UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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LISLE, IL 70532-4352

March 24, 2015

Mr. Thomas A. Vehec
Vice President
NextEra Energy Duane Arnold, LLC
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER - TRIENNIAL FIRE PROTECTION INSPECTION REPORT 05000331/2015008

Dear Mr. Vehec:

On February 13, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed a triennial fire protection inspection at your Duane Arnold Energy Center. The enclosed inspection report documents the inspection results, which were discussed on February 13, 2015, 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.

The NRC inspectors documented one finding of very-low safety significance (Green) in this report. This finding was determined to involve a violation of NRC requirements. However, because of its very-low safety significance, and because the issue was entered into your Corrective Action Program, the NRC is treating the issue as a Non-Cited Violation (NCV) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the subject or severity of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with 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 Duane Arnold Energy Center.

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/

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

Docket No. 50-331
License No. DPR-49

Enclosure:
Inspection Report 05000331/2015008
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No: License No:	50-331 DPR-49
Report No:	05000331/2015008
Licensee:	NextEra Energy Duane Arnold, LLC
Facility:	Duane Arnold Energy Center
Location:	Palo, IA
Dates:	January 13 through February 13, 2015
Inspectors:	I. Hafeez, Reactor Inspector M. Jeffers, Reactor Inspector D. Szwarc, Senior Reactor Inspector (Lead) R. Winter, Reactor Inspector
Accompanying Personnel:	H. Barrett, Senior Fire Protection Engineer L. Kozak, Senior Reactor Analyst P. Lain, Senior Fire Protection Engineer S. Laur, Senior Reliability and Risk Analyst
Approved by:	Robert C. Daley, Chief Engineering Branch 3 Division of Reactor Safety

SUMMARY OF FINDINGS

Inspection Report 05000331/2015008; 01/13/2015-02/13/2015; Duane Arnold Energy Center; Routine Triennial Fire Protection Baseline Inspection.

This report covers an announced Triennial Fire Protection Baseline Inspection. The inspection was conducted by Region III inspectors. One finding was identified by the inspectors. The finding 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 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.

NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of Title 10, *Code of Federal Regulations* (CFR) 50.48(c), and National Fire Protection Association Standard 805, Section 2.4.3.2 for the licensee's failure to address in the Fire Probabilistic Risk Assessment (PRA) the risk contribution with all potentially risk-significant fire scenarios. Specifically, the licensee did not address potential damage to safety relief valves (SRVs), or the SRV tailpipes as a result from fire induced overfill of the reactor pressure vessel. The licensee entered this issue into their Corrective Action Program to review the multiple spurious operations Expert Panel report, and properly disposition the scenario.

  The inspectors determined that the performance deficiency was more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of Protection against External Factors (i.e., fire), and it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the missed failure mechanism for the SRVs had the potential to impact the ability to achieve safe and stable conditions. In accordance IMC 0609, Appendix F, "Fire Protection SDP," Attachment 1, Step 1.6.1, "Screen by Licensee PRA-Based Safety Evaluation," the inspectors were able to use the Licensee's PRA to evaluate the safety significance of the finding. The increase in core damage frequency (CDF) as a result of the identified scenario was found to be approximately 2.6E-7 per year; therefore, the inspectors concluded that this finding was of very-low safety significance (Green). This finding did not have a cross-cutting aspect because it was not representative of current licensee performance. (Section 1R05.6.b)

Licensee-Identified Violations

No violations were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events and Mitigating Systems

1R05 Fire Protection (71111.05XT)

The inspectors conducted the inspection in accordance with U.S. Nuclear Regulatory Commission (NRC) Inspection Procedure (IP) 71111.05XT, "Fire Protection-National Fire Protection Association (NFPA) 805 (Triennial)," issued January 31, 2013. The inspectors reviewed the licensee's Fire Protection Program against the requirements of NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition," as incorporated by Title 10, *Code of Federal Regulation* (CFR) 50.48(c). The NFPA 805 standard establishes a comprehensive set of requirements for Fire Protection Programs at nuclear power plants. The standard incorporates both deterministic and risk-informed performance-based concepts. The deterministic aspects of the standard are comparable to traditional requirements.

The inspectors conducted a design-based, plant-specific, risk-informed, onsite inspection of the licensee's Fire Protection Program's defense-in-depth elements used to mitigate the consequences of a fire. The inspectors reviewed the licensee's Fire Protection Program to ensure that it met the fire protection concept of defense-in-depth for plant areas important to safety by:

- preventing fires from starting;
- rapidly detecting, controlling and extinguishing fires that do occur;
- providing protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by fire suppression activities will not prevent the safe-shutdown of the reactor plant; and
- taking reasonable actions to mitigate postulated events that could potentially cause loss of large areas of power reactor facilities due to explosions or fires.

The inspectors evaluated the licensee's Fire Protection Program by focusing on the design, installation, operational status, testing, and material condition of the Fire Protection Program, post-fire safe shutdown systems, and B.5.b mitigating strategies. The inspectors verified that the licensee's program is sufficiently implemented and maintained to satisfy that nuclear safety and radioactive release goals, objectives, and performance criteria for all operational modes and plant configurations.

In addition, the inspectors' review and assessment focused on the licensee's post-fire safe shutdown systems for selected risk-significant fire areas. Inspector emphasis was placed on determining that the post-fire safe shutdown capability and the fire protection features were maintained free of fire damage to ensure that at least one post-fire safe shutdown success path was available. The inspectors' review and assessment also focused on the licensee's B.5.b related license conditions, and the requirements of 10 CFR 50.54 (hh)(2). The inspector's emphasis was to ensure that the licensee could maintain or restore core cooling, containment, and spent fuel pool cooling capabilities utilizing the B.5.b mitigating strategies following a loss of large areas of power reactor facilities due to explosions or fires. Documents reviewed are listed in the Attachment to this report.

The fire areas, fire zones, and B.5.b mitigating strategies selected for review during this inspection are listed below, and in Section 1R05.15. The fire areas and fire zones selected constituted four inspection samples, and the B.5.b mitigating strategies selected constituted two inspection samples, respectively, as defined in IP 71111.05XT.

Fire Area	Fire Zone	Description
CB1	11A	Cable Spreading Room
CB3	10F	Division I Essential Switchgear Room
PH1	16F	Pumphouse Basement
RB1	02A	Reactor Building – North Control Rod Drive Module Area

.1 Protection of Safe Shutdown Capabilities

a. Inspection Scope

The inspectors reviewed the licensee's fire response abnormal operating procedures (AOPs), and compared them to the Nuclear Safety Capability Assessment (NSCA) documents to verify that the shutdown methodology properly identified the components and systems necessary to achieve and maintain safe and stable plant conditions. The inspectors performed a walk-through of the shutdown from outside of the control room AOP to ensure that operators could reasonably perform the actions specified in the procedure. For each of the selected fire areas, the inspectors reviewed the fire hazards analysis, NSCA, and supporting drawings and documentation to verify that safe shutdown capabilities were properly protected.

b. Findings

No findings were identified.

.2 Passive Fire Protection

a. Inspection Scope

For the selected fire areas, the inspectors evaluated the adequacy of fire area barriers, penetration seals, fire doors, electrical raceway fire barrier systems, and fire rated electrical cables. The inspectors walked down accessible portions of the selected fire areas to observe material condition, construction details, and the adequacy of design of fire area boundaries (including walls, fire doors, and fire dampers) to ensure they were appropriate for the fire hazards in the area. The inspectors reviewed license documentation, such as NRC NFPA 805 safety evaluation reports, and NFPA standards to verify that Fire Protection Program features met license commitments. The inspectors reviewed the installation, repair, and qualification records for a sample of penetration seals to ensure the fill material was of the appropriate fire rating, and that the installation met the engineering design. In addition, the inspectors reviewed a sample of surveillance and maintenance procedures for selected fire doors, fire dampers, and fire barrier penetration seals to assure they were properly inspected and repaired.

b. Findings

No findings were identified.

.3 Active Fire Protection

a. Inspection Scope

The inspectors walked down and evaluated the adequacy of fire suppression and detection systems to determine that they were installed, tested, and maintained to adequately control and/or extinguish fires associated with the hazards of the selected fire areas. The inspectors observed the material condition, operational lineup, and design of the installed fire detection and suppression systems, including the electric motor driven, diesel motor driven, jockey fire pumps, carbon dioxide system, manual fire hose and standpipe systems, and fire extinguishers in the selected fire areas. The inspectors reviewed fire pre-plans, and procedures for the selected fire areas to determine if appropriate information was provided to fire brigade members. In addition, the inspectors observed the placement of the fire hoses, fire extinguishers, fire hose nozzle types, and fire hose lengths to verify they were not blocked, and that adequate reach and coverage was provided consistent with the fire protection features and potential fire conditions described in the NFPA 805 fire safety analysis calculations. Additionally, fire brigade drill reports and scenarios that transpired since 2012 were reviewed to verify that fire brigade monitoring criteria were met.

b. Findings

No findings were identified.

.4 Protection from Damage from Fire Suppression Activities

a. Inspection Scope

The inspectors evaluated that one success path to achieve and maintain the Nuclear Safety Performance Criteria could be achieved, and would not be adversely affected due to damage from fire suppression activities or from the rupture or inadvertent operation of manual fire suppression systems. The inspectors walked down the selected fire areas to assess in-plant conditions including adequacy and material condition of equipment spray protection, elevations of vulnerable equipment and checked that water would either be contained in the fire affected area, or be safely drained off through floor drains or to other areas. The inspectors addressed the possibility that a fire in one fire area could lead to the migration of smoke or hot gases to other plant areas. Air flow paths out of the selected fire areas identified on heating ventilation and air conditioning drawings were reviewed to verify that inter-area migration of smoke or hot gases would not inhibit necessary post-fire recovery actions for the selected fire areas.

b. Findings

No findings were identified.

.5 Shutdown from a Primary Control Station

a. Inspection Scope

The inspectors' reviews focused on ensuring that the required functions for post-fire safe shutdown (SSD), and the corresponding equipment necessary to perform those functions were included in the fire response AOPs. The review included assessing whether safe and stable plant conditions from the primary control stations outside the main control room could be implemented and that transfer of control from the main control room to the remote shutdown panel could be accomplished in accordance with procedure AOP-915, "Shutdown Outside Control Room." The inspectors walked down the actions identified in the procedure with the licensee to verify operators were properly trained, assess human factors, and ensure the procedures could be completed as written.

b. Findings

No findings were identified.

.6 Circuit Analyses

a. Inspection Scope

The inspectors verified that the licensee performed an NSCA for the selected fire areas, and that the assessment identified the structures, systems, and components important for achieving safe and stable conditions. For each fire area, the inspectors reviewed the electrical schematics, flow diagrams, and the NSCA to identify any potential fire-induced cable damage that could directly affect post-fire SSD. The inspectors reviewed a sample of circuit diagrams to verify that all appropriate cables had been selected and incorporated into the NSCA. The inspectors then evaluated selected circuits to ensure all fire scenarios had been identified, and dispositioned for all modes of operation including shut down operations, and abnormal plant configurations.

The inspectors verified that the NSCA demonstrated that hot shorts, shorts to ground, or other failures that would result in a spurious actuation will not affect the capability to meet the performance criteria. The inspectors reviewed the licensee's breaker selected coordination analysis between 4.16 kilovolt essential safety features buses, and the standby transformer. The inspectors verified that the licensee's assessment identified circuits that may impact the Nuclear Safety Performance Criteria. The assessment demonstrated that hot shorts, shorts to ground or other failures that would not result in a spurious actuation will not affect the capability to meet the performance criteria. The inspectors reviewed fire scenarios and cable attributes, potential undesirable consequences, and common power supply/bus concerns.

The inspectors also reviewed the licensee's response to multiple spurious operations (MSOs) as identified by Nuclear Energy Institute's (NEI's) document, NEI 00-01, and the site's Expert Panel. The review ensured that the licensee followed the approved guidance provided by NEI 00-01, evaluated all appropriate MSO scenarios, and properly addressed any discrepancies.

b. Findings

Failure to Identify and Evaluate the Effects of Vessel Overfill Scenario

Introduction: The inspectors identified a finding of very-low safety significance (Green), an associated Non-Cited Violation (NCV) of 10 CFR 50.48(c), and NFPA 805, Section 2.4.3.2, for the licensee's failure to address in the Fire Probabilistic Risk Assessment (PRA) the risk contribution with all potentially risk-significant fire scenarios. Specifically, the licensee did not address potential damage to safety relief valves (SRVs), or the SRV tailpipes as a result of fire induced overfill of the reactor pressure vessel.

Description: The licensee conducted an initial MSO Expert Panel in August 2008 at Duane Arnold Energy Center (DAEC) to identify potential scenarios that may require analysis and treatment as part of the transition process to NFPA 805. The Expert Panel reconvened in March 2010 to address action items from the original panel, and to address changes associated with the Boiling Water Reactors Owner Group (BWROG) Generic MSO list in NEI document, NEI 00-01, Revision 2, "Guidance for Post-Fire SSD Circuit Analysis." Subsequently, the disposition of the MSO scenarios was documented in Report 0027-0042-000-002, "DAEC MSO Expert Panel Report."

One of the MSO Scenarios identified by the BWROG in NEI 00-01, Revision 2, included the possibility of vessel overfill as a result of fire induced failures of the reactor feed pumps (Reference NEI 00-01 Revision 2, Appendix G, Scenario 2ai). The generic scenario states:

> "Spurious operation of a feedwater, or booster pump, and a level control valve may cause uncontrolled feedwater injection into the reactor pressure vessel (RPV). This could also include continued operation of the Feedwater Pump (driven off the main turbine shaft). Fire damage to the feedwater pump clutch and/or associated controls could prevent tripping the pump, resulting in a serious overfeed situation."

However, the scenario was documented in the Expert Panel Report as, "Spurious operation of one Reactor Feedwater booster pump, and isolation valve pair, combined with one level control valve may cause uncontrolled feedwater injection into the RPV." Consequently, the Expert Panel dispositioned the scenario as not applicable to DAEC since the site did not have the specified equipment identified by the generic MSO scenario.

After discussions with the licensee the inspectors determined that the licensee did identify a scenario in the site's PRA related to uncontrolled feedwater. The Fire PRA models vessel overfill by feedwater as a failure mechanism for the high-pressure coolant injection (HPCI), and reactor core isolation cooling (RCIC) high-pressure injection systems. No credit is given for operator intervention to manually trip feedwater pumps, nor is credit given for operator intervention to restore HPCI or RCIC. However, the Fire PRA did not take into account the effects of the overfill scenario with regard to the effects on the SRVs.

In this scenario, the RPV would fill and flood the main steam lines because fire induced cable damage would prevent tripping the feedwater pump. In response to the continuing feedwater injection, SRVs would lift after RPV pressure reaches the SRV set point, and

RPV water would be discharged through the SRVs, and down comers to the suppression pool. This would result in high-pressure, high-temperature water discharge through SRVs, which would flash to two phase flow, and increase the likelihood of the SRVs sticking open after passing water. This would prevent allowing the operator to take positive control of the SRVs. The credited NSCA success path in these scenarios is for operators to control pressure via the SRVs, and low-pressure systems for inventory control.

The inspectors discussed this scenario with the licensee, and subsequently, the licensee performed an evaluation to determine the sensitivity of Fire PRA results to potential SRV and/or SRV tailpipe damage from RPV overfill by the feedwater system. The new scenario was modeled using the current success criteria associated with large loss of coolant accident (LOCA) events (i.e., SRVs stuck open). The increase in core damage frequency (CDF) was found to be approximately 2.6E-7/year. The increase in CDF is dominated by four fire scenarios initiated in the feedwater pump area where the fire is assumed to prevent the pumps from being tripped. The licensee entered this issue into their Corrective Action Program (CAP) as Action Request (AR) 2024869 to review the MSO Expert Panel Report, and properly disposition the scenario.

Analysis: The inspectors determined that the licensee's failure to identify the risk-contribution of all impacts as a result of a fire-induced vessel overfill scenario was contrary to NFPA 805, Section 2.4.3.2. and was a performance deficiency. Specifically, the licensee failed to include the potential damage to the SRVs, and subsequent LOCA as a result of SRVs being stuck open.

The performance deficiency was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of Protection Against External Factors (i.e., fire), and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to address the risk to include the potential to damage the SRVs which are part of the credited SSD success path.

The inspectors evaluated the finding in accordance with Inspection Manual Chapter 0609, "Significance Determination Process (SDP)," dated June 2, 2011, Attachment 4, "Initial Characterization of Finding," dated June 19, 2012, Table 2, and determined that the finding affected the Mitigating System cornerstone. The finding degraded fire protection defense-in-depth strategies, and the inspectors determined, using Table 3, that it could be evaluated using Appendix F, "Fire Protection SDP," dated September 20, 2013. The inspectors used Attachment 1, "Fire Protection SDP Worksheet," dated September 20, 2013, as the finding affected post-fire SSD, and screened the finding as of very-low safety significance (Green) in Step 1.6.1, "Screen by Licensee PRA-Based Safety Evaluation."

The Senior Reactor Analyst (SRA) performed a review of the results of the licensee's risk evaluation of the scenario, and performed an independent analysis using the licensee's fire frequencies and the NRC Standardized Plant Analysis Risk Model for Duane Arnold. The SRA concluded that the CDF for these scenarios was not greater than 1E-6/yr, and that there was no change in plant risk as result of the failure to include this scenario in the plant fire PRA.

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. The licensee's Expert Panel dispositioned this scenario in 2010.

Enforcement: License condition 2.C(3) requires the licensee to implement and maintain in effect all provisions of the approved Fire Protection Program that complies with 10 CFR 50.48(a) and 10 CFR 50.48(c), "NFPA Standard NFPA 805," as approved in the safety evaluation report dated September 10, 2013. Section 2.4.3.2 of NFPA 805 states that the Probabilistic Safety Assessment Evaluation shall address the risk-contribution associated with all the potentially risk-significant fire scenarios.

Contrary to the above, from September 10, 2013, until February 13, 2015, the licensee failed to identify the risk contribution of all potentially risk-significant fire scenarios. Specifically, the licensee did not address the potential damage to the SRVs and subsequent LOCA with the SRVs being stuck open as a result of a fire-induced vessel overfill scenario.

This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very-low safety significance (Green), and was entered into the licensee's CAP as AR 02024869. The licensee planned to review the MSO Expert Panel Report, and properly disposition the scenario. (NCV 05000331/ 2015008-01; Failure to Identify and Evaluate the Effects of Vessel Overfill Scenario)

.7 Communications

a. Inspection Scope

The inspectors reviewed, on a sample basis, the adequacy of the communication system to support plant personnel in the performance of alternative SSD functions, and fire brigade duties. The inspectors verified that plant telephones, page systems, sound powered phones, and radios were available for use, and maintained in working order.

b. Findings

No findings were identified.

.8 Emergency Lighting

a. Inspection Scope

The inspectors performed walkdowns of the selected fire zones, and observed the placement and coverage area of the fixed battery pack emergency lights credited for SSD. As part of the walkdowns, the inspectors focused on the existence of sufficient emergency lighting for access and egress to areas, and for performing necessary equipment operations. The inspectors verified that battery power supplies had sufficient capacity to support recovery actions necessary to meet the Nuclear Safety Performance Criteria. The inspectors reviewed the operability testing and maintenance of the lightning units to ensure that they followed licensee procedures, and accepted industry practice.

b. Findings

No findings were identified.

.9 Cold Shutdown Repairs

a. Inspection Scope

The inspectors determined that the licensee does not credit cold shutdown repairs to meet the Nuclear Safety Performance Criteria. The inspectors reviewed the NSCA to verify that the licensee had evaluated the need for cold shutdown repairs. The inspectors also interviewed licensee personnel, and determined that the licensee does not require transitioning to cold shutdown to achieve a safe and stable condition.

b. Findings

No findings were identified.

.10 Compensatory Measures

a. Inspection Scope

The inspectors conducted a review to verify that compensatory measures were in place for out-of-service, degraded, or inoperable fire protection, and post-fire SSD equipment, systems, or features (e.g., detection and suppression systems, and equipment, passive fire barriers, pumps, valves or electrical devices providing SSD functions or capabilities). The inspectors also conducted a review of the adequacy of short term compensatory measures to compensate for a degraded function or feature until appropriate corrective actions were taken.

b. Findings

No findings were identified.

.11 Radiological Release

a. Inspection Scope

The inspectors verified that the licensee had provided reasonable assurance that a fire would not result in a radiological release that adversely affects the public, plant personnel, or the environment in accordance with NFPA 805, Section 1.3.2. The inspectors verified that the licensee had evaluated the potential for radioactive releases to any unrestricted areas resulting from fire suppression activities were as-low-as-reasonably-achievable. The inspectors verified that the licensee had analyzed radioactive release on a fire area basis in accordance with NFPA 805, Section 2.2.4. The inspectors walked down the selected fire zones, and verified that the pre-fire plan tactics, and instructions were consistent with the potential radiological conditions identified in the fire hazards analysis.

b. Findings

No findings were identified.

.12 Non-Power Operations

a. Inspection Scope

The plant did not enter an outage during the inspection. However, the inspectors verified that the licensee had defined specific pinch points where one or more key safety functions could be lost during non-power operations. The inspectors reviewed the actions that the licensee would take during higher-risk evolutions where those key safety functions could be lost.

b. Findings

No findings were identified.

.13 Monitoring Program

a. Inspection Scope

The inspectors verified that the licensee had established a monitoring program to ensure that the availability and reliability of the fire protection systems, structures and components credited in the performance-based analyses are maintained, and to assess the performance of the fire protection program in meeting the performance criteria as specified in NFPA 805. The items in scope were being monitored for availability, reliability, and performance based on the established maintenance rule criteria with the results input into the system health report process. The inspectors also verified that the monitoring program utilized the CAP to return availability, reliability, and performance of systems that fall outside of established levels.

b. Findings

No findings were identified.

.14 Plant Change Evaluation

a. Inspection Scope

The inspectors reviewed plant change evaluations to verify that the modifications met the requirements of the fire protection license condition for self-approved changes to the fire protection program. Due to the small number of plant change evaluations performed since the implementation of NFPA 805, the inspectors reviewed a sample of engineering changes that had screened out from having to perform the fire protection plant change evaluation. Additionally, the inspectors reviewed the governing procedures related to engineering changes, and the requirements for performing plant change evaluations.

b. Findings

No findings were identified.

.15 B.5.b Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee's preparedness to handle large fires or explosions by reviewing selected mitigating strategies. This review ensured that the licensee continued to meet the requirements of their B.5.b related license conditions, and 10 CFR 50.54(hh)(2) by determining that:

- Procedures were being maintained and adequate;
- Equipment was properly staged, maintained, and tested;
- Station personnel were knowledgeable and could implement the procedures; and
- Additionally, inspectors reviewed the storage, maintenance, and testing of B.5.b related equipment.

The inspectors reviewed the licensee's B.5.b related license conditions, and evaluated selected mitigating strategies to ensure they remain feasible in light of operator training, maintenance/testing of necessary equipment and any plant modifications. In addition, the inspectors reviewed previous inspection reports for commitments made by the licensee to correct deficiencies identified during performance of Temporary Instruction 2515/171, or subsequent performances of these inspections.

The B.5.b mitigating strategies selected for review during this inspection are listed below. The offsite and onsite communications, notifications/emergency response organization activation, initial operational response actions and damage assessment activities identified in Table A.3 1 of NEI 06-12, "B.5.b Phase II and III Submittal Guidance," Revision 2, are evaluated each time due to the mitigation strategies' scenario selected.

NEI 06-12, Revision 2, Section	Licensee Strategy (Table)
2.3.2	Spent Fuel Pool External Spray (Table A.2-3)
3.4.1	Manual Operation of Reactor Core Isolation Cooling (Table A.5-1)

b. Findings

One finding was identified which is discussed in Inspection Report 05000331/2015404.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

a. Inspection Scope

The inspectors reviewed the licensee's CAP procedures, and samples of corrective action documents to verify that the licensee was identifying issues related to the Fire Protection Program at an appropriate threshold, and entering them in the CAP. The inspectors reviewed selected samples of condition reports, design packages, and fire protection system non-conformance documents.

b. Findings

No findings were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On February 13, 2015, the inspectors presented the inspection results to Mr. T. Vehec, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Archibald, Fire Marshal
J. Chaisson, Operations Support
D. Church, Program Engineering Manager
M. Davis, Licensing and EP Manager
R. Hanson, Fire Protection Engineer
C. Hill, Training Manager
B. Hopkins, PRA Engineer
K. Kleinheinz, Engineering Director
J. Kuehl, Program Engineer
G. Pry, Plant General Manager
R. Severson, PRA Engineer
T. Vehec, Site Vice President
T. Weaver, Licensing Engineer

U.S. Nuclear Regulatory Commission

R. Daley, Branch Chief, EB3
L. Haeg, Senior Resident Inspector
A. Klein, Branch Chief, NRR
M. Shuaibi, Deputy Division Director, DRS
J. Steffes, Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000331/2015008-01	NCV	Failure to Identify and Evaluate the Effects of Vessel Overfill Scenario (Section 1R05.6.b)

Discussed

None

LIST OF DOCUMENTS REVIEWED

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

CALCULATIONS

Number	Description or Title	Date or Revision
702-N-004	Room Temperature and Steel Temperature Transient Response to Fires	4
CAL-E08-006	AC Coordination	1
CAL-E08-007	250 VDC System Battery Sizing, Voltage Drop, Short Circuit, Coordination and Charger Sizing	0
CAL-M05-005	DAEC Control Building Heatup Analysis with GOTHIC	0
CAL-M06-007	DAEC Room Heat Up Analysis for DAEC during Station Blackout	1
FHA-200	NFPA 805 Fire Protection Design Document	16
FHA-500	Nuclear Safety Capability Assessment	8
FPLDA013-PR-011	At-Power Analysis for Fire Area CB3	0
FPLDA013-PR-016	At-Power Analysis for Fire Area PH-1	0

CORRECTIVE ACTION PROGRAM DOCUMENTS ISSUED DURING INSPECTION

Number	Description or Title	Date or Revision
2018050	2015 FP Triennial – Material was Not Removed from Cable Spreading Room	January 13, 2015
2018094	2015 FP Triennial FPE-B06-002 Typographical Error	January 13, 2015
2018259	2015 FP Triennial – Cleanliness of Cable Trays	January 14, 2015
2018314	Access to SAMP Equipment Partially Blocked	January 14, 2015
2018342	SAMP Flashlights Not Working Correctly	January 14, 2015
2018350	2015 FP Triennial – Outlets in Alternate Fire Brigade Building	January 14, 2015
2018412	2015 FP Triennial Drawing and DEI Database Error	January 14, 2015
2018457	Batteries in B.5.b Flashlight Need Replaced	January 14, 2015
2018466	B.5.b Boxes with Broken Seals	January 14, 2015
2018478	2015 FP Triennial – Access to B.5.b Equipment Blocked	January 14, 2015
2018598	Batch Change OP-025 and OP-025	January 15, 2015
2018694	2015 FP Triennial – FHA-400 Typographical Error in Reference	January 15, 2015
2018807	NSPEO Unable to Perform a Step in SAMP 703	January 15, 2015

CORRECTIVE ACTION PROGRAM DOCUMENTS ISSUED DURING INSPECTION

Number	Description or Title	Date or Revision
2018809	2015 FP Triennial – Confusing Step in SAMP 703	January 15, 2015
2018965	2015 FP Triennial Command and Control Issues B.5.b Scenario	January 16, 2015
2021159	2015 FP Triennial: Sheet Metal in Cable Spreading Room	January 27, 2015
2021323	2015 FP Triennial Door – 240 Contacts Railing	January 28, 2015
2021333	2015 FP Triennial – Incorrect Label on 1C388 Doors	January 28, 2015
2021364	AOP 915 – Shutdown Outside Control Room	January 28, 2015
2021700	Hole in Top of Panel 1D10 – NRC Identified	January 29, 2015
2021716	Cable Seals Separated into 1D11/1D13	January 29, 2015
2023571	UPS in Diesel Fire Pump Room	February 9, 2015
2024865	2015 FP Triennial – Detector Response Time for CSR	February 11, 2015
2024869	2015 FP Triennial Inspection – Vessel Overfill Impacts	February 11, 2015
2025187	2015 FP Triennial – Fire Vulnerability with Startup XFMR OOS	February 12, 2015

CORRECTIVE ACTION PROGRAM DOCUMENTS REVIEWED

Number	Description or Title	Date or Revision
1801920	Determine Modification Approach for Damper SR Batteries	September 11, 2012
1837296	Missed FPLCO and Missed FPSR For Fire Damper 1VFD-135	January 7, 2013
1892615	Cable Program Improvement Items	July 29, 2013
1935816	1D3301 and 1D3311 Smoke Damper PMs not Performed	January 24, 2014
1972543	Smoke Damper1V-SD-033 Impairment	June 17,2014
1977346	Question Regarding 1931545 Functionality Assessment	July 10, 2014
2020097	Gaps to NFPA 805 Monitoring Program	January 22, 2015

DRAWINGS

Number	Description or Title	Date or Revision
BECH-A055	Door Details	13
BECH-A057	Architectural Door Schedule	10
BECH-E021	Schematic Meter & Relay Diagram 4.16KV Bus 1A1 & Auxiliary Transformer	33
BECH-E022	Schematic Meter & Relay Diagram 4.16KV Bus 1A2 & Startup Transformer	24

DRAWINGS

Number	Description or Title	Date or Revision
BECH-E023	Schematic Meter & Relay Diagram 4160V System Essential SWGR 1A3 & 1A4	35
BECH-E027	Single Line Meter & Relay Diagram 125V DC System	33
BECH-E103(006)	Main Aux. Power, Start-up, Standby Transformer & Isolated Phase Bus	4
BECH-E104(011A)	4160V & 480V System Control & Protection	6
BECH-E104(013)	4160V & 480V System Control & Protection	14
BECH-E104(013A)	4160V & 480V System Control & Protection	5
BECH-E112	Alternate Shutdown Capability System	3
BECH-E112(033)	Alternate Shutdown Capability System	4
BECH-E121(041)	Reactor Core Cooling Systems	10
BECH-E122(002A)	Nuclear Steam Supply Shutoff System	9
BECH-E128(004)	480V Feeders	25
BECH-E511(012L)	Protective Relay Coordination Curves Diesel Generator (DG1&DG2) OC Relay 151 & 151V	3
BECH-M161	P&ID Air Conditioning System Control Building	53
FHA=M-02(1)	Fire Hazard Analysis Plan At ELEV 757'6"	34
FHA=M-02(2)	Fire Hazard Analysis Plan At ELEV 757'6"	24
FHA=M-15(1)	Fire Hazard Analysis Pump House Plans	6
FHA=M-15(2)	Fire Hazard Analysis Pump House Plans	3
M107-001(1)	Fire & Smoke Detection System	12
M107-001(2)	Fire & Smoke Detection System	4

EVALUATIONS

Number	Description or Title	Date or Revision
FPE-B06-002	Evaluation of Fire Barrier Between Fire Zones 11A and 12A	1
FPE-M13-002	NFPA 600 Code Conformance Evaluation	0
FPE-R13-004	Fire Area CB3 Nuclear Safety Capability Assessment	0
FPE-R13-009	Fire Area PH1 Nuclear Safety Capability Assessment	0
FPE-R13-011	Fire Area RB1 Nuclear Safety Capability Assessment	0
FPE-R13-018	NFPA 805 Monitoring Program Screening Report	0
FPE-R96-002	Credibility Evaluation of Adverse Effects on Credited Safe Shutdown Equipment From Non-Credited Safe Shutdown Equipment Failures within the Same Function Code	1
FPE-R96-004	Feasibility of Operator Manual/Recovery Actions and Verification of Alternate Shutdown Time Constraints	13

EVALUATIONS

Number	Description or Title	Date or Revision
FPE-R98-001	Evaluation of Fire Effects on Instrument Tubing and Indications	4
FPE-S02-001	Fire Detection Code Compliance Evaluation for Fire Plan Required and Fire PRA Higher Risk Areas	4

PROCEDURES

Number	Description or Title	Date or Revision
7884-13.1	Cardox Fire Protection System	1
ACP 103.10	Control of Time Critical Tasks	7
ACP 1412.2	Control of Combustibles	42
ACP 1412.4	Impairments to Fire Protection Systems	73
ACP 1412.5	NFPA 805 Monitoring Program	1
AOP 913	Fire	77
AOP 915	Shutdown Outside Control Room	54
EN-AA-105-1000	PRA Configuration Control and Model Maintenance	3
EN-AA-202-1004	Fire Protection Screening	1
FP-AA-104-1002	Fire Protection Configuration Control	0
FP-AB-100	Fire Protection Program	2
HPP 3106.04	Inspection, Maintenance and Quality Assurance of Respiratory Protection Equipment	29
OI 513	Fire Protection	115
OM-AA-101-1000	Shutdown Risk Management	9
OM-AA-101-1000(DAEC)	Shutdown Risk Management	15
PFP-CB-757	Pre-Fire Plan Control Building EL. 757	1
PFP-CB-772	Pre-Fire Plan Control Building EL. 772	1
PFP-PH-757	Pre-Fire Plan Pump House EL. 757	1
PFP-RB-757	Pre-Fire Plan Reactor Building EL. 757	1
PFP-RR-001	Pre-Fire Plan Radioactive Release	0
PFP-SOP-001	Pre-Fire Plan –Standard Operating Procedures	2
STP NS13D002-A	CO_2 Cardox System Functional Test	25

REFERENCES

Number	Description or Title	Date or Revision
Expert Panel Minutes	NFPA-805 Monitoring Vs. MR	December 19, 2013

VENDOR DOCUMENTS

Number	Description or Title	Date or Revision
0027-0042-000-002	Expert Panel for Addressing Multiple Spurious Operations	3

VENDOR DOCUMENTS

Number	Description or Title	Date or Revision
0027-0056-018-001	DAEC Change Evaluation Risk Analysis	0
GE-NE-A22-00100-50-01	DAEC Asset Enhance Program. Task T0611: Appendix R Fire Protections	9
M-1001	Johnson M-1001 and M-1010 Ionization Detectors	Not Available

WORK ORDERS

Number	Description or Title	Date or Revision
4018460 01	STP NS13P003"B"Fire Damper Internal Inspection & Function	April 17, 2013
40295799 01	Monthly Lighting Inspection	December 15, 2014
40297360 01	STP NS13B001 Semi Annual Fire Door Inspection	December 30, 2014
40300374 01	STP NS13B001 Fire Hose Station Inspection	January 7, 2015

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access and Management System
AOP	Abnormal Operation Procedure
AR	Action Request
BWROG	Boiling Water Reactor Owners' Group
CAP	Corrective Action Program
CDF	Core Damage Frequency
CFR	*Code of Federal Regulations*
DAEC	Duane Arnold Energy Center
DRS	Division of Reactor Safety
HPCI	High Pressure Coolant Injection
IP	Inspection Procedure
IMC	Inspection Manual Chapter
LOCA	Loss of Coolant Accident
MSO	Multiple Spurious Operation
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
NSCA	Nuclear Safety Capability Assessment
PARS	Publicly Available Records System
PRA	Probabilistic Risk Assessment
RCIC	Reactor Core Isolation Cooling
RPV	Reactor Pressure Vessel
SSD	Safe Shutdown
SDP	Significance Determination Process
SRA	Senior Reactor Analyst
SRV	Safety Relief Valve

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Sincerely,

/RA/

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

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