigating systems cornerstone attribute of Design Control and adversely affected the Cornerstone Objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, this plan modification added a Safety-Related function to a Nonsafety-Related component and, therefore, impacted the availability, reliability, and capability of the Safety-Related Battery Room ventilation system and the Safety-Related Motor Control Center, Switchgear, and Miscellaneous Electrical Equipment Area ventilation system. In addition, the associated violation 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.

Violations of 10 CFR 50.59 are dispositioned using the traditional enforcement process instead of the Significance Determination Process (SDP) because they are considered to be violations that potentially impede or impact the regulatory process. This violation is associated with a finding that has been evaluated by the SDP and communicated with an SDP color reflective of the safety impact of the deficient licensee performance. The SDP, however, does not specifically consider the regulatory process impact. Thus, although related to a common regulatory concern, it is necessary to address the violation and finding using different processes to correctly reflect both the regulatory importance of the violation and the safety significance of the associated finding.

The inspectors determined that finding could be evaluated using the SDP in accordance with Inspection Manual Chapter 0609, "Significance Determination Process." Using Attachment 0609.04, "Initial Characterization of Findings," Table 2, the inspectors determined that the finding affected the Mitigating Systems cornerstone. As a result, the inspectors evaluated the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 2, for the Mitigating Systems cornerstone. The inspectors answered "No" to question A.4 in Exhibit 2 – Mitigating System Screening Questions. Specifically, the inspectors determined the finding did not represent an actual loss of the Battery Room ventilation system or Motor Control Center, Switchgear, and Miscellaneous Electrical Equipment Area ventilation system because the automatic action had not been implemented at the time of the finding. Therefore, the inspectors determined the significance of this finding to be of very-low safety significance (Green).

In accordance with Section 6.1.d of the NRC Enforcement Policy this violation is categorized as Severity Level IV because the resulting changes were evaluated by the SDP as having very-low safety significance (i.e., green finding).

The finding was associated with the cross-cutting aspect of Procedure Adherence in the area of Human Performance, because the licensee failed to follow the screening criteria in Attachment 2 of Procedure NOBP-LP-4003A, FENOC 10 CFR 50.59 User Guidelines that would have required the licensee to perform a 50.59 evaluation for this modification. [H.8]

<u>Enforcement</u>: Title 10 CFR Part 50.59, "Changes, Tests, and Experiments," Section (d)(1) requires the licensee to maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant 10 CFR 50.59(c). These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment.

Contrary to the above, on January 2, 2013, the licensee failed to document the basis for performing a modification where a manual action was replaced with an automatic action. Specifically, the licensee did not evaluate whether adding a safety-related function to a nonsafety-related component was within the licensing basis of the facility.

This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was a Severity Level IV violation and was entered into the licensee's Corrective Action Program as CR2016-06714. The licensee's immediate corrective action included corrective actions to re-evaluate whether implementation of the portion of the modification that installed automatic action would ultimately be completed. **(NCV 05000440/2016007-02; "Failure to Document 50.59 Evaluation for Replacement of a Manual Action with an Automatic Action").**

.2 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed 14 permanent plant modifications that had been installed in the plant during the last 3 years. This review included in-plant walkdowns for portions of the modified Emergency Service Water piping and pipe supports and the Residual Heat Removal Service Water piping and pipe supports in the pump house. The modifications were selected based upon risk significance, safety significance, and complexity. The inspectors reviewed the modifications selected to determine if:

- the supporting design and licensing basis documentation was updated;
- the changes were in accordance with the specified design requirements;
- the procedures and training plans affected by the modification have been adequately updated;
- the test documentation as required by the applicable test programs has been updated; and
- post-modification testing adequately verified system operability and/or functionality.

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.

This inspection constituted 14 permanent plant modification samples as defined in Inspection Procedure 71111.17-04.

b. Findings

Use of Unapproved Standard for Site Flooding Modifications and Associated Analysis

Introduction: The inspectors identified a Severity Level IV, NCV of 10 CFR 50.59, "Changes, Tests, and Experiments," having very-low safety significance (Green) for the licensee's failure to conclude that site flooding modifications and associated analysis included a standard that resulted in a departure from the method of evaluation as described in the Updated Final Safety Analysis Report (UFSAR). Specifically, the licensee used a new method for evaluation of design basis flooding at the Perry Nuclear Power Plant that is different from the method described in the UFSAR and not approved by the NRC.

Description: The inspectors reviewed the licensee's 10 CFR 50.59 evaluation of engineering change Evaluation 14-01234, "Perry Nuclear Power Plant Flooding Modifications and Associated Analysis." In the 10 CFR 50.59 evaluation, the licensee answered "No" to the question on whether the modifications and associated analysis resulted in a departure from the method of evaluation as described in the UFSAR. However, the inspectors identified that the licensee had used a new method of evaluation that was not described in UFSAR and that had not been approved by the NRC. Specifically, Table 1.8-1 in the UFSAR states that Regulatory Guide 1.59 is the guidance used for the Perry Nuclear Power Plant design basis flooding analysis. The Regulatory Guide 1.59 explicitly endorses American National Standards Institute N-170-1976. The licensee however, implemented the use of American Nuclear Society 2.8-1992 Edition. Nuclear Energy Institute 96-07, "Guidelines for 10 CFR 50.59 Implementation," Section 4.3.8.2, states in part, that a licensee can make changes from one method evaluation to another method provided that the new method is approved by the NRC for the intended application. Standard American Nuclear Society 2.8-1992 does not have NRC approval for use in site flooding analysis. Therefore, the inspectors determined that the licensee's use of this new American Nuclear Society 2.8-1992 methodology for flooding modifications and associated analysis was a departure from the method of evaluation as described in the UFSAR and would require prior NRC review and approval.

Analysis: The inspectors determined that the licensee's use of an unapproved methodology for site flooding modifications and associated analysis that constituted a departure from a method of evaluation was contrary to 10 CFR 50.59(c)(2)(8) and was a performance deficiency. Specifically, the licensee used a new method for evaluation of design basis flooding at the Perry Nuclear Power Plant that is different from the method described in the UFSAR and not approved by the NRC. The performance deficiency was determined to be more-than-minor, and a finding, because it affected the cornerstone attribute of protection against external factors and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.

In addition, the associated violation was determined to be more-than-minor because the inspectors determined that there was a reasonable likelihood that the changes would have required prior NRC approval.

Violations of 10 CFR 50.59 are dispositioned using the traditional enforcement process instead of the SDP because they are considered to be violations that potentially impede or impact the regulatory process. This violation is associated with a finding that has been evaluated by the SDP and communicated with an SDP color reflective of the safety impact of the deficient licensee performance. The SDP, however, does not specifically consider the regulatory process impact. Thus, although related to a common regulatory concern, it is necessary to address the violation and finding using different processes to correctly reflect both the regulatory importance of the violation and the safety significance of the associated finding.

The inspectors determined that finding could be evaluated using the SDP in accordance with Inspection Manual Chapter 0609, "Significance Determination Process." Using Attachment 0609.04, "Initial Characterization of Findings," Table 2, the inspectors determined that the finding affected the Mitigating Systems cornerstone. As a result, the inspectors evaluated the finding using Appendix A, "The Significance Determination

Process (SDP) for Findings At-Power," Exhibit 2, for the Mitigating Systems cornerstone. The inspectors answered "Yes" to question A.1 in Exhibit 2 – Mitigating Systems Screening Questions. Specifically, the inspectors determined the finding did not result in systems, structures, and components not being able to maintain their operability or functionality. Therefore, the inspectors determined the significance of this finding to be of very-low safety significance (Green).

In accordance with Section 6.1.d of the NRC Enforcement Policy this violation is categorized as Severity Level IV because the resulting changes were evaluated by the SDP as having very-low safety significance (i.e., green finding).

The inspectors determined that this finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Problem Identification, for the licensee's failure to identify issues completely, accurately, and in a timely manner. Specifically, the licensee's 50.59 review committee failed to accurately identify the methodology change concern in Evaluation 14-01234 during a review documented in CR2015-14025. [P.1]

Enforcement: Title 10 CFR 50.59, "Changes, Tests, and Experiments," Section (c)(2)(VIII) states, in part, that "A licensee shall obtain a license amendment pursuant to Section 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would result in a departure from a method of evaluation described in the Final Safety Analysis Report (as updated) used in establishing the design basis or in the safety analyses."

Contrary to the above, on July 13, 2015, the licensee implemented modifications and associated analysis for site design basis flooding using a methodology that resulted in a departure from the method of evaluation described in the UFSAR without prior NRC review and approval. Specifically, the licensee failed to obtain a license amendment pursuant to Section 50.90 prior to use of a new flooding analysis methodology that was different than the methodology described in the UFSAR and that was not approved by the NRC for the intended application.

This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was a Severity Level IV violation and was entered into the licensee's Corrective Action Program as CR2016-06882. The licensee's immediate corrective actions included a prompt operability assessment to determine that there was reasonable assurance that systems, structures, and components would maintain operability/functionality under the existing UFSAR described methodology post modifications. **(NCV 05000440/2016007-03; "Use of Unapproved Standard for Site Flooding Modification and Associated Analysis").**

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

.1 Routine Review of Condition Reports

a. Inspection Scope

The inspectors reviewed several corrective action process 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 of changes, tests, and experiments. 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 inspectors are listed in the Attachment to this report.

b. Findings

Failure to Comply With American Society of Mechanical Engineers (ASME) Code Requirements for Repair on Code Class 1 Component

Introduction: A finding of very-low safety significance (Green) and associated NCV of 10 CFR 50.55a(g)(4) was identified by the inspectors for the licensee's failure to maintain the ASME Code Class 1 component in accordance with ASME Code Section XI requirements. Specifically, the licensee failed to measure and document the method of measuring the cavity created after removal of indications on the reactor water clean-up line prior to return to service.

Description: On May 10, 2016, the inspectors reviewed Condition Reports 2015-01443 and 2015-01439 that identified unacceptable indications on the reactor water clean-up replacement bottom head drain, which is an ASME Code Class 1 component. The inspectors identified that the licensee had removed the indications in accordance with ASME Section XI, repair/replacement requirements by grinding out the indications. However, the inspectors identified that the licensee had failed to record the method of measuring the depth of the cavity created on the component after grinding out the indications. Specifically, ASME Code Section XI, IWA 4150, requires that the licensee's repair/replacement plan shall include the method of measurement of the cavity created by removing a defect. The licensee's repair/replacement plan did not contain the measurement method because the licensee had not measured the cavity after defect removal and returned the component back to service without this measurement. The inspectors questioned the licensee on how the component meets the minimum design wall thickness requirements if the depth of the cavity created was not known.

The licensee documented this condition adverse to quality into its Corrective Action Program under CR2016-06372. As part of immediate corrective actions, the licensee interviewed the staff that performed the grinding on the component and determined that given the duration of grinding, sufficient material would not have been removed to challenge component functionality given the original wall thickness of the component. In addition, the licensee also intends to perform cavity measurements during the next refueling outage to confirm that the component meets minimum design wall thickness.

Analysis: The inspectors determined that the licensee's failure to maintain the ASME Code Class 1 component in accordance with ASME Code Section XI requirements was a performance deficiency. This performance deficiency was found to be more-than-minor, and a finding, because the performance deficiency, if left uncorrected could become a more significant safety concern. Specifically, absent NRC identification, the licensee would not have questioned the potential challenge to component functionality since the cavity measurements were not performed. This condition could potentially lead to the failure of the reactor water clean-up bottom head drain, which in turn, could lead to a potential loss of reactor coolant.

The inspectors reviewed the finding using Attachment 0609.04, “Initial Characterization of Findings,” Table 3 – SDP Appendix Router. The inspectors answered ‘No’ to the question in Section A of Table 3 and therefore the finding was evaluated using the SDP in accordance with IMC 0609, “The Significance Determination Process (SDP) for At-Power Operations,” Appendix A, Exhibit 1, “Initiating Events Screening Questions”. The inspectors answered “No” to the questions in Exhibit 1 and determined this finding to have a very-low safety significance (Green).

The inspectors determined that this finding has a cross-cutting aspect in the area of Human Performance, Design Margin, for the licensee’s failure to maintain equipment within design margins. Specifically, the licensee staff failed to ensure that metal removal performed on an ASME Code Class 1 component did not result in a condition where the minimum design wall thickness of the component was compromised, and therefore, failed to ensure design margin was maintained. [H.6]

<u>Enforcement</u>: Title 10 CFR 50.55a(g)(4) states in part that in part, that throughout the service life of a boiling or pressurized water reactor facility, components which are classified as ASME Code Class 1, 2, and 3 must meet requirements set forth in Section XI.

The ASME Code of record at Perry Nuclear Power Plant is the 2001 Edition through 2003 Addenda. Section XI, subsection IWA 4152, “Repair/Replacement Program and Plan”, requires, in part, that the “method of measurement of the cavity created from defect removal shall be documented in the repair/replacement plan”.

Contrary to the above, in February 2015, the licensee failed to document, in its repair/replacement plan, the method of measurement of the cavity created after removal of indications identified in the reactor water clean-up bottom head drain. Because this issue is of very-low safety significance, and was entered into the licensee’s Corrective Action Program under CR2016-06372, it is being treated as an NCV consistent with Section 2.3.2 of the NRC enforcement policy. As part of immediate corrective actions, the licensee interviewed the staff that performed the grinding on the component and determined that given the duration of grinding, sufficient material would not have been removed to challenge component functionality given the original wall thickness of the component. In addition, the licensee also intends to perform cavity measurements during the next refueling outage to confirm that the component meets minimum design wall thickness. **(NCV 05000440/2016007-03, “Failure to Comply With ASME Code Requirements for Repair on Code Class 1 Component”).**

4OA6 Management Meetings

.1 <u>Exit Meeting Summary</u>

The inspectors presented the inspection results to Mr. D. Hamilton and other members of the licensee staff on May 27, 2016. The licensee personnel acknowledged the inspection results presented, and did not identify any proprietary content.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Hamilton, Vice President Nuclear PY
F. Payne, Plant General Manager
T. Brown, Director Performance Improvement
D. Reeves, Director Site Engineering
L. Zerr, Supervisor Regulatory Compliance
D. Lockwood, Licensing Engineer
D. Haviland, Senior Consultant Design Engineering
R. Briggs, Supervisor Design Engineering - Electrical

U.S. Nuclear Regulatory Commission

M. Marshfield, Senior Resident Inspector
J. Nance, Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000440/2016007-01	NCV	Failure to Document 50.59 Evaluation for Replacement of a Manual Action with an Automatic Action (Section 1R17.1.b)
05000440/2016007-02	NCV	Use of Unapproved Standard for Site Flooding Modifications and Analysis (Section 1R17.2.b)
05000440/2016007-03	NCV	Failure to Comply With ASME Code Requirements for Repair on Code Class 1 Component (Section 4OA2.1)

Discussed

None

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
CFR	*Code of Federal Regulations*
IMC	Inspection Manual Chapter
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
PARS	Public Available Records System
SDP	Significance Determination Process
UFSAR	Updated Final Safety Analysis Report
ASME	American Society of Mechanical Engineers

LIST OF DOCUMENTS REVIEWED

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

10 CFR 50.59 EVALUATIONS

Number	Description or Title	Revision
13-02880	Revision of Design Basis Accident Dose Calculations	01
14-01234	PNPP Flooding Modifications and Associated Analysis	00
14-03283	Design Change to Permit Use of ASME Section VIII Seal Coolers for Reactor Water Cleanup Pumps	00
13-01317	Liquid Radwaste Control System Upgrade	03
14-03283	ECP-14-0625 Alternative Design Code replacement of Reactor Water Clean Up (RWCU) Pump seal coolers inclusive of USAR Change Request 14-237	00

10 CFR 50.59 SCREENINGS

Number	Description or Title	Revision
09-01913	Temporary Modification to Impair Drywell Zone 8 Fire Alarm	02
12-00745	Replace Division I & II Starting Air Compressors	02
13-00811	Installation of Sequence Start Time Delay Relay in the Control Logic for the Diesel Driven Fire Pump	02
14-02853	Revision of Off-Normal Instruction (ONI R-36-2) Extreme Cold Weather Procedure	00
16-00412	Reduce Control Power Fuse Size for 1E51F0076/77/78	00
13-00457	SOI-R44 rev17, Division 1 and 2 DIESEL STARTING AIR SYSTEM	17
12-00745	ECP 11-0671, Replace Division I & II Starting Air Compressors	01
12-01464	PERP-0687, Replacement Evaluation for Division 1 and 2 Diesel Generator Starting Air Dryer, Kahn Model MPS-100S for 1R44D0001A, 1B, 2A, &2B.	00
14-03283	Design Change to Permit Use of ASME Section VIII Seal Coolers for Reactor Water Clean Up Pumps	00
10-05813	Transformer Sudden Pressure Alarm Relay	02
13-04641	Temporary Modification to Jumper out Limit Switch 1N22N0336	00
13-04661	Perry Power Plant Lubrication Manual	00
14-00037	Unit 1 Startup Transformer Replacement	02
14-03694	FLEX Modification for Plant Electrical Systems	06
15-02798	Hardcards	00
14-01572	Qualified Life Calculations for Weed RTD/RTDT & TC Assemblies	00
10-05339	Replace the current Fieldbus H1 card, 1N23R0477, 1N23R0480 and 1N23R0476 with new Series 2 Fieldbus H1 Card	02
11-05239	Setpoint Change for Reactor Booster Pump Differential Pressure Indicating Switch, 1N27N0060	02
12-00928	Division 1 Rosemount Trip Units Re-rack and the Addition of a 125VDC to 24 VDC Converter in Control Room Panel 1H13-P629	03

CORRECTIVE ACTION PROGRAM DOCUMENTS INITIATED DURING INSPECTION

Number	Description or Title	Date
2016-06566	Less than Timely Revision to Calculation R45-012	05/10/2016
2016-06882	Departure in Method Required Prior NRC Approval	05/18/2016
2016-06372	Depth of Indications Not Documented	05/05/2016
2016-06714	50.59 Deficiencies Associated with ECP 10-0811	05/13/2016

CORRECTIVE ACTION PROGRAM DOCUMENTS REVIEWED

Number	Description or Title	Date
2015-01439	Liquid Penetrant Pre-Service Examination Reveals Unacceptable Indication	02/03/2015
2015-04025	Review Committee Identified Concerns with 50.59 Eval	10/16/2015
2016-03290	50.59 Screen for ECP 15-0489 Needs Revision	03/10/2016
2016-03300	FO-SA-2015-0005: Revision of Wirelists Not Identified	03/10/2016
2015-17263	ARC Chute Crack Identified on Breaker Main Supply	12/28/2015
2015-08036	Prompt Functionality Assessment for Site Flooding Analysis	06/12/2015
2013-09255	Perry Drywell unidentified leakage inspection results 6/15/2013 B33 Vent Valve	06/15/2013
2016-01199	Weld on 1B33F0060B Vent Line Appendage Not Per Design	01/26/2016
2016-01255	Weld Failed to Meet Minimum Diameter Criteria	01/28/2016
2014-10651	Safety Related Breaker Refurbishments Impacted By Unavailability Of C/GC 757 Lubricant	06/19/2014
2014-12331	Engineering Change Package (ECP) related Order was closed without implementing the ECP	07/29/2014
2014-14116	Potential Part 21 - Schulz Electric - Event No. 50428	09/09/2014
2014-14505	Start-up Transformer CT wiring not in accordance with Temp Mod 11-0626-001	09/17/2014
2014-16403	Test Switch Mislabeling on L Buses.	10/30/2014
2015-00141	Audit identified active clearance which may exceed 90 days in place.	01/06/2015
2015-14807	Arc Flash current methodology concern	10/30/2015
2014-04755	Wiring configuration on new Freedom MCC bucket	03/11/2014
2014-18574	Two 120v breakers found in the trip free position during the performance of ELI.	12/20/2014
2016-02620	Intermittent Ground on F-1-B 480V Bus	02/24/2016
2015-09807	DIV 2 DG Potential non-conforming condition based on exceeding Design environmental Temperature limits in the DIV 2 DG room during the last performance of SVI-R43-T1318	07/20/2015
2014-11413	Regulatory Applicability Determination (RADs) Incorrectly Applied the Maintenance Exemption (Exemption 1.3)	07/08/2014
2014-00790	2014 Pre-CDBI Assessment – Additional Information needed for DG 7 Day Loading Calculation Assumptions	02/16/2014
2014-15021	Failure to replace activated carbon in normal ventilation systems tested to RG 1.140 in a timely manner following carbon efficiency failure.	10/29/2014

DRAWINGS

Number	Description or Title	Revision
304-0601-00103; Sheet 1	Cover sheet: Reactor Recirculation Valve, Flow Control System, Reactor Bldg.	J
304-0601-00103; Sheet 2	Piping Isometric: Reactor Recirculation Valve, Flow Control System, Reactor Bldg.	J
304-0601-00103; Sheet 3	Detail sheet: Reactor Recirculation Valve, Flow Control System, Reactor Bldg.	J
304-0601-00103; Sheet 4	Detail sheet: Reactor Recirculation Valve, Flow Control System, Reactor Bldg.	J
304-0601-00103; Sheet 5	Detail sheet: Reactor Recirculation Valve, Flow Control System, Reactor Bldg.	J
304-0601-00103; Sheet 6	Bill of Material: Reactor Recirculation Valve, Flow Control System, Reactor Bldg.	J
256-0036-00000	One Line Diagram Non-Class 1E 480V Bus F-2-C	LL
206-0036-00000	One Line Diagram Non-Class 1E 480V Bus F1C	AAAA
206-0020-00000	Main One Line Diagram 480V	GG
206-0010-00000	Main One Line Diagram 13.8KV & 4.16KV	CC
208-0116-00003	Battery Room Exhaust Trip Logic, Auto, Switchover “A” & Alarms	R
208-0116-00001	Battery Room Exhaust Fan C001A	V
208-0116-00004	Battery Room Exhaust Trip Logic, Auto, Switchover “A” & Alarms	P
208-0115-0007	MCC Switchgear & Miscellaneous Electric Equipment Area HVAC System Relay Isolation Logic	K

MODIFICATIONS

Number	Description or Title	Date or Revision
15-0028-003	Fire Barrier and Access Hatch CC-433 and CC-434 Installation - EL 639'-6"	00
13-0649-001	Fuel Handling Building Crane upgrade	02
12-0298-003	Division 3 Diesel Generator Room Heat Detector Spacing	02
G00000010007	Generic Package for the Procurement and commercial grade dedication of spiral wound gaskets	08
100042952	Commercial grade dedication package for BUNA-S gasket	02
94114635	Commercial grade dedication package for lubricating oil	01
14-0625	Alternative Design Code Replacement of Reactor Water Clean Up (RWCU) Pump Seal Coolers Inclusive of USAR Change Request 14-237	00
16-0052	Document-Only ECP for Documenting “Use-As-Is” Disposition of Weld on PY-1B33F0060B Vent Appendage.	00
16-0057	Evaluate and Update Existing Documentation to Reflect Actual Weld Configuration for Valves 1B33F0013A & 1B33F0014A	00
PERP 0687	Equivalent Replacement for Div. 1 & 2 diesel starting air dryers, Kahn model MPS-100S, for 1R44D0001A/1B/2A/2B	01
1S11S0002-13-0286-001	POWER TRANSFORMER 345/13.8 KV UNIT STARTUP XFMRS Setpoints	05/20/2013

MODIFICATIONS

Number	Description or Title	Revision
12-0124-002	Upgrade the existing 480-volt Motor Control(MCC) Automatic Transfer Switch and Relay in MCCF-1d08	00
13-0521-001	Installs Two New Breakers at 4160V Bus EH21	01
14-0246-001	Addition of Fuse Protection for Safety-related DC Ammeters	01
10-0811	Replace obsolete pneumatic Air Monitor Flow Control Stations 0M24K030B, 0M23K040B, & 0M23K060B with electronic Moore Fieldpac Flow Control Stations	05
11-0461	Setpoint Change for Reactor Booster Pump Differential Pressure Indicating Switch, 1N27N0060	02
10-0833	Division 1 Rosemount Trip Units Re-rack and the Addition of a 125VDC to 24 VDC Converter in Control Room Panel 1H13-P629	04

PROCEDURES

Number	Description or Title	Revision
SOI-R44	Division 1 and 2 Diesel Generator Starting Air System	17
NOBP-CC-7001	Procurement Packages (§4.4 Commercial Grade Dedication Procedure)	21
NOP-CC-3302	Calculations	07
NOBP-CC-7001	Procurement Packages	21
NOP-OP-1001	Clearance/Tagging Program	23
FSG40.1	Supplying Alternate Power to Vital Unit 1 Busses	00

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Sincerely,

/RA Mark Jeffers Acting for/

Robert C. Daley, Chief
Engineering Branch 3
Division of Reactor Safety

Docket No. 50–440
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