X. Other Requests for Approval

10 Pages Attached

Approval Request 1

On February 19, 1987 Callaway Plant submitted a license amendment request via ULNRC-01447 to delete fire protection Technical Specifications and relocate those requirements to the FSAR under licensee control in accordance with Generic Letter 86-10, "Implementation of Fire Protection Requirements."

On October 30, 1987 Callaway Plant responded to NRC questions related to this license amendment request via ULNRC-01667. Specifically, the following question and response is documented in ULNRC-01667.

> NRC Question
>
> The shutdown requirement of Specification 3.7.10.1 ACTION b should be retained in an appropriate commitment document.
>
> Response
>
> As part of implementing the proposed revisions to the Technical Specifications, the requirements of Specification 3.7.10.1 ACTION b will be retained and will not be modified without prior approval from the Nuclear Regulatory Commission (NRC). The 'requirements of Specification 3.7.10.1 ACTION b will be added' to FSAR (USAR for Wolf Creek) Table 9.5.1-2 with a statement that no modifications to these requirements will be made without prior approval of the NRC.

On January 13, 1988 the NRC issued Amendment No. 30 to Facility Operating License No. NPF-30. In the accompanying safety evaluation the staff noted the following:

> The licensee had originally proposed to delete the shutdown requirement of Specification 3.7.10.1 Action b. The staff's position is that the loss of the normal fire protection water supply and the inability to establish a back-up fire suppression water system within 24 hours warrant plant shutdown. The licensee responded that the requirements of Specification 3.7.10.1 Action b. will be added to the FSAR with commitment that no modifications to these requirements will be made without prior approval from NRC. The staff considers this response to be acceptable.

In response to the above, Callaway Plant has maintained the following statement in FSAR Table 9.5.1-2 for the Fire Suppression Water System, requirements a, b and c.

> With the Fire Suppression Water System in this condition, establish a backup Fire Suppression Water System within 24 hours. If this required action cannot be met, the requirements of Technical Specification 3.0.3 shall be initiated. Modifications to these requirements shall not be made without prior approval of the NRC.

As part of the transition to NFPA 805, it is being requested that the NRC Staff review and approve the elimination of the requirement currently listed in FSAR Table 9.5.1-2 to enter Technical Specification LCO 3.0.3 for an inoperable Fire Suppression Water System coupled with the inability to provide a backup fire suppression water system within 24 hours.

Basis for Request:

The current application of Technical Specification LCO 3.0.3 for plant configurations that do not meet the specific criteria of 10 CFR 50.36(c)(2) for inclusion into the plant Technical Specifications is inappropriate.

Technical Specification LCO 3.0.3 is intended to be applied when a Technical Specification LCO is not met and the associated Technical Specification required actions are not met, an associated Technical Specification required action is not provided, or if directed by the

associated Technical Specification required actions. Technical Specification LCO 3.0.3 was not meant to be applied to "non-technical specification" plant configurations.

The requirements for FSAR Table 9.5.1-2, "Fire Protection System Requirements," were previously approved for relocation from the Technical Specifications to the FSAR in an NRC SER dated January 13, 1988. The commitment made by Union Electric in ULNRC-01667 was that the requirements of Specification 3.7.10.1 ACTION b would be added to the FSAR with a statement that no modifications to these requirements will be made without prior approval of the NRC.

The current FSAR Table 9.5.1-2 text is as follows:

> With the Fire Suppression Water System in this condition, establish a backup Fire Suppression Water System within 24 hours. If this required action cannot be met, the requirements of Technical Specification 3.0.3 shall be initiated. Modifications to these requirements shall not be made without prior approval of the NRC.

The existing requirement to enter Technical Specification LCO 3.0.3 would not be consistent with NFPA 805 which indicates that compensatory actions should be appropriate with the level of risk created by the unavailable equipment.

NFPA 805, Section 3.2.3 states:

> *"Procedures shall be established for implementation of the fire protection program. In addition to procedures that could be required by other sections of the standard, the procedures to accomplish the following shall be established:*
>
> *(2) Compensatory actions implemented when fire protection systems and other systems credited by the fire protection program and this standard cannot perform their intended function and limits on impairment duration."*

Section 3.2.3 is supplemented by the following guidance from NFPA 805 Appendix A:

> *"A.3.2.3(2) Compensatory actions might be necessary to mitigate the consequences of fire protection or equipment credited for safe shutdown that is not available to perform its function. Compensatory actions should be appropriate with the level of risk created by the unavailable equipment. The use of compensatory actions needs to be incorporated into a procedure to ensure consistent application. In addition, plant procedures should ensure that compensatory actions are not a substitute for prompt restoration of the impaired system."*

As stated in Attachment A, NEI 04-02 Table B-1, Callaway Plant procedure APA-ZZ-00703, "Fire Protection Operability Criteria and Surveillance Requirements," will be used to establish the required compensatory actions and impairment durations following the transition to NFPA 805. FSAR Table 9.2.1-2 will be eliminated.

Nuclear Safety and Radiological Release Performance Criteria:

An inoperable Fire Suppression Water System with the inability to establish a backup system within 24 hours is an off-normal occurrence. NFPA 805 Section 3.2.3(2) requires compensatory actions to be implemented when fire protection systems and other systems credited by the fire protection program cannot perform their intended function. NFPA 805 Section 3.2.3(2) also requires that limits be established on the impairment duration. Callaway Plant procedure APA-ZZ-00703, "Fire Protection Operability Criteria and Surveillance Requirements," will be used to establish the required compensatory actions and impairment durations following the transition to NFPA 805. There is no impact on the nuclear safety performance criteria.

An inoperable Fire Suppression Water System with the inability to establish a backup system within 24 hours has no impact on the radiological release performance criteria. The radiological release performance criteria are satisfied based on the determination of limiting radioactive release (Attachment E), which is not affected by impacts on the fire protection system due to this condition.

Safety Margin and Defense-in-Depth:

An inoperable Fire Suppression Water System with the inability to establish a backup system within 24 hours is an off-normal occurrence. NFPA 805 Section 3.2.3(2) requires compensatory actions to be implemented when fire protection systems and other systems credited by the fire protection program cannot perform their intended function. NFPA 805 Section 3.2.3(2) also requires that limits be established on the impairment duration. Callaway Plant procedure APA-ZZ-00703, "Fire Protection Operability Criteria and Surveillance Requirements," will be used to establish the required compensatory actions and impairment durations following the transition to NFPA 805. Because actions resulting from this condition will not deviate from the approved fire protection program, there is no impact on safety margin and defense-in-depth.

Conclusion:

NRC approval is requested to eliminate the current requirement to enter Technical Specification 3.0.3 for an inoperable Fire Suppression Water System with the inability to establish a backup system within 24 hours.

NFPA 805 Section 3.2.3(2) requires compensatory actions to be implemented when fire protection systems and other systems credited by the fire protection program cannot perform their intended function. NFPA 805 Section 3.2.3(2) also requires that limits be established on the impairment duration. As identified in the Section 3.2.3(2) compliance basis in Attachment A, Callaway Plant procedure APA-ZZ-00703, "Fire Protection Operability Criteria and Surveillance Requirements," will be used to establish the required compensatory actions and impairment durations following the transition to NFPA 805. FSAR 9.2.1-2 will be eliminated following NRC approval of this request and during the transition to NFPA 805.

Approval Request 2

Approval is requested for a deviation from common enclosure analysis requirements of NFPA 805 Section 2.4.2 for specific Current Transformer (CT) configurations where a fire-induced open-circuit failure could result in a secondary fire.

A fire in plant fire area C-21, Lower Cable Spreading Room, Control Building El. 2032' or in plant fire area C-27, Main Control Room, Control Building El. 2047', results in an open circuit failure for circuits associated with the Main Generator Current Transformers (CTs) TVMA10A, B, and C. Due to the design of the CTs (turns ratio, relaying accuracy class, isolation circuit design) a secondary fire due to overheating can occur in plant fire area TB-1 at the CTs.

Requirements

Section 2.4.2 of NFPA 805 requires consideration of fire-induced open-circuit failure modes and specifies that circuits which share a common enclosure with circuits required to achieve nuclear safety performance criteria, be evaluated to ensure that such electrical faults will not cause the fire to extend beyond the immediate (initial) fire area. As discussed in NFPA 805 B.3.4.2 the evaluation of common enclosure issues should include consideration of Current Transformers that are constructed such that an open secondary circuit could cause ignition of the transformer. Specific details of the requirements are addressed below.

NFPA 805 Section 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Section 2.4.2.2.2 of NFPA 805 states:

> *"Other Required Circuits. Other circuits that share common power supply and/or common enclosure with circuits required to achieve nuclear safety performance criteria shall be evaluated for their impact on the ability to achieve nuclear safety performance criteria.*
>
> *(a) Common Power Supply Circuits. Those circuits whose fire-induced failure could cause the loss of a power supply required to achieve the nuclear safety performance criteria shall be identified. This situation could occur if the upstream protection device (i.e., breaker or fuse) is not properly coordinated with the downstream protection device.*
>
> *(b) Common Enclosure Circuits. Those circuits that share enclosures with circuits required to achieve the nuclear safety performance criteria and whose fire induced failure could cause the loss of the required components shall be identified. The concern is that the effects of a fire can extend outside of the immediate fire area due to fire-induced electrical faults on inadequately protected cables or via inadequately sealed fire area boundaries."*

NFPA 805 Section B.3.4.2 of appendix B to NFPA 805 states in part:

> *A special type of common enclosure issue involves current transformers. ...An opening in the secondary circuit causes excessively high voltages in the current transformer secondary circuit in an attempt to maintain this ratio, which can result in an ignition of the transformer materials.*

Regulatory Guide 1.205 Rev 1 Section 3.3, "Circuit Analysis," states:

Chapter 3 of industry guidance document NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," Revision 2, issued May 2009 (Ref. 12), when used in conjunction with NFPA 805 and this regulatory guide, provides one acceptable approach to circuit analysis for a plant implementing an FPP under 10 CFR 50.48(c).

NEI 00-01 Section 3.5.2.1, "Circuit Failure Due to An Open Circuit," states:

This section provides guidance for addressing the effects of an open circuit for safe shutdown equipment. An open circuit is a fire-induced break in a conductor resulting in the loss of circuit continuity. An open circuit will typically prevent the ability to control or power the affected equipment. An open circuit can also result in a change of state for normally energized equipment. For example, a loss of power to the main steam isolation valve (MSIV) solenoid valves [for BWRs] due to an open circuit will result in the closure of the MSIV.

NOTE: The EPRI circuit failure testing indicated that open circuits are not likely to be the initial fire-induced circuit failure mode. Consideration of this may be helpful within the safe shutdown analysis. Consider the following consequences in the safe shutdown circuit analysis when determining the effects of open circuits:

- *Loss of electrical continuity may occur within a conductor resulting in deenergizing the circuit and causing a loss of power to, or control of, the required safe shutdown equipment.*
- *In selected cases, a loss of electrical continuity may result in loss of power to an interlocked relay or other device. This loss of power may change the state of the equipment. Evaluate this to determine if equipment fails safe.*
- *Open circuit on a high voltage (e.g., 4.16 kV) ammeter current transformer (CT) circuit may result in secondary damage.*

Basis for Request:

The following assumes a plant fire results in an open circuit and resultant overheating and secondary fire at one or more of the Main Generator CTs TVMA10A, B, and C. Additionally, when the secondary fire occurs and the site fire brigade is assumed to be unavailable to respond due to the initial fire in either C-21 or C-27, the overheating and secondary fire at the CTs will result in a Main Generator trip and subsequent plant trip and also result in de-energizing the CTs. Therefore, the basis below addresses the secondary fire and its effects.

The TVMA10A, B, and C CTs are doughnut CTs placed around the generator bushings located under the Main Generator. The generator sits over a rectangular opening in the concrete turbine deck. The generator bushings are in the opening that comes out of the bottom of the generator. Figure X-1 shows the typical arrangement for the CTs.

- There are no redundant systems, cables or components required to meet the Nuclear Safety Capability Assessment that will be affected by the secondary fire.
- The fire will be limited to the CTs due to the lack of insitu combustibles surrounding the CTs. The area around the CTs is either concrete or metal. Other than the CTs there are no close exposed combustibles. The field leads and wiring to the CTs are in enclosed metal conduit and will not propagate fire. The generator bushings are composed of ceramic material and are not affected by the CT overheating or fire.

- The fire will not be challenging or severe due to the lack of insitu combustibles design of the CTs. Due to the lack of insitu combustibles the heat release rate resulting from the fire will be not be significant. Due to the size and open nature of the Turbine Building a hot gas layer is not possible from this fire.
- The fire will not result in ignition of any adjacent transient combustibles that could result in fire growth beyond the CTs. Access to the CTs is by ladder from the 2033' elevation to an elevated walkway with grated flooring. As indicated in Figure X-1, there is significant free space between the CTs and the elevated walkway.
- As defense-in-depth, the area of the TB-1 elevation 2033' is protected by a full area wide automatic pre-action suppression system.

Figure X-1- Typical Configuration of Main Generator CTs

Acceptance Criteria Evaluation:

Nuclear Safety and Radiological Release Performance Criteria:

The secondary fire will be specific to the CT transformers and will not affect any redundant systems, cables or components required to meet the Nuclear Safety Capability Assessment for fires in C-21 or C-27 the initiating event fire areas.

The secondary fire would have no impact on the radiological release performance criteria. The CTs and the area surrounding the CTs is outside the permanent radiological controlled area

(RCA) and it is not a storage location for radioactive materials. The access to the CTs is by ladder to a raised platform with grated flooring.

Safety Margin and Defense-in-Depth:

There are conservatisms in the circuit failure analysis. The safety margin in the analysis for the fire event in C-21 and C-27 has been preserved. The postulated secondary fire will not affect any assumptions or analysis utilized for the evaluation of fire affects in fire areas C-21 and C-27.

As defense-in-depth, the area of the TB-1 elevation 2033' is protected by a full area wide automatic pre-action suppression system. The defense-in-depth analysis for the fire events in C-21 or C-27 has been preserved and is not impacted by the secondary TB-1 CT fire.

Conclusion:

Approval is requested for a deviation from common enclosure analysis requirements of NFPA 805 Section 2.4.2 for specific CT configurations where a fire-induced open-circuit failure could result in a secondary fire.

A fire in plant fire area C-21, Lower Cable Spreading Room, Control Building El. 2032' or in plant fire area C-27, Main Control Room, Control Building El. 2047', could result in a secondary fire at the Main Generator CTs TVMA10A, B, and C located in fire area TB-1. However the secondary fire will have no adverse impact on meeting the Nuclear Safety Capability criteria for fires in C-21 and C-27.

Approval Request 3

Request:

Approval is requested for a deviation from the deterministic requirements of NFPA 805 Section 4.2.3.4 for five (5) specific Reactor Building (RB) configurations where the twenty or more feet of separation is present but the areas are not free of intervening combustibles. Where this occurs, intervening combustibles are present in the form of exposed cable trays. Callaway Plant Calculation KC-26, "Nuclear Safety Capability Assessment," subdivides fire area RB-1 into five (5) analysis areas, RB-1 through RB-5, each with a unique compliance strategy.

These five (5) configurations where the twenty of more feet of separation is present but it is not free of intervening combustibles are summarized below:

1. RB-2 (2026'-0") to RB-5 (2047'-6" & 2049'-0"): Open floor grating. Cable trays do not pass through the open grating (Ref. Callaway Drawing E2R2421, Detail Section "A"); however, cable tray 5U2J37 is routed below and parallel to the grating at El. 2049'-0" (2'-0" below grating).

2. RB-1 (2026'-0") to RB-2 (2026'-0"): Multi-foot thick solid bio-shield wall. Raceways 6A2A15, 6U2A15, 6J2A15 penetrate the wall. (Ref. Callaway Drawing E-2R2421)

3. RB-1 (2026'-0") to RB-5 (2047'-6"): Open floor grating. Cable trays do not pass through the open grating (Ref. Callaway Drawings E-2R2411, E-2R2421, E-2R2511); however, a cable tray 6A2B23 is routed below and parallel to the grating at El. 2039'-9" (>7' below grating).

4. RB-2 (2026'-0") to RB-5 (2047'-6"): Open floor grating. Cable trays do not pass through the open grating (Ref. Callaway Drawings E-2R2411, E-2R2511); however, cable trays 5U2A05, 5U2A10 are routed below and parallel to the grating at El. 2033'-0" (approximately 14' below grating).

5. RB-2 (2026'-0") to RB-5 (2047'-6"): Open floor grating. Cable trays do not pass through the open grating (Ref. Callaway Drawings E-2R2521, E-2R2421); however, cable trays 5U2J33 and 5U2J35 are routed below and parallel to the grating at El. 2047'-2" (approximately 6" below grating).

Requirements:

For non-inerted containments, Section 4.2.3.4 of NFPA 805 requires separation of required cables and equipment of redundant success paths by a horizontal distance of more than 20 feet with no intervening combustible materials or fire hazards.

Basis for Request:

For the five (5) configurations identified above, the following items serve as the basis for acceptance of this deviation:

- Cables in conduit do not contribute to fire growth and spread, and are therefore not considered to be intervening combustible materials.
- Separation of cables is in accordance with Callaway Plant Engineering Design Guide EE-002, "Electrical Separation Criteria." The criteria for separation of this engineering design guide meets the separation requirements of IEEE std. 383-1974 and Reg. Guide

1.75. The separation requirements are outlined in Callaway drawing E-2R8900. For all areas except the cable spreading areas, which applies to RB-1, the minimum separation required between redundant cable trays is 1) 3 ft. horizontally between side rails, and 2) 5 ft. vertically between bottom of top tray side rail and top of side rail of bottom tray.

- Cables in the subject intervening trays within the Reactor Building are IEEE-383 qualified thermoset cables and are not susceptible to electrically generated fires (i.e. capable of self-ignition). In addition, IEEE-383 qualified thermoset cables have a very low heat release rate and flame spread rating. Per NUREG/CR-6850, cable trays containing thermoset cables have a rate of flame spread equal to 3.54 ft/hr. Using this cable tray flame spread value, the expected elapsed duration for a cable tray containing thermoset cables to spread 10 feet (i.e. fire starts in the middle of the required 20-foot separation distance) and 20 feet is approx. 2.8 hours and 5.6 hours, respectively. Due to the installed detection and, in certain areas, suppression systems and the presence of an onsite Fire Brigade, a fire of this duration is not considered credible.
- Although some cable trays routed below open grating are not fully enclosed throughout their entire route, trays have been determined to be fully enclosed at intermittent locations based on review of Callaway conduit drawings. The intermittent cable tray enclosures provide additional fire resistance to fire growth and flame spread to adjacent analysis areas.
- Fire spread through penetrations from RB-1 to RB-2 is not expected due to the presence of thermoset cable insulation and the multi-foot thick solid bio-shield wall through which a cable tray would penetrate (which will also act as a radiant energy barrier).
- Manual suppression is provided by the plant fire brigade. During normal operation, extinguishers and/or hose for hose stations are located just outside the containment personnel hatch. The hose station locations are such that all accessible areas of the reactor building are adequately covered by at least one hose stream.
- The Fire PRA analysis for RB-1 was performed using detailed fire modeling in accordance with approved methodologies (e.g., NUREG/CR-6850). The analysis was performed for the entire RB-1 compartment without credit for the five analysis area boundaries mentioned above. In addition, the analysis considered exposed cables in trays as intervening combustibles, and modeled these using the appropriate empirical rule sets for fire growth, propagation, and spread. The Fire PRA quantification for RB-1, including detailed fire modeling, indicates that there is insignificant impact due to any fire spread across the analysis area boundaries as it resulted in a total Fire CDF of less than 5E-7 and a total Fire LERF of less than 1E-8, which is considered to be low risk.

Acceptance Criteria Evaluation:

Nuclear Safety and Radiological Release Performance Criteria:

Based on the analysis presented above, the presence of intervening combustibles in the form of exposed cable trays across RB-1 analysis area boundaries will not have an adverse impact on the Nuclear Safety Performance Criteria for a fire in the Reactor Building.

The radiological release performance criteria will be met considering the potential fires that could occur within the Reactor Building as described in Section 4.4 and Attachment E.

Safety Margin and Defense-in-Depth:

Based on the analysis presented above, the presence of intervening combustibles in the form of exposed cable trays across RB-1 analysis area boundaries will not have an adverse impact on the Nuclear Safety Performance Criteria for a fire in the Reactor Building, and therefore, the safety margin inherent in the analysis has been preserved.

The identified RB-1 analysis area boundary configurations and associated postulated fire locations do not directly compromise automatic and manual fire suppression systems or the Nuclear Safety Performance Criteria, and therefore, defense-in-depth is maintained.

Conclusion:

Based on the analysis presented above, approval is requested for a deviation from the deterministic requirements of NFPA 805 Section 4.2.3.4 for Reactor Building (RB) configurations where the twenty or more feet of separation is present but the areas are not free of intervening combustibles. The configurations will not have an adverse impact on satisfying the Nuclear Safety Performance Criteria.