UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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December 26, 2006

Charles D. Naslund, Senior Vice
President and Chief Nuclear Officer
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P.O. Box 620
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SUBJECT: CALLAWAY PLANT - NRC TRIENNIAL FIRE PROTECTION INSPECTION REPORT 05000483/2006008 AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Naslund:

On November 13 through November 30, 2006, the NRC completed an inspection at your Callaway Plant. The enclosed report documents the inspection findings which were discussed on November 30, 2006, with Mr. Charles Naslund and other members of your staff.

During this triennial fire protection inspection, the inspection team examined activities conducted under your license related to safety and compliance with the Commission's rules and regulations and the conditions of your license. The inspection consisted of selected examination of procedures and records, observations of activities and installed plant systems, and interviews with personnel.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document 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/

Linda J. Smith, Chief
Engineering Branch 2
Division of Reactor Safety

Docket: 50-483
License: NPF-30

Enclosure: NRC Inspection Report 05000483/2006008
w/attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-483
License No.:	NPF-30
Report No.:	05000483/2006008
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, Missouri
Dates:	November 13 through 30, 2006
Inspectors:	J. Mateychick - Senior Reactor Inspector G. Pick - Senior Reactor Inspector R. Mullikin - Consultant
Approved By:	L. J. Smith, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000483/2006-008; November 13 - 30, 2006; Callaway Plant: Triennial Fire Protection Inspection

The report covered a 2-week period of inspection by region-based specialist inspectors and a contractor. No findings of significance were identified. The significance of most findings is indicated by its color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the Significance Determination Process does not apply may be green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

Callaway Plant formally committed to converting their Fire Protection Program to comply with the requirements of 10 CFR Part 50.48.(c) and National Fire Protection Association Standard 805. This involves using a risk-informed methodology. Because the conversion and licensing process are expected to identify and address a variety of difficult issues that are normally the subject of triennial fire protection inspections, and because any findings in this area would have to be addressed under the new, rather than the existing, program, the NRC has adapted its inspection and enforcement of certain issues for plants in this situation. As a result, the scope of this inspection was modified, and some issues raised in this inspection are documented but subject to enforcement discretion.

A. NRC-Identified and Self Revealing Findings

No findings of significance were identified during this inspection.

B. Licensee-Identified Findings

None.

REPORT DETAILS

1 REACTOR SAFETY

1R05 Fire Protection

The purpose of this inspection was to review the Callaway Plant Fire Protection Program (FPP) for selected risk-significant fire areas. The inspection was performed in accordance with Inspection Procedure (IP) 71111.05TTP, "Fire Protection-NFPA 805 Transition Period (Triennial)," dated May 9, 2006, for a plant in transition to National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition. The NRC reduced the scope of this inspection by not specifically targeting safe shutdown circuit configurations for inspection. Emphasis was placed on verification of the post-fire safe shutdown capability. The inspection was performed in accordance with the NRC regulatory oversight process using a risk-informed approach for selecting the fire areas and attributes to be inspected. The team used the Individual Plant Examination for External Events for the Callaway Plant to choose risk-significant areas for detailed inspection and review. Inspection Procedure 71111.05TTP requires selecting a minimum of three fire areas for review. The three fire areas reviewed during this inspection were:

- A-8 (Auxiliary Building - 2000' Elevation, General Area)
- A-18 (Auxiliary Building - 2026' Elevation, Electrical Penetration Room (North))
- A-29 (Auxiliary Building - 2000' Elevation, Feedwater Pump Valve Compartments)

For each of these fire areas, the inspection focused on fire protection features, systems and equipment necessary to achieve and maintain safe shutdown conditions, and licensing basis commitments.

Documents reviewed by the team are listed in the attachment.

.1 Shutdown From Outside Main Control Room

a. Inspection Scope

The team reviewed the functional requirements identified by the licensee as necessary for achieving and maintaining hot shutdown conditions to ensure that at least one post-fire safe shutdown success path was available in the event of fire in each of the selected areas and alternative shutdown for the case of control room evacuation. The team reviewed piping and instrumentation diagrams of systems credited in accomplishing safe shutdown functions to independently verify whether licensee's shutdown methodology had properly identified the required components. The team focused on the following functions that must be available to achieve and maintain safe shutdown conditions:

- Reactivity control capable of achieving and maintaining cold shutdown reactivity conditions,
- Reactor coolant makeup capable of maintaining the reactor coolant inventory,
- Reactor heat removal capable of achieving and maintaining decay heat removal,
- Supporting systems capable of providing other services necessary to permit extended operation of equipment necessary to achieve and maintain hot shutdown conditions,
- Verify that a safe shutdown can be achieved and maintained with and without off-site power.

A review was also conducted to ensure that all required components in the selected systems were included in the licensee's safe shutdown analysis. The team identified the systems required for each of the primary safety functions necessary to achieve and maintain shutdown conditions. These systems were then evaluated to identify the systems that interfaced with the selected fire areas and were the most risk significant systems required for reaching hot shutdown conditions.

b. Findings

No findings of significance were identified.

.2 Protection of Safe Shutdown Capabilities

a. Inspection Scope

For the selected fire areas/zones, the team evaluated the potential for fires, the combustible fire load characteristics, potential exposure fire severity, and the separation of systems necessary to achieve and maintain safe shutdown.

In accordance with Inspection Procedure 71111.05TTP for a plant in transition to NFPA Standard 805, the NRC reduced the scope of this inspection by only targeting safe shutdown circuit configurations for which the licensee has completed their NFPA Standard 805 assessment. Since the licensee has not completed their NFPA Standard 805 assessment for any fire area, there was no additional inspection of circuits performed.

b. Findings

Introduction. The team identified an apparent violation of License Condition 2.C.(5), "Fire Protection (Section 9.5.1.7, SER and Section 9.5.1.8, SSER #3)", concerning failure to assure safe shutdown systems are protected in accordance with the provisions of the approved fire protection program. The licensee credited manual actions to mitigate the effects of fire damage in lieu of providing the physical separation, physical protection, or an appropriate diverse means of accomplishing the safe shutdown

function, which adversely affected the ability to achieve and maintain safe shutdown in the event of a fire.

Description. Standardized Nuclear Unit Power Plant System (SNUPPS) Final Safety Analysis Report, Appendix 9.5E, provided the design comparison between the plant's fire protection program and 10 CFR Part 50, Appendix R. The comparison to Section III.G, "Fire Protection of Safe Shutdown Capability," states, "Redundant trains of systems required to achieve and maintain hot standby are separated by 3-hour rated fire barriers, or the equivalent provided by Section III.G.2, or else a diverse means of providing the safe shutdown capability exists that is unaffected by the fire." (Emphasis added)

The licensee has interpreted "diverse means" to mean by any reasonable means including local valve and breaker operations as long as they are within the scope of normal operator duties. The NRC found that the original fire protection program as approved in License Condition 2.C.(5)(c) required no local manual actions outside of the control room for fires other than a control room fire. Review of Fire Protection Procedure FPP-ZZ-00001, "Auxiliary Building Prefire Strategies," Revision 17, identified fire areas in addition to the control room where a fire would now require local manual actions in accordance with the licensee's current fire protection program. The licensee considers the use of local manual actions to be acceptable in accordance with License Condition 2.C.(5).

For plants licensed after 1979, the NRC staff does not recognize the use of manual actions used in lieu of the protection means specified in 10 CFR Part 50, Appendix R, Section III.G.2, as meeting the regulatory requirements unless those manual actions do not pose an adverse affect on the ability to achieve and maintain safe shutdown. The components being operated are identified as required for operation of safe shutdown systems or are subject to potential spurious operation impacting the shutdown. The local manual actions are being performed because of fire damage to electrical cables and are being used to compensate for damage to or maloperation of safe shutdown equipment caused by fire. The substitution of local manual actions for the physical separation or physical protection in the original approved fire protection program introduced additional vulnerabilities because of human error concerns. The NRC staff considers the current Callaway plant FPP to provide a reduced level of protection to assure post-fire safe shutdown from that in the fire protection program approved in License Condition 2.C.(5)(c) and, therefore, does not meet the criteria of License Condition 2.C.(5)(d) for an allowable change not requiring prior NRC approval.

The case of a fire in Fire Area A-8, "Auxiliary Building - El. 2000, General Area Rooms 1301, 1302, 1306, 1307, 1308, 1311, 1312, 1313, 1314, 1315, 1316, 1317, 1318, 1319, 1320, and 1321," will be used for examples of local manual actions.
Procedure FPP-ZZ-00001 divides Fire Area A-8 into Fire Zones A-8A through A-8E with a pre-plan for each fire zone in Attachments 21 and 25 through 29.

Example 1: In the current fire protection program, a fire in Fire Area A-8D, Rooms 1301, 1314, 1315 and 1320, could damage circuits for normally closed Valves EJHV8811A and EJHV8811B. Spurious opening of either valve would establish a drainage path from the refueling water storage tank (RWST) to the containment sump

diverting the supply of water credited for providing sufficient coolant inventory control and boration for cold shutdown. The licensee must recognize the abnormal lowering of the RWST level and perform a local manual action to close Valve EJHV8812A or EJHV8812B, as required, to respond to this spurious actuation instead of providing physical separation or physical protection of the circuits to one of the two valves in each train.

Example 2: A fire in Fire Area A-8D could spuriously open Atmospheric Steam Dump Valve ABPV0001 with a loss of control from the control room. In the current fire protection program, an operator can locally close the valve and instructions are provided for a control room operator to determinate a wire in a control room back panel to restore control.

Example 3: A fire in Fire Area A-8D could spuriously open Atmospheric Steam Dump Valve ABPV0003 with a loss of control from the control room. In the current fire protection program, an operator is directed to manually close the valve using its local manual control station.

In the fire protection program originally approved in License Condition 2.C.(5)(c), SNUPPS Final Safety Analysis Report, Appendix 9.5B, "Fire Hazards Analysis," Section A.8.7.2, "Safe Shutdown Capability," the evaluation of Fire Area A-8 did not identify these issues or deviations for the use of local manual actions. Therefore, the use of these manual actions were not approved in the original fire protection program as defined in License Condition 2.C.(5)(c). The NRC considers this change to adversely affect the ability to achieve and maintain safe shutdown in the event of a fire because of the introduction of additional vulnerabilities related to human error not present in the fire protection program approved in License Condition 2.C.(5)(c) because of the nature of the actions and time allotted for completion. Therefore, the current fire protection program does not meet the criteria of License Condition 2.C.(5)(d) for an allowable change not requiring prior NRC approval.

Analysis. This finding is of greater than minor safety significance because it impacted the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to external events (such as fire) to prevent undesirable consequences. The team reviewed Procedure FPP-ZZ-00001 and stepped through the manual actions directed in the procedure with licensee operations personnel for the sample fire areas selected for inspection. The team found that the manual operator actions were reasonable (as defined in Enclosure 2 of IP 71111.05TTP), could be performed within the analyzed time limits assuming prompt recognition of the condition by control room operators and could be credited as part of or in whole as a compensatory measure. Since the manual operator actions were considered reasonable as interim compensatory measures, the significance determination process was not entered.

Enforcement. Callaway Plant License Condition 2.C.(5) states:

(c) The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the SNUPPS Final Safety Analysis Report for the facility through Revision 15, the Callaway

site addendum through Revision 8, and as approved in the SER through Supplement 4, subject to provision d below.

(d) The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

SNUPPS Final Safety Analysis Report, Appendix 9.5E, provided a comparison between 10 CFR Part 50, Appendix R, Section III.G, "Fire Protection of Safe Shutdown Capability," and the plant design. The SNUPPS response stated:

> Final Safety Analysis Report, Appendix 9.5B, provides an area-by-area analysis of the SNUPPS power block that demonstrates that no single fire can prevent safe shutdown.
>
> Redundant trains of systems required to achieve and maintain hot standby are separated by 3-hour rated fire barriers, or the equivalent provided by Section III.G.2, or else a diverse means of providing the safe shutdown capability exists and is unaffected by the fire.

The original fire protection program as approved in License Condition 2.C.(5)(c) required no local manual actions outside of the control room to achieve hot shutdown conditions for fires other than a control room fire. This was confirmed by the NRC in the Callaway plant SER, Supplement 3, which states, in part, "The applicant's fire hazards analysis demonstrated that, except for inside containment and inside the control room, redundant systems and cabling needed for safe shutdown are separated in accordance with Section C.5.b of BTP CMEB 9. 5-1," and "The applicant's analysis indicated that the only area outside containment where redundant divisions are not separated by barriers in accordance with Section C.5.b of BTP CMEB 9.5-1 is the control room." Section C.5.b of BTP CMEB 9.5-1, "Safe Shutdown Capability," requires, in part, "One train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station(s) is free of fire damage."

Contrary to the above, the licensee implemented changes to the approved fire protection program without prior approval of the Commission, such as the examples cited above, which adversely affected the ability to achieve and maintain safe shutdown in the event of a fire. Specifically, the licensee has implemented a methodology that utilizes local manual operator actions outside of the control room to mitigate the effects of fire damage in lieu of providing physical separation or physical protection from fire damage without demonstrating no adverse affect on achieving and maintaining safe shutdown and without obtaining Commission approval. The licensee had entered this issue into their corrective action program under Callaway Action Request 200608640.

Because the licensee committed, prior to December 31, 2005, to adopting NFPA Standard 805 and changing their FPP license basis to comply with 10 CFR Part 50.48.(c), this issue is covered by enforcement discretion in accordance with the NRC Enforcement Policy. Specifically, this issue would have been expected to be identified and addressed during the licensee's conversion to NFPA Standard 805,

was entered into the licensee's corrective action program and will be corrected, and was of very low safety significance. The manual actions are to remain in effect as compensatory measures until the issue is resolved and compliance restored. The team's review concluded that this violation meets the criteria for enforcement discretion for plants in transition to a risk-informed, performance-based fire protection program as allowed per 10 CFR Part 50.48(c). Since all the criteria were met, the NRC is exercising enforcement discretion for this issue.

.3 Passive Fire Protection

a. Inspection Scope

For the selected fire areas, the team evaluated the adequacy of fire area barriers, penetration seals, fire doors and electrical raceway fire barriers. The team observed the material condition and configuration of the installed barriers, seals and doors. The team compared the as-installed configurations to the approved construction details and supporting fire tests. In addition, the team reviewed license documentation, such as inspection procedures and surveillance records, to verify that the passive fire protection features met license commitments.

b. Findings

No findings of significance were identified.

.4 Active Fire Protection

a. Inspection Scope

For the selected fire areas, the team evaluated the adequacy of fire suppression and detection systems. The team observed the material condition and configuration of the installed fire detection and suppression systems. The team reviewed design documents, inspection procedures, surveillance records and fire impairments. In addition, the team reviewed license basis documentation, such as NRC safety evaluation reports, and deviations from NRC regulations to verify that fire suppression and detection systems met license commitments.

b. Findings,

No findings of significance were identified.

.5 Protection From Damage From Fire Suppression Activities

a. Inspection Scope

For the sample areas, the team verified that redundant trains of systems required for hot shutdowns were not subject to damage from fire suppression activities or from the rupture or inadvertent operation of fire suppression systems including the effects of flooding.

b. Findings

No findings of significance were identified.

.6 Alternative Shutdown Capability

a. Inspection Scope

The team reviewed the licensee's alternative shutdown methodology to determine if the licensee properly identified the components, systems, and instrumentation necessary to achieve and maintain safe shutdown conditions from the auxiliary shutdown panel and alternative shutdown locations. The team focused on the adequacy of the systems selected for reactivity control, reactor coolant makeup, reactor heat removal, process monitoring and support system functions. The team verified that hot and cold shutdown from outside the control room could be achieved and maintained with offsite power available or not available. The team verified that the transfer of control from the control room to the alternative locations was not affected by fire-induced circuit faults by reviewing the provision of separate fuses for alternative shutdown control circuits.

The team reviewed the operational implementation of the licensee's alternative shutdown methodology. Team members observed a walk-through of the control room evacuation procedures with licensee personnel. The team observed operators simulate performing the steps of Procedure OTO-ZZ-00001, “Control Room Inaccessibility,” Revision 26, which provided instructions for performing an alternative shutdown from the remote shutdown panel and for manipulating equipment in the plant. The team verified that the minimum number of available operators, exclusive of those required for the fire brigade, could reasonably be expected to perform the procedural actions within the applicable plant shutdown time requirements and that equipment labeling was consistent with the procedure. Also, the team verified that procedures, tools, dosimetry, keys, lighting, and communications equipment were available and adequate to support successfully performing the procedure as intended. The team also reviewed records for operator training conducted on this procedure.

b. Findings

No findings of significance were identified. See Section 1R05.9 for a related finding concerning emergency lighting.

.7 Circuit Analyses

This segment is suspended for plants in transition to NFPA Standard 805.

.8 Communications

a. Inspection Scope

The team reviewed the adequacy of the communication system to support plant personnel in the performance of alternative safe shutdown functions and fire brigade duties. The team verified that plant telephones, page systems, sound powered phones,

and radios were available for use and maintained in working order. The team reviewed the electrical power supplies and cable routing for these systems to verify that either the telephones or the radios would remain functional following a fire. The team discussed system design, testing, and maintenance with the system engineer.

b. Findings

No findings of significance were identified.

.9 Emergency Lighting

a. Inspection Scope

The team reviewed the emergency lighting system required to support plant personnel in the performance of alternative safe shutdown functions to verify it was adequate to support the performance of manual actions required to achieve and maintain hot shutdown conditions, and for illuminating access and egress routes to the areas where manual actions are required. The locations and positioning of emergency lights were observed during a walk-through of Procedure OTO-ZZ-00001 and during review of manual actions implemented for the sample fire areas.

b. Findings

Introduction. The team identified an apparent violation of License Condition 2.C.(5)(c) for failure to ensure that required emergency lighting was provided to perform alternate shutdown actions to isolate the steam generator atmospheric relief valves. Specifically, the licensee failed to have the required emergency lighting in Fire Area A-23 and provided insufficient lighting in Fire Area C-10. However, this violation will not be cited since the licensee met the Enforcement Policy criteria for enforcement discretion for a plant committed to adopting NFPA Standard 805.

Description. The inspectors conducted a walk-through of Procedure OTO-ZZ-00001 with qualified licensed and non-licensed operators to validate the actions required for each watchstation. The inspectors questioned the quality of emergency lighting for manipulating breaker controls on Motor Control Center NG02A in Fire Area C-10 (Train B 4160 VAC Emergency Switchgear Room), for manipulating isolation valves for the steam generator atmospheric relief valves in Fire Area A-23 (Area 5), and for manipulating four valves in Fire Area A-24 (north piping penetration room). The licensee documented these lighting concerns as a condition adverse to quality in Callaway Action Request 200609759.

For Fire Area A-23, the inspector determined that the licensee had no emergency lights focused on any of the nitrogen and instrument air isolation valves for each of the steam generator atmospheric relief valves. The operator would need to isolate the steam generator atmospheric relief valves to prevent a reactor coolant system overcooling event. The inspectors verified that the balance of plant reactor operator had isolated the affected steam generator atmospheric relief valve within 11 minutes. The inspectors determined that the thermal hydraulic analysis allowed up to 21 minutes and required

the action to be completed within 29 minutes to prevent pressurizer level from decreasing to zero percent.

The inspectors verified each operator with alternative shutdown operations had a prestaged, inventoried equipment bag that included a flashlight that could be used for simple manipulations. The inspectors considered it within the skill of a licensed operator to be able to hold a hand held flashlight while manipulating the required valves for the steam generator atmospheric relief valves.

On November 30, 2006, the licensee performed a test of the emergency lights in Fire Area C-10 and in Fire Area A-23. For Motor Control Center NG02A in Fire Area C-10, the inspectors determined that insufficient lighting existed for the reactor operator to perform the breaker manipulations without using a hand held flashlight to read the procedure and to read the breaker labels. The emergency light was poorly aimed and had cable trays obstructing the light field. For the steam generator atmospheric relief valves in Fire Area A-23, the area surrounding the isolation valves was blacked out because of the lack of emergency lights and insufficient shine from the emergency lights aimed at the main steam isolation valves. The inspectors confirmed that manipulation of the isolation valves under those conditions could be accomplished using the hand held flashlight.

During the test in Fire Area A-23, two emergency lights provided for access and egress to the platform for manipulating the steam generator atmospheric relief valves were inoperable. The inspectors verified that the last annual performance test was satisfactory and reviewed records indicating that during quarterly tests for the last three years demonstrated the affected lights had operated satisfactory. The licensee initiated CAR 200609806 to document this condition adverse to quality.

From the inspectors observations related to the lighting levels in Fire Area A-23 and in Fire Area C-10, the inspectors informed the licensee that the light levels for the four valves in Fire Area A-24 would not be sufficient to perform the required actions without the use of a hand held flashlight. The inspectors agreed that the interim compensatory measures related to using the hand held flashlights would enable an operator to perform the component manipulations, if needed.

Analysis. This performance deficiency resulted from the licensee failing to provide adequate emergency lighting as required. This deficiency was more than minor in that it had the potential to impact the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to external events (such as fire) to prevent undesirable consequences. Consequently, the inspectors evaluated these deficiencies using Manual Chapter 0609, Appendix F, “Fire Protection Significance Determination Process.” Because of the simplicity of the tasks, the time available to perform the tasks and the availability of portable lighting, the inspectors concluded this issue had a low degradation rating; therefore, the inspectors concluded the issue was of very low safety significance in Phase 1. The finding has cross-cutting aspects related to human performance for failure to ensure that procedures could be implemented as required.

Enforcement. License Condition 2.C.(5)(c) specified that, "The licensee shall maintain in effect all provisions of the approved fire protection program as described in the SNUPPS Final Safety Analysis Report for the facility through Revision 15, the Callaway site addendum through Revision 8, and as approved in the Safety Evaluation Report through Supplement 4." The SNUPPS Final Safety Analysis Report Appendix 9.5E comparison to Section III.J, "Emergency Lighting," of 10 CFR Part 50, Appendix R states, "Complies. As stated in Section 9.5.3.2.3, emergency lighting units with eight-hour batteries are located in all plant areas required to manned for safe shutdown and also those areas necessary for access and egress." The Callaway site addendum Section 9.5.1.11, "Emergency Lighting," states, "Emergency Lighting Units with at least an eight-hour battery power supply shall be provided in all areas needed for operation of safe shutdown equipment and in access and egress routes thereto."

Procedure OTO-ZZ-00001 provided, in part, guidance to evacuate the control room in the event of a fire, which included actions in Fire Area A-23 for isolating the steam generator atmospheric relief valves and actions in Fire Area C-10 for manipulating breakers on Motor Control Center NG02A. Contrary to the above, a test performed on November 30, 2006, demonstrated that insufficient lighting was provided by emergency lights to accomplish the manual actions required to safely shutdown the reactor following a control room fire without the use of hand held flashlights. The licensee included this issue in the corrective action program as Callaway Action Request 200609759.

Because the licensee committed to adopting NFPA Standard 805 and changing their FPP license basis to comply with 10 CFR 50.48.(c), this issue is covered by enforcement discretion in accordance with the NRC Enforcement Policy. Specifically, this issue would have been expected to be identified and addressed during the licensee's conversion to NFPA Standard 805. The licensee: (1) had entered this issue into their corrective action program and implemented appropriate compensator measures, (2) was not associated with a finding that would be categorized under the Revised Oversight Process as Red or a Severity Level I violation, and (3) had submitted the letter of intent prior to December 31, 2005. The team's review concluded that this violation meets the criteria for enforcement discretion for plants in transition to a risk-informed, performance-based fire protection program as allowed per 10 CFR Part 50.48(c). Since all the criteria were met, the NRC is exercising enforcement discretion for this issue.

.10 Cold Shutdown Repairs

a. Inspection Scope

The team reviewed licensee procedures to determine whether repairs were required to achieve cold shutdown. The only repairs credited are the replacement of fuses for cold shutdown components that may have blown prior to transferring control to the remote shutdown panel were for a fire in the control room. The team verified that the procedures, equipment, and materials to accomplish repairs of components required for cold shutdowns are available and accessible. The team also verified that damaged components can be made operable, and cold shutdowns achieved within required time limits.

b. Findings

No findings of significance were identified.

.11 Compensatory Measures

a. Inspection Scope

The team reviewed the licensee's program with respect to compensatory measures in place for out-of-service, degraded, or inoperable fire protection and post-fire safe shutdown equipment, systems or features.

The team reviewed the Procedures APA-ZZ-00701, "Control of Fire Protection Impairments," APA-ZZ-00703, "Fire Protection Operability Criteria and Surveillance Requirements," and a sample of fire impairments to determine whether the procedures adequately controlled compensatory measures for fire protection systems, equipment and features (e.g., detection and suppression systems and equipment, and passive fire barriers).

The team reviewed Procedures APA-ZZ-00315, "Configuration Risk Management Program," EDP-ZZ-01129, "Callaway Plant Risk Assessment" and PDP-ZZ-00023, "Work Screening and Processing," to determine whether the procedures adequately controlled compensatory measures for out-of-service, degraded, or inoperable equipment that could affect post-fire safe shutdown equipment, systems or features.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

a. Inspection Scope

The team selected a sample of condition reports associated with the licensee's fire protection program to verify that the licensee had an appropriate threshold for identifying deficiencies. In addition the team reviewed the corrective actions proposed and implemented to verify that they were effective in correcting identified deficiencies.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

(Closed) Licensee Event Reports 2002-006-00 and 2002-006-01: Hot short issue reveals potential for draining refueling water storage tank to containment sump

a. Inspection Scope

An NRC inspector and a senior reactor analyst performed an in-office review of the licensees interim measures and risk assessment to determine if the licensee had demonstrated that the significance of the issues were less than high safety significance (Red). The inspector performed this inspection by reviewing corrective action documents and plant procedures. The inspector and senior reactor analyst discussed the issues with the fire protection engineer and licensee probabilistic safety assessment personnel. During the on-site portion of this inspection, the inspector confirmed information used and assumptions made during the evaluation.

The inspector performed the evaluation in this manner because Callaway formally committed to converting their FPP to comply with the requirements of 10 CFR 50.48(c) and NFPA Standard 805 prior to December 31, 2005. This involves using a risk-informed methodology. Because the conversion and licensing process is expected to identify and address a variety of difficult issues that are normally the subject of triennial fire protection inspections and because any findings in this area would have to be addressed under the new, rather than the existing, program, the NRC has adapted its inspection and enforcement of certain issues for plants in this situation.

b. Findings

Introduction. The team identified an apparent violation of License Condition 2.C.(5), "Fire Protection (Section 9.5.1.7 SER and Section 9.5.1.8, SSER No. 3)", concerning failure to assure safe shutdown systems are protected in accordance with the provisions of the approved fire protection program. The licensee credited manual actions to mitigate the effects of fire damage in lieu of providing the physical separation, physical protection, or an appropriate diverse means of accomplishing the safe shutdown function, which adversely affected the ability to achieve and maintain safe shutdown in the event of a fire. Specifically, a fire could potentially result in draining the refueling water storage tank to the containment sump. The issues identified in these licensee event reports are considered to be examples of the issues resulting in the apparent violation with enforcement discretion discussed in Section 1R05.2, Protection of Safe Shutdown Capabilities.

Description. On June 7, 2002, the licensee evaluated whether a hot short involving Valves EJHV8811A, Train A containment emergency recirculation sump isolation, and BNHV8812A, Train A refueling water storage tank to residual heat removal suction isolation, (or the Train B Valves EJHV8811B and BNHV8812B) could result in draining the refueling water storage tank to the containment emergency recirculation sump.

The licensee confirmed that the potential for hot shorts existed since a common cable tray contained the control cables for Valves EJHV8811A and BNHV8812A in Fire Areas A-1, A-2, and A-8. Similarly a different, although common, cable tray contained the control cables for Valves EJHV8811B and BNHV8812B in Fire Areas A-4 and A-1. The licensee concluded that a fire in any of the fire areas had the potential to damage the EJHV8811A or B valve circuits in such a manner to spuriously open and damage the BNHV8812A or B valve circuits in such a manner that they would not respond to a "close" signal. The control cables for both Trains A and B are routed through Fire Area A-1 and could potentially disable all four valves affecting the redundant safe shutdown trains. The licensee determined that the cables did not meet the separation

criteria required by their fire protection program and there was not a diverse water supply in Fire Area A-1.

The licensee attributed the root cause to a failure to understand during the original analysis that cable-to-cable hot shorts were required to be evaluated. As immediate corrective actions the licensee initiated hourly roving fire watches in the affected fire areas. Long term corrective actions included (1) modifying the control circuits of the motor-operated valves to prevent the over-thrust protection circuits from being bypassed and (2) updating fire pre-plans to instruct operators to manually close the appropriate valve to prevent draining the refueling water storage tank.

Analysis. This finding is of greater than minor safety significance because it impacted the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to external events (such as fire) to prevent undesirable consequences. Specifically, a fire in Fire Area A-1 had the potential to damage the Trains A and B circuits, which could result in draining the refueling water storage tank to the containment emergency recirculation sump.

This licensee event report documented that opening of the correct combination of valves would drain the refueling water storage tank to the containment sump in approximately 43 minutes. The licensee performed a risk evaluation of this condition and documented that evaluation in Probabilistic Risk Analysis Evaluation Report (PRAER) 02-172, "Post-Fire Safe Shutdown Circuit Concern," Revision 1. The analyst used fire initiation frequencies (FIF) documented in the Callaway Individual Plant Examination of External Events (IPEEE) to identify the risks in each individual fire area.

Fire Area A-1: The FIF for Fire Area A-1 (FIF_{A-1}) is 3.3×10^{-3}/yr. Probabilistic Risk Analysis Evaluation Report 02-172, Attachment B, "Evaluation of Fire Area A-1A Fire Modeling Scenarios," demonstrated that only one of the modeled scenarios, Scenario 6, had the potential to damage the cables for either trains valves. However, this fire scenario also failed the entire emergency core cooling system. As such, the draining of the refueling water storage tank to the containment sump would not impact the core damage frequency in this area; therefore, the frequency of a fire in Fire Area A-1 that would be impacted by the subject performance deficiency (λ_{A-1}) was zero.

Fire Area A-2: The FIF for Fire Area A-2 (FIF_{A-2}) is 2.6×10^{-3}/yr. Probabilistic Risk Analysis Evaluation Report 02-172, Attachment C, "Fire Area A-2," concluded that a fire in this area could result in the draining of the refueling water storage tank via the Train A valves as a result of the subject performance deficiency. Further, PRAER 02-172 documented that a fire induced transient with loss of main feedwater would only require emergency core cooling system injection if one of the following occurred: (1) the reactor does not trip upon demand; (2) a pressurizer power-operated relief valve fails to close after opening; (3) reactor coolant pump seal cooling is lost; or (4) there is no auxiliary feedwater available.

Probabilistic Risk Analysis Evaluation Report 02-172, Attachment F, "Estimation of the Probability for Needing ECCS," documented that a fire in Fire Area A-2 would not directly affect rod insertion, pressurizer power-operated relief valves, seal cooling, nor auxiliary feedwater. Therefore, only nominal failures of these systems would result in a

requirement for emergency core cooling system injection. Using the plant-specific standardized plant analysis risk model, the analyst calculated the probability of failure upon demand for each of these functions as 5.2×10^{-6}, 2.4×10^{-4}, 4.8×10^{-4}, and 1.2×10^{-5}, respectively. The probability that emergency core cooling system (P_{ECCS}) would be required following a fire in this area can be calculated as the sum of these four components (7.4×10^{-4}). Therefore, the frequency at which a fire would initiate in Fire Area A-2 and emergency core cooling system would be required (λ_{A-2}) was calculated as follows:

$$\lambda_{A-2} = FIF_{A-2} * P_{ECCS} = 2.6 \times 10^{-3}/yr * 7.4 \times 10^{-4} = 1.9 \times 10^{-6}/yr$$

The analyst noted that λ_{A-2} was a bounding frequency for a fire in Fire Area A-2 that could be impacted by the subject performance deficiency.

Fire Area A-4: The FIF for Area A-4 (FIF_{A-4}) is 2.8×10^{-3}/yr. Probabilistic Risk Analysis Evaluation Report 02-172, Attachment D, "Fire Area A-4," specified that a fire in this area could result in the draining of the refueling water storage tank via the Train B valves as a result of the subject performance deficiency.

Like the evaluation for Fire Area A-2, a fire induced transient with loss of main feedwater would only require emergency core cooling system injection if one of the following occurred: (1) the reactor does not trip upon demand; (2) a pressurizer power-operated relief valve fails to close after opening; (3) reactor coolant pump seal cooling is lost; or (4) there is no auxiliary feedwater available.

Therefore, the frequency at which a fire would initiate in Fire Area A-4 and would be required (λ_{A-4}) was calculated as follows:

$$\lambda_{A-4} = FIF_{A-4} * P_{ECCS} = 2.8 \times 10^{-3}/yr * 7.4 \times 10^{-4} = 2.1 \times 10^{-6}/yr$$

The analyst noted that λ_{A-4} was a bounding frequency for a fire in Fire Area A-4 that could be impacted by the subject performance deficiency.

Fire Area A-8: The FIF for Fire Area A-8 (FIF_{A-8}) is 8.0×10^{-4}/yr. Probabilistic Risk Analysis Evaluation Report 02-172, Attachment E, "Evaluation of Fire Area A-8 Fire Modeling Scenarios," demonstrated that three of the ten modeled scenarios (Scenarios 3, 9, and 10) had the potential to damage the cables for the Train A valves. Since Scenario 3 failed only one valve and the performance deficiency would only affect the core damage frequency if both valves failed, the analyst did not evaluate Scenario 3. In Scenario 9, the entire emergency core cooling system failed directly from the fire. As such, the draining of the refueling water storage tank to the containment sump had no additional impact on the core damage frequency in this area.

The FIF for Scenario 10 was 3.93×10^{-4} and is characterized as a transient fire from maintenance refuse impacting Raceway 4U3B6F. The licensee quantified a u factor (u), the probability that transient combustibles would be stored within damage range of the targets, as 3×10^{-2}. The analyst reviewed this assumption and reassessed the appropriate value for the u factor.

Fire Area A-8 contains approximately 11,000 ft^2 of floor space (A_{floor}) with an estimated 300 ft^2 of safety-related cable and conduit potentially impacted by transient fires in Scenario 10. Using the techniques in Tasks 2.3.3 and 2.3.4 of Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," the analyst considered the zone of influence for a 75th percentile transient fire impacting thermoset cables to provide a reasonable estimate for the combustion source. Specifically, Table 2.3.1, "Mapping of General Fire Scenario Characterization Type Bins to Fire Intensity Characteristics," listed 70 kW as the value for the 75th percentile fire resulting from solid transients. Table 2.3.2, "Calculated Values (in feet) for Use in the Ball and Column Zone of Influence Chart for Fires in an Open Location Away from Walls," indicates that the radius of influence would be approximately 1.3 feet from the fire.

In Manual Chapter 0609, Appendix F, Attachment 5, "Characterizing Non-Simple Fire Ignition Sources," describes a maintenance refuse fire of 70 kW. The analyst selected the largest item in the list of examples of 70 kW fires, a wooden pallet, as a bounding assumption. For a 4' x 4' wood pallet and a radius of influence of 1.3 feet from the edges, the zone of influence was estimated to be 6.6' x 6.6' or 43.6 ft^2. The analyst assumed that, the remaining unknown, the shape of the target (the 300 ft^2 of cables) was either a perfect square with 17.3 ft sides or two side-by-side cable trays 6.1 ft total width and a length of 49.2 ft. By assuming that any fire within 1.3 feet of the target could cause damage, the total effective source area (A_{source}), representing the entire area that the pallet could be located within and still affect the target cables, was calculated to be between 778 ft^2 and 999 ft^2, respectively. Therefore the analyst calculated the highest assumed u factor by dividing the maximum possible source area by the total area of the room as follows:

$$u = A_{source} \div A_{floor} = 999\ ft^2 \div 11{,}000\ ft^2 = 9.1 \times 10^{-2}$$

This represents the probability that maintenance refuse left randomly in the area would be located such that a fire would affect the target cables. Therefore, the frequency of a Scenario 10 (λ_{10}) fire initiating was quantified as follows:

$$\lambda_{10} = FIF * u = 3.93 \times 10^{-4}/yr * 9.1 \times 10^{-2} = 3.6 \times 10^{-5}/yr$$

The analyst noted that the frequency of a fire in Fire Area A-8 that could be impacted by the subject performance deficiency (λ_{A-8}) was equal to the frequency of λ_{10}.

Because the initiation of fires within different fire areas occur independently, the analyst determined that the probability of a fire would occurring within a 1-year assessment period (t_{EXP}) that would be impacted by the subject performance deficiency (P_{IMPACT}) would be equal to the sum of the four fire area probabilities. Therefore, this value was calculated as follows:

$$P_{IMPACT} = (\lambda_{A-1} * t_{EXP}) + (\lambda_{A-2} * t_{EXP}) + (\lambda_{A-4} * t_{EXP}) + (\lambda_{A-8} * t_{EXP})$$

$$= 0.0 + (1.9 \times 10^{-6}/yr * 1yr) + (2.1 \times 10^{-6}/yr * 1yr) + (3.6 \times 10^{-5}/yr * 1yr)$$

$$= 4.0 \times 10^{-5}$$

Given that the probability of a fire occurring within a 1-year assessment period that would be impacted by the subject performance deficiency is less than the 1 x 10^{-4} threshold, the analyst determined that this noncompliance was not associated with a finding that the Reactor Oversight Process Significance Determination Process would evaluate as Red.

Enforcement. License Condition 2.C.(5)(c) specified that, "The licensee shall maintain in effect all provisions of the approved fire protection program as described in the SNUPPS Final Safety Analysis Report for the facility through Revision 15, the Callaway site addendum through Revision 8, and as approved in the Safety Evaluation Report through Supplement 4." SNUPPS Final Safety Analysis Report, Appendix 9.5E states, "Redundant trains of systems required to achieve and maintain hot standby are separated by 3-hour fire barriers, or the equivalent provided by Section III.G.2, or else a diverse means of providing the safe shutdown capability exists and is unaffected by the fire."

Contrary to the above, on June 7, 2002, the licensee determined that they had not protected redundant trains of safe shutdown equipment in Fire Areas A-1, A-2, A-4, and A-8. Specifically, a cable-to-cable hot short involving Valves EJHV8811A and BNHV8812A or Valves EJHV8811B and BNHV8812B could result in draining the refueling water storage tank to the containment emergency recirculation sump.

The issues identified in these licensee event reports are considered to be examples of the issues resulting in the apparent violation with enforcement discretion discussed in Section 1R05.2, Protection of Safe Shutdown Capabilities.

4OA5 Other Activities

a. Inspection Scope

An NRC inspector and a senior reactor analyst performed an in-office review of the licensees interim measures and risk assessment to determine if the licensee had demonstrated that the significance of the issues were less than high safety significance (Red). The inspector performed this inspection by reviewing corrective action documents and plant procedures. The inspector and senior reactor analyst discussed the issues with the fire protection engineer and licensee probabilistic safety assessment personnel. During the on-site portion of this inspection, the inspector confirmed information used and assumptions made during the evaluation.

The inspector performed the evaluation in this manner because Callaway formally committed to converting their fire protection program to comply with the requirements of 10 CFR 50.48(c) and NFPA Standard 805 prior to December 31, 2005. This involves using a risk-informed methodology. Because the conversion and licensing process is expected to identify and address a variety of difficult issues that are normally the subject of triennial fire protection inspections and because any findings in this area would have to be addressed under the new, rather than the existing, program, the NRC has adapted its inspection and enforcement of certain issues for plants in this situation.

b. Findings

.1 (Closed) Unresolved Item 05000483/1999004-002: Capability of 72 valves to reach safe shutdown position following a control room fire

Introduction. The team identified an apparent violation of License Condition 2.C.(5)(c) for failure to ensure motor-operated valve circuits needed for alternative shutdown would be available. Specifically, the licensee had not modified the control circuits for 72 motor-operated valves to ensure that a hot short would not result in bypassing the torque and limit switches. However, this violation will not be cited since the licensee met the Enforcement Policy criteria for enforcement discretion for a plant committed to adopting NFPA Standard 805.

Description. The licensee had not demonstrated the capability of motor-operated valves to properly operate in the event of a control room fire. The licensee stated that they had accounted for functional failures of plant equipment in their evaluation of alternative shutdown capability for a control room fire. These failures included spurious actuations, but it was always assumed that any mispositioned components could be re-positioned after the components were isolated from the control room circuitry.

In 1999, NRC inspectors disagreed with this conclusion. The inspectors' concern was that the alternative shutdown capability must not be compromised in a control room fire. Specifically, the alternative shutdown capability must be designed such that components required for alternative shutdown capability, which spuriously actuate to an unsafe orientation, must not be damaged to an extent that they cannot be restored to a safe orientation. Therefore, the inspectors concluded that the licensee must ensure that motor-operated valves required for safe shutdown were capable of being repositioned following a hot short occurring prior to isolation from the control room. Further, the NRC had alerted licensees to the potential for motor-operated valve damage as a result of hot shorts in Information Notice 92-18, "Potential For Loss of Remote Shutdown Capability During a Control Room Fire." Valve or actuator damage was postulated because, for some circuit designs, a hot short would bypass torque or limit switch control features and permit the full power of the motor to be applied to the actuator and valve assembly.

The licensee initiated Request For Resolution 19056, "Design Change Request for Motor-Operated Valve Hot Short Issue," Revision A, in June 1998 that identified the population of motor-operated valves in the plant that (1) could be affected by spurious actuations from a control room fire and (2) could affect the ability to achieve hot shutdown if not in the assumed safe position. The licensee implemented a modification that reconfigured the motor-operated valve control circuits to prevent damaging the valves.

Analysis. This finding is of greater than minor safety significance because it impacted the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to external events (such as fire) to prevent undesirable consequences. Specifically, a control room fire had the potential to damage the control circuits of 72 motor-operated valves.

A senior reactor analyst evaluated the likelihood that operators would leave the main control room because of a fire. The IPEEE documents a FIF for the main control room of 1.01×10^{-2}/yr. A severity factor (P_{SF}) of 0.1 is appropriate because most fires are

self-limiting to a single component or are extinguished prior to damaging equipment. Finally, the IPEEE used the conditional probability that the control room would need to be evacuated, calculated in NSAC-181. This probability was modified to account for the dependency with the severity factor, and a value (P_{EVAC}) of 3.4×10^{-2} was used. The analyst then calculated the frequency of a control room evacuation (λ_{EVAC}) as follows:

$$\lambda_{EVAC} = FIF * P_{SF} * P_{EVAC} = 1.01 \times 10^{-2}/yr * 0.1 * 3.4 \times 10^{-2} = 3.4 \times 10^{-5}$$

Since the frequency of a control room evacuation (3.4×10^{-5}) at Callaway is lower than the Yellow/Red threshold of 1×10^{-4}, no issue involving control room evacuation, which does not increase the likelihood of leaving the control room, is associated with a finding that would evaluate as Red in accordance with the Significance Determination Process.

Enforcement. License Condition 2.C.(5)(c) specified that, "The licensee shall maintain in effect all provisions of the approved fire protection program as described in the SNUPPS Final Safety Analysis Report for the facility through Revision 15, the Callaway site addendum through Revision 8, and as approved in the Safety Evaluation Report through Supplement 4." SNUPPS Final Safety Analysis Report Table 9.5E-1 states, "An auxiliary shutdown panel, described in Section 7.4, in conjunction with certain local controls, provides a means of achieving and maintaining hot standby in the event that the main control room is uninhabitable," and "The auxiliary shutdown panel contains the controls and indication necessary to maintain reactor coolant inventory, remove decay heat, and provide boration for hot standby." Procedure OTO-ZZ-00001 provided guidance to evacuate the control room in the event of a fire and required operation of motor-operated valves. Contrary to the above, the potential existed for a control room fire to result in damage to motor-operated valves related to maintaining reactor coolant inventory, removing decay heat or borating the reactor coolant system which could prevent the safe shutdown function from being accomplished.

Because the licensee committed to adopting NFPA Standard 805 and changing their Fire Protection Program license basis to comply with 10 CFR 50.48.(c), this issue is covered by enforcement discretion in accordance with the NRC Enforcement Policy. Specifically, the licensee: (1) had entered this issue into their corrective action program (Callaway Action Request 200106750) and implemented appropriate compensatory measures, (2) was not associated with a finding that would be categorized under the Revised Oversight Process as Red or a Severity Level I violation, and (3) had submitted the letter of intent prior to December 31, 2005. The team's review concluded that this violation meets the criteria for enforcement discretion for plants in transition to a risk-informed, performance-based fire protection program as allowed per 10 CFR Part 50.48(c). Since all the criteria were met, the NRC is exercising enforcement discretion for this issue.

.2 (Closed) Unresolved Item 05000483/2003007-001: Failure to protect associated circuits

A previous inspection had raised concerns about the potential failure to protect safe shutdown circuits in Fire Areas A-18, A-21 and C-9. The team reviewed the detailed circuit diagrams and cable routing information for each of the identified fire areas and determined that appropriate circuit separation existed. Consequently, no violation of regulatory requirements had occurred. This unresolved item is closed.

.3 (Closed) Unresolved Item 05000483/2003007-02: Failure to perform alternative shutdown manual actions within required times

Introduction. The inspectors identified an apparent violation of License Condition 2.C.(5)(c) for failure to ensure that alternate shutdown actions could be completed in the required response times. However, this violation will not be cited since the licensee met the Enforcement Policy criteria for enforcement discretion for a plant committed to adopting NFPA Standard 805.

Description. In a 2003 inspection, the inspectors conducted a walk-through of Procedure OTO-ZZ-00001, Revision 19 with qualified licensed and non-licensed operators to validate the actions required for the assigned watchstation. Some of the required safe shutdown actions took longer than the time specified in the safety evaluation report. The licensee could not provide documentation that demonstrated the actions and times satisfied the 10 CFR Part 50, Appendix R, Section III.L.2 performance goals. The inspectors identified several instances where the time specified in a safety evaluation report or calculation were exceeded:

Required Action	**Safety Evaluation Report**	**Calculation Time**	**Measured Time**
Establish auxiliary feedwater flow	5 min	17 min	9:06 min
Open two steam generator PORVs	5 min		9:30 min
Isolate letdown	5 min	61 min	8:05 min
Isolate pressurizer PORVs	5 min	7 min	8:05 min
Isolate auxiliary pressurizer spray		20 min	not in procedure

No detailed thermo-hydraulic analysis had been performed at the time of original licensing so the impact of the changes could not be evaluated at the time of the last inspection. The licensee has performed a control room fire transient analysis, “Evaluation of Alternative Shutdown Manual Actions, Transient Analyses and Operator Timelines to Address NRC URI 2003007-02," to analyze postulated fire-inducted transients and establish time requirements to restore system functions or mitigate spurious operations. Procedure OTO-ZZ-00001, “Control Room Inaccessibility,” was confirmed to accomplish the actions required to establish safe hot shutdown conditions within the analyzed time limits.

Analysis. The team determined that this was a performance deficiency because the licensee failed to ensure that alternate shutdown actions could be completed in the required times documented in the approved fire protection program as required by License Condition 2.C.(5)(c). This deficiency was more than minor in that it had the potential to impact the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to external events (such as fire) to prevent undesirable consequences.

A senior reactor analyst determined the likelihood that operators would leave the main control room because of a fire. The Callaway IPEEE documents a FIF for the main control room of 1.01 x 10^{-2}/yr. This is in line with peers in the industry. A severity factor (P_{SF})of 0.1 is appropriate because most fires are self-limiting to a single component or are extinguished prior to damaging equipment. Finally, the Callaway IPEEE used the conditional probability that the control room would need to be evacuated, calculated in NSAC-181. This probability was modified to account for the dependency with the severity factor, and a value (P_{EVAC}) of 3.4 x 10^{-2} was used. The analyst then calculated the frequency of a control room evacuation (λ_{EVAC}) as follows:

$$\lambda_{EVAC} = FIF * P_{SF} * P_{EVAC} = 1.01 \times 10^{-2}/yr * 0.1 * 3.4 \times 10^{-2} = 3.4 \times 10^{-5}$$

Since the frequency of a control room evacuation (3.4 x 10^{-5}) at Callaway is lower than the Yellow/Red threshold of 1 x 10^{-4}, no issue involving control room evacuation, which does not increase the likelihood of leaving the control room, is associated with a finding that would evaluate as Red in accordance with the Significance Determination Process.

Enforcement. License Condition 2.C.(5)(c) specified that, "The licensee shall maintain in effect all provisions of the approved fire protection program as described in the SNUPPS Final Safety Analysis Report for the facility through Revision 15, the Callaway site addendum through Revision 8, and as approved in the Safety Evaluation Report through Supplement 4." SNUPPS Final Safety Analysis Report Appendix 9.5E states, "The auxiliary shutdown panel contains the controls and indication necessary to maintain reactor coolant inventory, remove decay heat, and provide boration for hot standby." Procedure OTO-ZZ-00001, provided guidance to evacuate the control room in the event of a fire. Contrary to the above, the licensee failed to ensure that operators could perform required alternate shutdown response actions within the required response times. The licensee included this issue in the corrective action program as Callaway Action Request 200307160.

Because the licensee committed to adopting NFPA Standard 805 and changing their Fire Protection Program license basis to comply with 10 CFR 50.48.(c), this issue is covered by enforcement discretion in accordance with the NRC Enforcement Policy. Specifically, the licensee: (1) had entered this issue into their corrective action program and implemented appropriate compensator measures, (2) was not associated with a finding that would be categorized under the Revised Oversight Process as Red or a Severity Level I violation, and (3) had submitted the letter of intent prior to December 31, 2005. The team's review concluded that this violation meets the criteria for enforcement discretion for plants in transition to a risk-informed, performance-based fire protection program as allowed per 10 CFR Part 50.48(c). Since all the criteria were met, the NRC is exercising enforcement discretion for this issue.

4OA6 Management Meetings

Exit Meeting Summary

On November 30, 2006, the team leader presented the inspection results to Mr. Charles Naslund, Senior Vice-President and Chief Nuclear Officer, and other members of licensee management at the conclusion of the onsite inspection. The inspectors

confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

R. Barton, Shift Assent Operations Manager
K. Bruckerhoff, Fire Marshal
F. Diya, Manager, Engineering Services
L. Eitel, Fire Protection Engineer
D. Fanguy, System Engineer
D. Fitzgerald, Manager, Regulatory Affairs
J. Fortman, Major Projects Engineer
G. Harris, Major Projects Engineer
A. Heflin, Vice-President, Nuclear
J. Herrmann, Vice-President, Nuclear Engineering
J. Hiller, Regulatory Affairs Engineer
D. Hollabaugh, Superintendent, Employee Concerns
L. Kanuckel, Manager, Quality Assurance
R. Kempf, Systems Engineering-BOP
S. Magloi, Superintendent, Nuclear Engineering-Systems
K. Mills, Supervising Engineer, Regulatory Affairs
T. Moser, Manager, Nuclear Engineering-Plant
C. Naslund, Senior Vice-President and Chief Nuclear Officer
D. Neterer, Manager, Nuclear Operations
E. Olson, Superintendent, Performance Improvement
J. Patterson, Manager PS&O
S. Petzel, Region Regulatory Affairs Engineer
S. Reed, Supervising Engineer, Nuclear Engineering Systems - BOP
D. Rickard, Corrective Action Program Supervisor
S. Sardboth, SAPA Engineer
J. Schaefer, EOP Coordinator
D. Walker, Engineering

NRC

M. Peck, Senior Resident Inspector
D. Dumbacher, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

None

Closed

05000483/2002-006-00 and -01	LER	Hot short issue reveals potential for draining RWST to containment sump (Section 4OA3.b)
05000483/1999004-02	URI	Capability of 72 valves to reach safe shutdown position following a control room fire (Section 4OA5.b.1)
05000483/2003007-01	URI	Failure to protect associated circuits (Section 4OA5.b.2)
05000483/2003007-02	URI	Failure to perform alternative shutdown manual actions within required times (Section 4OA5.b.3)

LIST OF DOCUMENTS REVIEWED

The following documents were selected and reviewed by the team to accomplish the objectives and scope of the inspection.

CALLAWAY ACTION REQUEST SYSTEM (CARS)

200106750	200402257	200505106	200507641	200601382	200607577
200203631	200403528	200505419	200507763	200602563	200608584
200307092	200406765	200506312	200508005	200602620	200608640
200307160	200500140	200506422	200508172	200602646	200608918
200307169	200503420	200506546	200509837	200602657	200608927
200307199	200504012	200506610	200600114	200603258	200608947
200307204	200504941	200506771	200600134	200607119	200609759*
200307232	200504949	200506892	200600313	200607547	200609760*

*Initiated due to inspection activities.

DRAWINGS

Number	Title	Revision
A-2802	Architectural Fire Delineation Floor Plan, El. 2000' - 0"	12
A-2803	Architectural Fire Delineation Floor Plan, El. 2026' - 0"	9
E-2F1301	Fire Detection/Protection System - Auxiliary Building El. 2000' -0"	8
E-2F1401	Fire Detection/Protection System - Auxiliary Building El. 2026' -0"	6
E-23AB01	Schematic Diagram - Main Steam Supply Valve to Turbine Driven Aux Feedwater Pump	8
E-23AB28	Schematic Diagram - Main Steam Isolation Valves Control - Active Side	9
E-23AB29	Schematic Diagram - Main Steam Isolation Valves Control - Standby Side	6
E-23AL01B	Schematic Diagram - Motor Driven Auxiliary Feedwater Pump B	6
E-23AL05A	Schematic Diagram - Aux Feedwater Pumps, Discharge Control - Air Operated Valve	4
E-23AL05B	Schematic Diagram - Aux Feedwater Pumps, Discharge Control - Air Operated Valve	3
E-23BB03	Schematic Diagram - RCP Thermal Barrier CCW Iso Valves	10
E-23BB30	Schematic Diagram - RCS Head Vent Valves	2
E-23BG01A	Schematic Diagram - Centrifugal Charging pump B	5
E-23BG12A	Schematic Diagram, Volume Control Tank Outlet Isolation Valve	9
E-23BG19	Schematic Diagram - Reactor Coolant System Spray Valves	4
E-23BN01	Schematic Diagram, Refueling Water Storage Tank to Charging Pump MOV	5
E-23QN71	Schematic Diagram, Antenna For Radio Systems Control Building Comm. Corridor & Turbine Bldg	4
E-23RP15	Schematic Diagram - Lock-Out Relay Control Circuits RP334/RP335	2

Number	Title	Revision
J-25001	Local Control Panel RP118 - Auxiliary Shutdown Panel	0
M-0H1251	Heating Ventilating & Air Cond., Auxiliary Building El. 1974'-0', 1989'-0' & 2000'-0' Area 5	7
M-0H1905	Heating Ventilating & Air Cond., Fire Damper Details	4
M-2G020	Equipment Location Reactor and Auxiliary Buildings Plan - Basement EL. 1974' - 0"	39
M-2G021	Equipment Location Auxiliary Building Partial Plan - EL. 1988' - 0" & EL. 2013' - 6"	10
M-2G022	Equipment Location Reactor and Auxiliary Buildings Plan - Ground Floor EL. 2000' - 0"	58
M-2G023	Equipment Location Reactor and Auxiliary Buildings Plan - EL. 2026' - 0"	41
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ENGINEERING INFORMATION RECORDS

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51-5050606-00	Callaway Appendix R Secondary Side Depressurization Analysis Basis Document	00
51-5051110-00	Callaway Appendix R Maximum RCS Overcooling Analysis Basis Document	00
51-5051812-00	Callaway Appendix R Maximum RCS Depressurization Analysis Basis Document	00
51-5054393-00	Callaway Appendix R Maximum RCS Overheating Analysis Basis Document	00

FIRE IMPAIRMENTS

14050 14195 14288

PROCEDURES

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APA-ZZ-00315	Configuration Risk Management Program	004
APA-ZZ-00395	Significant Operator Response Timing	007

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APA-ZZ-00700	Fire Protection Program	013
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OSP-KC-00020	Pre-Action Sprinkler Supervisory Air Alarm Test	005
OSP-KC-03003	Fire Main Flow Test	000
OSP-KC-03004	Fire Pump Performance Test	001
OSP-ZZ-65045	Fire Barrier Seal Visual Inspection	005
OSP-ZZ-65046	Fire Barrier Inspection	010
OTO-ZZ-00001	Control Room Inaccessibility	26
PDP-ZZ-00023	Work Screening and Processing	010

MISCELLANEOUS DOCUMENTS

Number	Title	Revision
AREVA Report	Evaluation of Alternative Shutdown Manual Actions, Transient Analyses and Operator Timelines to Address NRC URI 2003007-02	0
AREVA Report	Manual Action Timeline For Control Room Evacuation Scenario - Attachment 2	0
AP96-004	Quality Assurance Department Audit Report	March 20, 1996
Facility Operating License No. NPF-30	Union Electric Company, Docket No. STN 50-483, Callaway Plant Unit No. 1, Facility Operating License	48
Letter 85-0109	Letter from Nicholas A. Petrick, SNUPPS, to Harold R. Denton, NRR, Subject: Fire Protection Review	August 23, 1984
FSAR - 9.5.1	Final Safety Analysis Report - Section 9.5.1, Fire Protection System	OL-15 (5/06)
FSAR - 9.5.1	Final Safety Analysis Report - Section 9.5.1, Fire Protection System	Rev 14 (3/84)
NUREG-0830	Safety Evaluation Report Related to the Operation of Callaway Plant, Unit No. 1	October 1981
NUREG-0830 Supplement No. 3	Safety Evaluation Report Related to the Operation of Callaway Plant, Unit No. 1	May 1984
NUREG-0830 Supplement No. 4	Safety Evaluation Report Related to the Operation of Callaway Plant, Unit No. 1	October 1984

Number	Title	Revision
RFR NO. 20817	Requests For Resolution - Evaluate Fire Detector Spacing Requirements	A
Surveillance Task S705109	Fire Damper Drop Test & Lubrication	02/28/04
Surveillance Task S705110	Fire Damper Drop Test & Lubrication	02/25/04
	Annual Preventive Maintenance Work Document (I.E. PM-17) For QD (Emergency Lighting System)	
	Emergency Lighting Maintenance Rule Table	
	Emergency Lighting Specifications For TC6L and TC6N Series and Chairman Series Emergency Lighting Equipment	January 1985
	Fire Protection Emergency Battery Lights	April 30, 1996
	Maintenance Rule Evaluations For QD (Emergency Lighting System) 11/12/05 - 11/14/05	
	Plant Health Report For QD (Emergency Lighting System)	November 14, 2006
	Training Records for RO/SRO/NLO Participation for Procedure OTO-ZZ-000001, “Control Room Inaccessibility,” Cycle 2006-01	

MODIFICATIONS

Number	Title	Revision
MP 02-1019	Install Darmatt fire wrap on several electrical components in Aux. Building	A