UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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October 13, 2016

EA-09-313

Mr. Joel Gebbie
Senior VP and Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: DONALD C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2 - TRIENNIAL FIRE PROTECTION INSPECTION REPORT 05000315/2016009; 05000316/2016009

Dear Mr. Gebbie:

On September 2, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed a Triennial Fire Protection Inspection at your Donald C. Cook Nuclear Power Plant, Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on September 2, 2016, with Mr. S. Lies 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. Additionally, a licensee identified violation is listed in Section 4OA7 of this report.

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 Donald C. Cook 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/

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

Docket Nos. 50–315; 50–316
License Nos. DPR–58; DPR–74

Enclosure:
IR 05000315/2016009; 05000316/2016009

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No: License No:	50-315; 50-316 DPR-58; DPR-74
Report No:	05000315/2016009; 05000316/2016009
Licensee:	Indiana Michigan Power Company
Facility:	D.C. Cook Nuclear Power Plant, Units 1 and 2
Location:	Bridgman, MI
Dates:	August 2 – September 2, 2016
Inspectors:	A. Dahbur, Senior Reactor Inspector I. Hafeez, Reactor Inspector D. Szwarc, Senior Reactor Inspector (Lead)
Accompanying Personnel:	H. Barrett, Senior Fire Protection Engineer L. Kozak, Senior Reactor Analyst
Approved by:	Robert C. Daley, Chief Engineering Branch 3 Division of Reactor Safety

SUMMARY

Inspection Report 05000315/2016009, 05000316/2016009; 08/02/2016 – 09/02/2016; Donald C. Cook Nuclear Power Plant, Units 1 and 2; 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 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 0609, "Significance Determination Process," dated April 29, 2015. Cross-cutting aspects were determined using Inspection Manual Chapter 0310, "Aspects Within the Cross Cutting Areas." Findings for which the Significance Determination Process 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 February 4, 2015. 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.

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very-low safety significance (Green) and an associated Non-Cited Violation of license conditions 2.C(4) and 2.C(3)(o) for the licensee's failure to implement the approved. Specifically, the licensee failed to analyze the double-break circuits design for valves using risk-informed, performance-based techniques for several fire areas. In the event of a fire in several fire areas, fire induced-circuit failures (i.e., inter-cable shorting) for a double-break design for several valves (i.e., Power Operated Relief Valves) could potentially result in spurious operation of the valves. The circuit analysis for these valves in these areas was analyzed using the deterministic approach instead risk-informed, performance-based techniques. The licensee entered the issue into their Corrective Action Program and took credit for existing fire protection features and controls as compensatory measures and planned to review the multiple spurious operations Expert Panel Report and properly disposition the scenario.

  The performance deficiency was determined to be more-than-minor because if left uncorrected, it would potentially lead to a more significant safety concern. Specifically, the failure to properly evaluate and disposition all potential fire-induced circuit failures for all cables in a fire area could impair the plant's ability to safely shutdown in the event of a fire. The performance deficiency was also associated with the Mitigating Systems cornerstone. The finding was of very-low safety significance because it did not impact the reactor's ability to reach and maintain a safe shutdown condition. This finding did not have a cross-cutting aspect because it was not representative of current licensee performance. (Section 1R05.6.b)

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 of the *Code of Federal Regulations* (CFR), Part 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 (SSD) 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 SSD 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 SSD systems for selected risk-significant fire areas. Inspector emphasis was placed on determining that the post-fire SSD capability and the fire protection features were maintained free of fire damage to ensure that at least one post-fire SSD 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 and 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 one inspection sample, respectively, as defined in IP 71111.05XT.

Fire Area	**Fire Zone**	**Description**
AA22	17G	Unit 2 East Motor Driven Auxiliary Feedwater Pump Room (Deterministic)
AA32	29A, 29B, 29E, 29G	Unit 1 Essential Service Water Pump Area and Unit 1 and Unit 2 Basement Motor Control Center Room (Deterministic)
AA40	41	Unit 1 Engineered Safeguards Systems and Motor Control Center Room (Performance-Based)
AA45A	47A	Unit 2 AB Switchgear Room (Performance-Based)

.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 safety analysis, NSCA, and supporting drawings and documentation to verify that SSD capabilities were properly protected.

b. Findings

No findings of significance 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 (SERs), 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 of significance 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 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.

b. Findings

No findings of significance 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.

b. Findings

No findings of significance 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 SSD, and the corresponding equipment necessary to perform those functions were included in the fire response AOPs. The licensee did not credit any primary control stations outside the main control room. The inspectors reviewed Procedure 1OHP-4025-001-001, "Emergency Remote Shutdown," Revision 13, to assess whether safe and stable plant

conditions could be implemented outside the main 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. In addition, the inspectors conducted interviews with operators and other station personnel.

b. Findings

No findings of significance 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

Inadequate Resolution for Double-Break Circuits Design for Several Valves

Introduction: The inspectors identified a finding of very-low safety significance (Green) and an associated Non-Cited Violation (NCV) of the Donald C. Cook Nuclear Power Plant Operating License for the licensee’s failure to implement the Fire Protection Program as approved in the SER. Specifically, the licensee failed to correctly analyze a double-break design for multiple valves in several fire areas using performance-based analysis techniques and instead incorrectly used the deterministic approach.

Description: The pressurizer is equipped with two types of devices for pressure relief, pressurizer safety valves and power-operated relief valves (PORVs). The PORVs are air operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room. Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident. As such, block valve closure terminates the reactor coolant system depressurization and coolant inventory loss.

The PORVs 1(2)-NRV-151, 152, and 153 at the Cook Nuclear Plant are of an ungrounded direct-current, double-break circuit design. The licensee stated in analysis PRA-FIRE-17663-0002b-LAR, "DC Cook MSO Expert Panel Report," that the PORVs and the associated block valves could potentially simultaneously be impacted by a fire in 17 areas of Units 1 and 2 combined. The simultaneous failure of the pressurizer PORVs and block valves were evaluated in the PRA-FIRE-17663-0002b-LAR analysis. The licensee credited actions to remove control power fuses at the control room panel, or to de-energize the control room panel from outside the control room, in order to mitigate the spurious operation of the PORVs for all cases except where failure of the PORVs/Block Valves was determined by the Fire Probabilistic Risk Assessment (PRA) not to occur at a scenario level. For this exception, the failure of the pressurizer PORVs variance from the deterministic requirement was evaluated and it was determined that the risk, safety margin and defense-in-depth met the acceptance criteria of NFPA 805 Section 4.2.4 with no further action required. For the remaining areas, control room action or recovery action to remove power to the Pressurizer PORV control circuits was determined by the licensee to be feasible and resulted in mitigation of the potential spurious operation.

The License Amendment Request (LAR) for Cook Nuclear Plant, NFPA 805 Transition Report, Attachment B, NEI 04-02 Table B-2 – Nuclear Safety Capability Assessment Methodology Review, for attribute 3.5.1.5 of NEI 00-01, "Circuit Analysis Criteria/Assumptions," indicated that circuits which could cause the undesired spurious actuation of SSD components were identified in the SSD System Analysis. In order for a spurious actuation to occur, various conditions must exist synergistically at the cable fault location, including a jacket material type and number of conductors. The licensee took credit for a design standard that required that the control switch and relay contacts "double-break" the positive and negative control leads for components which spurious operation could affect SSD (e.g., solenoid and motor-operated valves). The implementation of this design standard for these control circuits (250-Vdc) at Cook Nuclear Plant prevents single cable-to-cable faults from initiating spurious operation. The licensee stated in the NFPA 805 Transition Report that:

Although the approach was reviewed and approved by the NRC in the November 22, 1983, Safety Evaluation for Appendix R, the "double-break" design credited under Appendix R was not be carried forward in the transition to the new licensing basis. Circuit failures identified as a result of inclusion of new cables added to the NFPA 805 analysis for these double-break valves will be captured, analyzed and addressed using risk-informed, performance-based techniques.

The revised circuit analysis process and methods for addressing fire induced cable damage and MSOs are documented in approved project procedures and being incorporated into the NFPA 805 transition. The process includes considering the effects of fire damage on both thermoplastic and thermoset cable, any possible combination of conductors shorting within intra-cable and does not limit the number of cables when addressing spurious operations due to inter-cable shorting."

Consistent with the licensee LAR, the NRC SER, dated October 24, 2013, for this attribute 3.5.1.5, indicated that the process used by the licensee considered the effects of fire damage on both thermoplastic and thermoset cable, any possible combination of conductors shorting within intra-cable and did not limit the number of cables when addressing spurious operations due to inter-cable shorting. Although under this attribute the licensee described the previous licensing basis consideration of the double-break design at Cook Nuclear plant, which precluded the need to consider multiple cable-to-cable hot shorts, the approach used during transition considered the possibility of spurious actuation using risk-informed, performance-based analysis techniques. Based on this discussion, the NRC staff concluded that the methods as described by the licensee were sufficiently similar to the specific methods in NEI 00-01.

While reviewing control room actions associated with the pressurizer PORVs in several fire areas (i.e., Fire Area AA7, Unit 1 Quadrant 1 Cable Tunnel) in the Fire Response Guidelines Procedure 12-OHP-4025-001-002, the inspectors noticed that the licensee indicated that cable damage in this Fire Area could result in spurious operation of Pressurizer PORVs and credited control room actions to remove control power fuses to close the valves. The inspectors also noticed that the Fire Safety Analysis (FSA), Revision 1, concluded that the nuclear safety and radioactive release performance criteria of NFPA 805 were met for a fire in Fire Area AA7. The licensee concluded in the FSA that for Fire Area AA7 a success path remains free of fire damage using the deterministic approach per Section 4.2.3 of NFPA 805. The licensee did not identify a variance from deterministic requirements for this fire area. The inspectors were concerned that transitioning of Fire Area AA7 using the deterministic approach did not meet the circuit analysis approach as described in the licensee's LAR and the NRC SER for double-break circuits, specifically, for the PORV circuits. The inspectors determined that the licensee failed to address the potential of spurious operations due to inter-cable shorting after completing the action to remove the fuses to close the PORV. Per the LAR and the SER, the PORV circuits analysis should have considered the possibility of spurious actuation due to multiple cable-to-cable hot shorts and Fire Area AA7 should have been analyzed using the risk-informed, performance-based approach instead of the deterministic approach.

The licensee entered this issue into their Corrective Action Program (CAP) as Action Request (AR) 2016-9854 and concluded that the existing fire detection, manual suppression and defense-in-depth Fire Protection Program administrative controls were considered acceptable compensatory measures.

Analysis: The inspectors determined that the licensee's failure to implement the Fire Protection Program as approved in the SER was contrary to the Donald C. Cook Nuclear Power Plant Operating License conditions for the Fire Protection Program and was a performance deficiency.

The performance deficiency was determined to be more-than-minor because if left uncorrected, it would potentially lead to a more significant safety concern. Specifically, the failure to correctly consider all potential fire induced circuit failures for all cables in a fire area could potentially affect the Fire Protection Program if those potential circuit failures were not properly evaluated and dispositioned. The performance deficiency was also associated with the Mitigating Systems cornerstone and affected the 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 (i.e., core damage).

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 Systems 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 1.3.1, "Ability to Achieve Safe Shutdown." The inspectors answered "yes" to Question 1.3.1, "Is the reactor able to reach and maintain safe shutdown (either hot or cold) condition?"

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. The licensee completed the analysis for these fire areas in 2011 during the transition to NFPA 805.

<u>Enforcement</u>: License conditions 2.C(4) and 2.C(3)(o) of the Donald C. Cook Nuclear Power Plant, Unit 1 and Unit 2, Operating Licenses, respectively, require, in part, that the licensee implement and maintain in effect all provisions of the approved Fire Protection Program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee's amendment request dated July 1, 2011, as supplemented and as approved in the Safety Evaluation dated October 24, 2013.

The licensee indicated in Attachment B of the License Amendment Request, "NFPA 805 Transition Report," dated July 1, 2011, in part, that, "circuit failures identified as a result of inclusion of new cables added to NFPA 805 analysis for the double-break valves will be captured, analyzed and addressed using risk-informed, performance-based techniques. The process includes considering the effects of fire damage on both thermoplastic and thermoset cable, any possible combination of conductors shorting within intra-cable and does not limit the number of cables when addressing spurious operations due to inter-cable shorting."

Section 3.2.1.2 of the SER dated October 24, 2013, also indicated that, "the process used by the licensee considered the effects of fire damage on both thermoplastic and thermoset cable, any possible combination of conductors shorting within intra-cable and does not limit the number of cables when addressing spurious operations due to inter-cable shorting." The SER further stated that, "The approach used during transition considered the possibility of spurious actuations using risk-informed, performance-based analysis techniques."

Contrary to the above, from September 10, 2011, until September 2, 2016, the licensee failed to implement all provisions of the approved Fire Protection Program. Specifically, in the event of a fire in several fire areas, the licensee failed to correctly analyze the double-break design for multiple valves (i.e., PORV) as specified in the licensee's amendment request. The licensee analyzed the double-break design using a deterministic approach instead of the risk-informed, performance-based analysis techniques that they committed to and which were approved in the SER.

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 2016-9854. The licensee took credit for existing fire protection features and controls as compensatory measures and planned to review the MSO Expert Panel Report and properly disposition the scenario. (NCV 05000315/2016009-01; 05000316/2016009-01, Inadequate Resolution for Double-Break Circuits Design for Several Valves)

.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. In the event of a fire the licensee initially credits the plant public address system and control room intercom units for notification only. The plant radio communications system is credited to supplement communications necessary to perform the majority of post-fire activities. The licensee did not evaluate the plant public address system and sound powered phones to be free of fire damage for fires in various areas. As a result the licensee took credit for face to face and radio communications. The inspectors reviewed a sample of actions that would utilize face to face and radio communications. The inspectors verified that radios were available for use and maintained in working order.

b. Findings

No findings of significance 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. The licensee credited the existing fixed emergency lighting system together with portable emergency lighting. The inspectors verified that the portable emergency lighting units were being properly tested and maintained. The inspectors verified that the portable emergency lighting units would be sufficient 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 of significance 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 of significance 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 of significance 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 of significance 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 of significance 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 inspectors review that 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

A licensee identified violation is discussed in Section 4OA7.

.14 Plant Change Evaluation

a. Inspection Scope

The inspectors reviewed a sample of 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. Additionally, the inspectors reviewed the governing procedures related to engineering changes, and the requirements for performing plant change evaluations.

b. Findings

No findings of significance 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 strategy selected for review during this inspection is 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)**
3.3.4	Manually Depressurize Steam Generators and Use Portable Pump (Table A.4-4)

b. Findings

No findings of significance were identified.

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 of significance were identified.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

.1 (Reviewed) Licensee Event Report 05000315/2008-02-00: 250Vdc Cable Separation Criteria for 10 CFR Part 50, Appendix R Not Met

The inspectors documented the review and closure of Licensee Event Report (LER) 05000315/2008-02-00 in Inspection Report 05000315/2009006; 05000316/2009006 (ADAMS ML093500625). The concern documented in the LER was related to an Appendix R cable separation issue and the NRC granted enforcement discretion based on the criteria specified in "Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)," published in Federal Register Notices dated June 16, 2004, January 14, 2005, and April 18, 2006. One of the criteria for receiving enforcement discretion was that the licensee would correct the issue during the transition to 10 CFR 50.48(c) (NFPA 805). During the current inspection the inspectors verified that the licensee had corrected the issue and that the basis for the original disposition and closure was valid. The inspectors reviewed the licensee's corrective actions, drawings, documents, interviewed licensee staff, and performed walkdowns. Documents reviewed as part of this inspection are listed in the Attachment.

This LER was previously closed. This report documents the NRC's review of the licensee's corrective actions associated with the concern that lead to the issuance of the original LER and the NRC's granting of enforcement discretion in closing that LER (EA-09-313).

This review does not constitute a sample as defined in IP 71153-05.

4OA6 Management Meetings

.1 Exit Meeting Summary

On September 2, 2016, the inspectors presented the inspection results to Mr. S. Lies, 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.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as an NCV.

- The licensee identified a finding of very-low safety significance (Green) and associated NCV of License Conditions 2.C.4 and 2.C.3.o for Units 1 and 2 respectively for the licensee's failure to establish an appropriate Monitoring Program in accordance with NFPA 805, Section 2.6. Section 2.6 of NFPA 805 required, in part, that monitoring shall ensure that the assumptions in the engineering analysis remain valid. Contrary to the above, the licensee failed to ensure that the assumptions in the engineering analysis remained valid for the

availability and reliability of the auxiliary feedwater pumps in the Monitoring Program. The licensee used Maintenance Rule availability criteria to monitor the auxiliary feedwater pumps which did not bound the Fire PRA assumptions for the unavailability of the components.

The performance deficiency was determined to be more-than-minor because the issue adversely impacted the Mitigating Systems cornerstone objective to ensure the capability of systems that respond to initiating events and prevent undesirable consequences due to external events such as fire. Specifically, the failure to adequately monitor plant equipment credited for post-fire SSD could result in that equipment being unavailable for longer periods of time than had been analyzed. The inspectors screened the finding using Inspection Manual Chapter 0609, “Significance Determination Process,” Appendix F, “Fire Protection Significance Determination Process.” Since the reactor was still able to reach and maintain a SSD condition, the finding screened as very-low safety significance (Green). The licensee entered the issue into the CAP as AR 2016-7239, “NFPA 805 Monitoring Program FPRA\Maintenance Performance,” and revised Maintenance Rule administrative procedures to consider the unavailability criteria of components and the impact on the fire PRA.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

J. Gebbie, Chief Nuclear Officer
S. Lies, Site Vice President
A. Lloyd, Operations Fire Protection Manager
M. Lloyd, Vice President Engineering
D. MacDougall, Fire Protection Program Manager
S. Mitchell, Nuclear Regulatory Affairs Supervisor
R. Pletz, Fire Protection Engineer
J. Ross, Plant Manager
M. Scarpello, Nuclear Regulatory Affairs Manager
J. Turner, Fire Marshall
R. Wynegar, Nuclear Regulatory Affairs

U.S. Nuclear Regulatory Commission

R. Daley, Branch Chief, EB3
J. Ellegood, Senior Resident Inspector
M. Shuaibi, Deputy Division Director, DRS
T. Taylor, Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000315/2016009-01; 05000316/2016009-01	NCV	Inadequate Resolution for Double-Break Circuits Design for Several Valves (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	Revision
PRA-FIRE-17663-0012-LAR	DC Cook Fire PRA Human Reliability Analyses	1
PRA-FIRE-17663-011B-LAR	DC Cook Fire PRA Main Control Room Analyses [Unit 1 and Unit 2]	1
PRA-FIRE-17663-701-AA40	Fire Risk Evaluation – Fire Area AA40	0
PRA-FIRE-17663-701-AA45A	Fire Risk Evaluation – Fire Area AA45A	0
R1900-0024-003	Nuclear Safety Capability Assessment – Compartment AA40 and AA45A	0
R1900-0026-001	Recovery Action Transition in Support of NFPA 805	2

CORRECTIVE ACTION PROGRAM DOCUMENTS ISSUED DURING INSPECTION

Number	Description or Title	Date
2016-8909	12-OHP-4026-EDM-002 Validation Walkdown	08/03/2016
2016-8957	No Component Identification Label on 1-FW-257	08/04/2016
2016-8958	1-FW-142-1 Missing Fire Pre-Plans Volume 3 Tag	08/04/2016
2016-8961	12-FPP-2270-066-030	08/04/2016
2016-9281	Terminal Box Not Grounded in Accordance with 1-2-EDS-382	08/15/2016
2016-9335	Fire Pre-Plans Volume 1 – AA40 & AA43	08/17/2016
2016-9339	Enhancement to the FSA is Needed	08/17/2016
2016-9455	1/2 OHP-4025-001-001 Needs to Add a RNO	08/18/2016
2016-9463	Communications Not Using Accredited System	08/18/2016
2016-9464	1-BAT-LIT-455 Not Aimed Correctly	08/18/2016
2016-9465	Fire Extinguisher Not in Fire Pre-Plan	08/18/2016
2016-9490	1/2 OHP-4025-001-001 Procedural Enhancements	08/19/2016
2016-9760	Track Issuance of NEI 00-01 Rev. 4	08/29/2016
2016-9854	Inadequate Circuit Analysis Resolution Fire Analysis Area 7	08/31/2016
2016-9917	Installed Gaitronics Missing from Fire Pre-Plans Volume 1	09/01/2016

CORRECTIVE ACTION PROGRAM DOCUMENTS REVIEWED

Number	Description or Title	Date
2013-7967	Cable Identified in 20' Separation Area	05/30/2013
2013-8836	Inadequate Technical Evaluation 11.15 Rev. 3	06/17/2013
2014-10225	12V Diesel Trailer Switch	08/29/2014
2014-9068	Combustible Materials Being Stored in the Aux 650'	08/01/2014
2015-12275	Could Not Perform 6 Month TRM Drop Test on 2-DR-TUR260	09/18/2015
2016-1442	Recovery Action Feasibility AA40 RCP Trip	02/05/2016
2016-2392	Fire Pre-Plans Volume 1 for Fire Zone 44S	02/29/2016

CORRECTIVE ACTION PROGRAM DOCUMENTS REVIEWED

Number	Description or Title	Date
2016-7239	NFPA 805 Monitoring Program FPRA/Maintenance Performance	06/16/2016
825805	Appendix R Compliance for Train B 4kv Room Fire	02/06/2008
833844	EDMG-2	06/24/2008
833915	Fire Pre-Plan Volume III	06/24/2008
833919	Isolation of Fire Protection Pumps' Common Recirculation Line	06/25/2008

DRAWINGS

Number	Description or Title	Revision
1-1433R-1	Appendix "R" Plan of Electrical Equipment Conduits & Cable Toughs Auxiliary Building Below Elevation 625'-10"	1
OP-1-12001-86	Main Auxiliary One-Line Diagram Bus "A" & "B" Engineered Safety System	86
OP-1-12003-35	250V DC Main One-Line Diagram Engineered Safety System(Train "A, B &N" & BOP)	35
OP-1-12014-15	MCC Aux One-Line 600V Bus 11A, 11B Engineered Safety System (Train "B")	15
OP-1-12033-34	MCC Aux One-Line 600V Bus 11C, 11D Engineered Safety System (Train "A")	34
OP-1-12070-29	DC Aux One-Line 250V DC Bus CD Engineered Safety System(Train "A")	29
OP-1-12072-21	DC Aux One-Line 250V DC Bus CD Engineered Safety System(Train "A")	21
OP-1-5104C-7	Composite Flow Diagram – Engineered Safety Systems	7
OP-1-5153E-6	Flow Diagram Fire Protection CO2 Lower 4KV Areas Unit 1	6
OP-1-98201-16	Elementary Diagram – Reactor Coolant System, Sheet 1	16
OP-1-98282-24	Elementary Diagram - Emergency Core Cooling (Safety Injection) Sheet 2	24
OP-1-984151-23	Essential Service Water System West Sheet #1 Elementary Diagram	23
OP-1-98415-47	Essential Service Water System East Sheet #1 Elementary Diagram	47
OP-2-98573-39	Elementary Diagram – Emergency Plant Shutdown and Cooldown Local Indication	39
OS-1-98046-35	4KV 600V Auxiliary Transformers 11B &11D Elementary Diagram Sheet 1 of 2	35
PS-1-91311-8	Battery CD Control Panel "BC-CD" Wiring Diagram	8
PS-2-91311-7	Battery CD Control Panel "BC-CD" Wiring Diagram	7

EVALUATIONS

Number	Description or Title	Date
FPEEE 12.19	CO2 Fire Suppression Systems in Fire Zones Containing Concentrations of Cable Insulation	11/14/2014
FPPR-2015-0076	Turbine Rollup Doors 1-DR-TUR253 and 2-DR-TUR260	10/29/2015

PROCEDURES

Number	Description or Title	Revision
12-EHP-2270-FPMP-001	Fire Protection Required SSC Monitoring Program	3
12-EHP-4075-TCA-001	Operator Time Critical Actions	11
12-FPP-2270-066-030	Other Fire Protection Equipment Inspections	25
12-FPP-2270-066-037	Functional Check and Inspection of the Hale Portable Pumps	6
12-OHP-4025-001-002	Fire Response Guidelines	11
12-OHP-4026-EDM-001	Extensive Damage Mitigation Initial Response	5
12-OHP-4026-EDM-002	Extensive Damage Mitigation Enhanced Site Response Strategies, Attachment 2	4
12-OHP-4026-EDM-002	Extensive Damage Mitigation Enhanced Site Response Strategies, Attachment 4	4
1-OHP-4025-001-001	Emergency Remote Shutdown	13
1-OHP-4025-LS-1	Process Monitoring from LSI Panels	3
1-OHP-4025-LS-6	RCS Make-Up with CVCS Cross-Tie	7
1-OHP-4025-LTI-4	Local RCP Trip and Isolation	3
1-OHP-4025-LTI-5	Spurious Valve Isolation	4
1-OHP-5030-028-005	Miscellaneous Fan Checks	4
Fire Pre-Plans; Vol. 1	Fire Areas AA 22, AA32, AA 40, and AA 45A	25
Fire Pre-Plans; Vol. 3	Fire Protection Response to a Large Fire or Explosion Event, Attachment 8	21
FP-C-9018	Site Specific Hazards	3
OHI-4023	Abnormal/Emergency Procedure User’s Guide	37

OTHER DOCUMENTS

Number	Description or Title	Revision
DB-12-250V	Design Basis Document for the 250V DC System	1
1-OHP-4030-101-044	Unit One LSI Panel Surveillance	6

WORK ORDERS

Number	Description or Title	Revision
12-FPP-2270-066-011	Fire Watch Activities (dated August 11, 2016)	14

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AOP	Abnormal Operating Procedure
AR	Action Request
CAP	Corrective Action Program
CFR	*Code of Federal Regulations*
FSA	Fire Safety Analysis
IP	Inspection Procedure
LAR	License Amendment Request
LER	Licensee Event Report
MSO	Multiple Spurious Operations
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
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
SER	Safety Evaluation Report
SSD	Safe Shutdown

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Sincerely,

/RA/

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

Docket Nos. 50–315; 50–316
License Nos. DPR–58; DPR–74

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IR 05000315/2016009; 05000316/2016009

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